APPENDIX IV

RADIO-ISOTOPIC ANALYSIS OF RADIOACTIVE MATERIAL IN THE DORF STRUCTURE BEFORE DECOMMISSIONING

INTRODUCTION

This report documents information of the amount and type of radioactive material that will be present in the structure and building of the Diamond Ordnance Radiation Facility after removal of the reactor fuel in the spring of 1978. Such information is required for decommissioning plans and must be supplied to the Army Reactor Committee for Health and Safety (ARCHS) prior to their approval of such plans. The information is also needed by the waste-disposal area directorate who must budget for specific volumes and radioactive levels. Finally, the isotopic composition of the radioactive waste is necessary for labeling containers at the time of shipment.

The first section of this report is a summary for those who need only the final results on type, location and amount of residual radioactivity. Section two describes the investigative procedures, discusses the possible sources of radioactivity and the properties of the radioactive isotopes found. Graphs of isotopic analyses and calculations, which convert detector response to specific activities, are included in this section. The second section also provides the detailed calculations of volumes, weights and total radioactivity in the various sections of DORF. The final section contains recommendations based on things discovered during this study.

SUMMARY

The radioactivity that will remain at DORF after the fuel removal in the spring of 1978 has been carefully estimated based on criteria, measurements and necessary assumptions documented in this report. A concise summary of that radioactivity is given in Table I. The most predominant radioactive isotopes in the concrete are cobalt-60 and europium-152 and -154. The most predominant isotopes in lead are antimony-124 and silver-110 . The wood and steel (mainly in the lead-shield hoist) are not very radioactive and are easy to dispose of. The aluminum itself is almost non-radioactive but there is a radioactive Phenoline

liner which tends to stick to the aluminum. Its radioactivity comes from cobalt-60 and zinc-65. All of these radioactive isotopes have half-lives in excess of 60 days.

TABLE I. Summary[1/] of total radioactivity to be expected from materials in the DORF structure after core removal.

Material	Mass (lbs)	Volume (ft^3)	Radioactivity (millicuries)
CONCRETE	82,170	412	36.24
(If whole plug door included).	(170,050)	(850)	(36.24)
LEAD	55,753	112	13.34
ALUMINUM	2,288	15	75.71
WOOD	34,944	1344	0.33
STEEL	2,662	5.5	0.03
GRAND TOTAL	177,817 lbs (89 tons)	1889 ft^3	0.126 Curies

[1/] This represents a summation of the values given in Table X.

IDENTIFICATION OF THE RADIOACTIVITY

Isotope Identification:

The principal method of identification was gamma-radiation spectroscopy with a germanium lithium-drifted detector, or Ge(Li) crystal. The crystal is housed inside a very low-activity-lead cave lined with wood. Numerous background analyses confirm that for photons with energies greater than 140 keV, samples with low activities (two to three times background) can be successfully analyzed for specific photon energies. A plot of a multichannel analyzer spectrum of the background is given in Fig. 1. The principal higher energy peaks in the background spectrum are the 511-keV gammas associated with annihilation radiation and the 1461-keV peak from ^{40}K, a radioactive isotope which is found naturally in almost all "non-radioactive" materials.

The method of analyses provides for very good resolution of the photon energies in the range 140 keV to 2500 keV at approximately 9.7 keV per channel of 256 total channels. The electronic equipment is sufficiently stable over counting periods of 50,000 seconds to permit energy assignment within two percent. Graphs of the gamma spectra of the various materials investigated are shown. (See Fig. 2 through 5).

The method does not provide for the ultimate in accuracy for determining specific activity. The crystal efficiency (disintegrations per count as a function of energy) can only be accurately assigned for a well-defined geometry. The samples in the present situation varied in size and shape. Therefore, they were suspended above the crystal so that their centers of mass were approximately three centimeters from the active volume of the

detector and efficiencies were determined with calibrated point sources. The error associated with this procedure is estimated to be no greater than 50%, based on a volume integration of point-source response at points in space representative of the sample size. For the task at hand such accuracy is sufficient.

Rational of Sample Selection

The job was to identify the radioactive content and quantity of materials that will have to be removed from the DORF site so that it can be certified, by post-decommissioning radioactive survey, as an unrestricted area for possible public use. This survey, to be conducted by the Army Environmental Health Agency (AEHA), must be accomplished prior to any filling, sealing or burying activities. This presented two problems. How can we identify the radioactivity in presently inaccessible areas, such as below the reactor pool, before the reactor fuel and higher-level radioactive structures have been removed? What amount of material will have to be removed from walls and floors to reach an acceptable AEHA level?

The first problem was attacked as follows. Representative samples of all the material types are accessible in the exposure-room area. Because of the significantly larger thermal-neutron cross sections of materials and the fact that the DORF-TRIGA reactor is zirconium-hydride moderated and water-cooled reactor, the thermal component of the spectrum is the dominate source of induced radioactivity. As will be discussed later, the predominance of radioactive europium confirms this. Therefore, isotopic analyzes of exposure room samples are representative of those in presently inaccessible

areas. Furthermore, with facility dosimetry data for the various locations, we can estimate the residual radioactivity in remote locations with significantly different flux exposure levels.

The second problem of how much material to remove is more complex because we do not have good guidance on the amount of radioactivity in volume that can remain. NRC Regulation 1.86, the current guide, clearly specifies levels for removal surface contamination but is, at best, vague on volume activity and how to detect it. The criteria set for the analysis in this report are as follows:

(1) Once the reactor support structure has been removed there will be no high-level radioactive waste remaining in the DORF structure. Our analyses confirm this.

(2) Based on existing allowable concentrations of radioactive materials in water and a specific activity proportional to material density, we can set an allowable specific activity of 2×10^{-5} microcuries per gram as the maximum permissible concentration of radionuclides in water when it is known that Sr 90, I 129, (I 125, I 126, I 131, Table II only), Pb 210, Ra 226, Ra 228, Cm 248, and Cf 254 are not present.* Since the density of water is one g/cm^3 and there are 28317 cm^3/ft^3, 2×10^{-5} $\mu Ci/g$ corresponds to 0.57 $\mu Ci/ft^3$ of water.

(3) It is assumed that the radioactivity is distributed in the material to be removed in proportion to the incident thermal fluence (flux-time product) and attenuated exponentially according to thermal-neutron relaxation lengths, (i.e., the inverse of microscopic removal cross sections for broad beams). Half-life decay is taken into consideration for the period until spring 1978. Therefore, the depth of material to be removed, D in centimeters, is determined

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* 10 CFR20, note to Appendix B

by relative fluence level at the surface, ϕ/ϕ_o, and

$$D = L \ln \frac{\phi/\phi_o \times A}{0.57 \mu Ci/ft^3} \qquad (1)$$

where A is the activity in $\mu Ci/ft^3$ estimated from this study. The values of relaxation length are given in Table II.

TABLE II. Material densities and relaxation lengths, L

Material	Density	Relaxation Length
Concrete	2.35 g/cm^3	1.6 cm
Lead	11.0 g/cm^3	4.2 cm
Wood	0.42 g/cm^3	2.9 cm

Measured Radioactivity

Samples taken from the DORF exposure room were concrete, wood, aluminum, lead and a tar-paper-like liner installed between the aluminum pool tank and the concrete pool base. Although the aluminum itself has very little residual radioactivity (less than $8 \times 10^{-6} \mu Ci/gm$ for the sections counted), the Phenoline paper (i.e., the tar-paper liner) has the highest specific activity of all the materials examined. Since this liner tends to stick to the aluminum, for all practical purposes the aluminum tank exhibits this activity.

Tables IV through IX give a breakdown of the isotopic composition of the radioactivity in the various samples. Tables IV and V are composed of several additional pages that serve as detailed examples of the methods of analyses and are self explanatory when reference is made to the graphs of the multichannel-analyzer output. Figs. 1 through 5 are the multichannel-analyzer gamma spectra for the various types of samples. The energy of the photopeaks is related to

TABLE III. GAMMA* SPECIFIC ACTIVITY AND THE NUMBER OF MICROCURIES PER UNIT OF MATERIAL FOR VARIOUS RADIOACTIVE MATERIALS FROM THE DORF EXPOSURE ROOM

	Type of Material	Location in Exposure Room	Specific Activity (d/s·g)	Activity per unit of material
1.	PHENOLINE PAPER	On aluminum tank near exposure room end of pool	636	3.6 μCi/sq ft
2.	CONCRETE	From front part of room about 4 feet from reactor	78	140 $\mu Ci/ft^3$
3.	CONCRETE	Very near reactor at exposure room end of tank	141	252 $\mu Ci/ft^3$
4.	LEAD	From curtain above the movable lead shield	72	0.62 μCi/lb
5.	LEAD	From brick in middle of the movable lead shield	205	1.30 μCi/lb
6.	WOOD	From very near reactor and concrete sample #3, above.	1.3	0.40 $\mu Ci/ft^3$

*From gross beta plus gamma analyses, the beta-to-gamma activity of all these different materials is approximately 1.8.

spectra for the various types of samples. The energy of the photopeaks is related to start the channel number (abcissa) by the following equation:

$$E(keV) = (\text{channel} + 2.5) \times 9.69 \pm 2\% \quad (2)$$

For clarification, the gamma-ray peaks are indentified by isotope and their energies in keV (and in parenthsis) are given for most of the peaks.

The specific activity (d/s·g) for each measured sample is compared in Table III. This table also provides the number of microcuries per unit most practical for that type of material. This latter information is used in Table X to determine the total radioactivity in the volumes of radioactive materials at DORF.

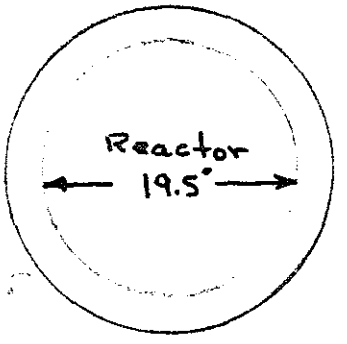
CONCRETE EXCAVATION

The maximum observed concrete-volume activity was 252 μCi/ft³. If we assume that the maximum fluence at any location is 100 times larger than at this monitored exposure room location, Eq. (1) ... the excavation-depth formula ... requires

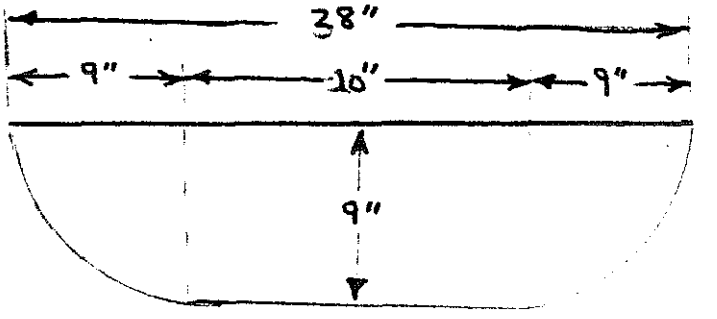
$$D = 1.6 \ln\left(\frac{100 \times 252}{0.57}\right) = 17.1 \text{ cm or } 6.7 \text{ inches}$$

of course, fast neutrons will also penetrate and thermalize but the factor of 100 is already very conservative so excavation depths of 9 to 10 inches in the immediate vicinity of the reactor and at the end positions of the pool should adequately remove the radioactivity. It is assumed that because of H₂O attenuation that the flux will be decreased by a factor of 400 at the 38" wide extremes.

Excavation below reactor in Pos #3 & radiography position:



Excavation diameter
38"



Volume: $\pi r_1^2 l + \frac{1}{4}(2\pi^2 R r_2^2)$

$\pi\left[(10^2)9 + \frac{1}{2}\left(\pi(14.5)(9^2)\right)\right]$

$V = (2827 + 5795) \div 1728 = \mathbf{2.52\ ft^3}$

Activity in concrete below Pos. 3: Activity will not exceed measured maximum of 252 μCi/ft³ by more than a factor of two. Assign 500 μCi/ft³.

$$5.0 \times 500 = 2500\ \mu Ci$$

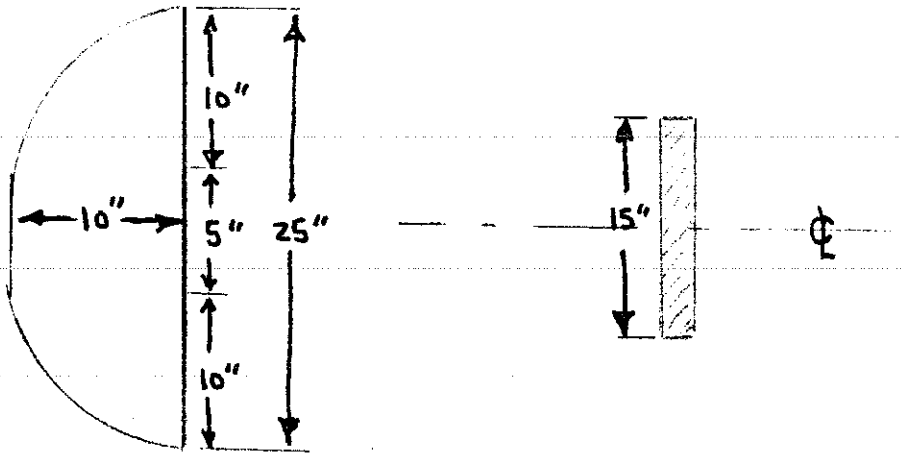
Activity in concrete below radiography position:

Assign 250 μCi/ft³ (conservative)

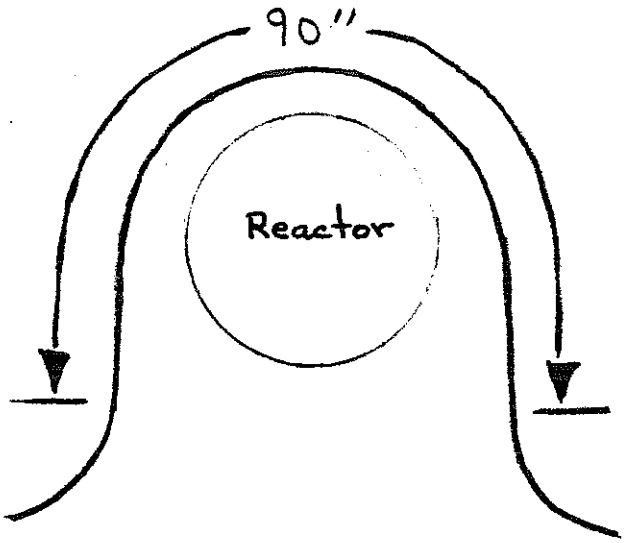
$$5.00 \times 250_9 = 1250\ \mu Ci$$

Excavation from tank-support walls in Pos. #1 & #3:
(active fuel length is 15 inches)

FOR 1 & 3:



Volume for Pos.3 end of pool:



Volume $L \times (\frac{1}{2}\pi r^2 + t \times d)$

$90\left(\frac{\pi}{2}(10^2) + 5 \times 10\right)$

$V = 90(157 + 50) \div 1728 = 10.8\ ft$

Volume for Pos.1 end of pool: Because tank projects into the exposure room, L for pos.1 in taken as 40" rather than 90" and $V = \frac{4}{9}(10.8) = 4.8\ ft^3$

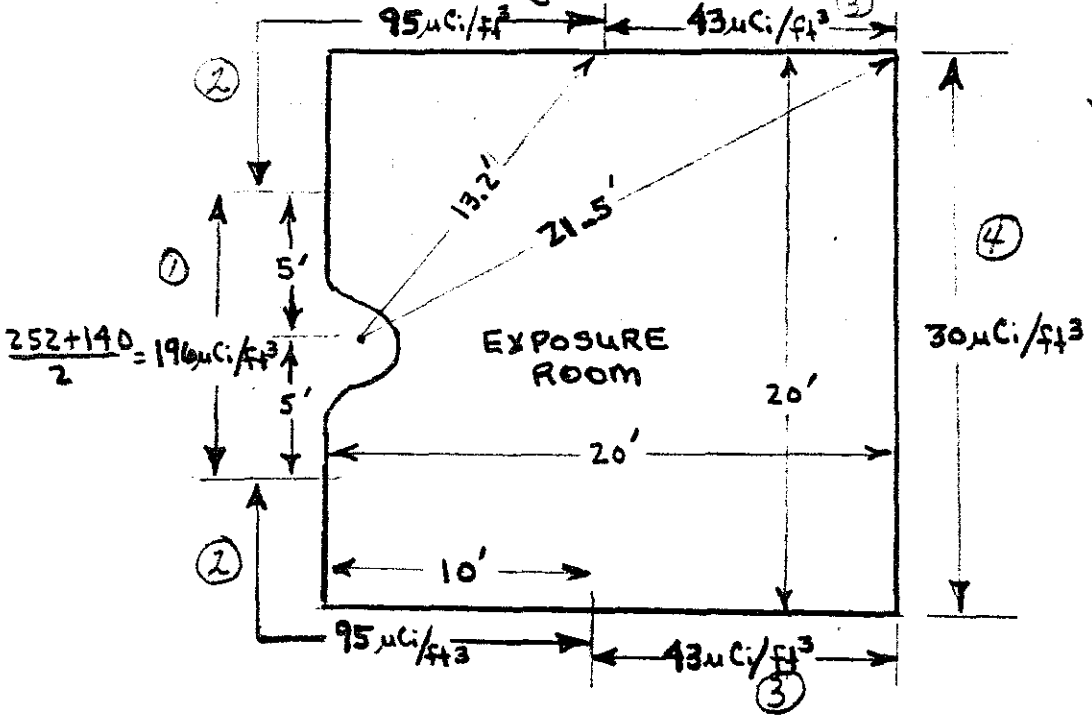
Activity for wall excavation in Pos. 3 & 1: Consistant with conservative estimate for concrete below reactor in Pos.3, assign 500 μCi/ft³

Activity for Pos.3 end: $500 \times 10.8 = 5400\ \mu Ci$

Activity for Pos.1 end: $500 \times 4.8 = 2400\ \mu Ci$

Excavation of concrete from walls, floor and ceiling of the DORF exposure room.

The following diagram shows the estimated volume activity in the concrete of the exposure room walls. Because of thermalization of fast flux by the wood and scatter from the walls, the effective thermal flux is assumed to fall off more nearly as $1/R$ than as $1/R^2$.



room heighth 8 feet.

Excavation depths:

No.	D (inches)	Assign Activity/ft³
1	3.7 "	196 μCi/ft³
2	3.2	95 "
3	2.7	43 "
4	2.5	30 "

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE IV. CONCRETE SAMPLE FROM "FRONT" OF EXPOSURE ROOM

$M = 2.47\ g$ $T = 54,252$ seconds $(MT = 1.34 \times 10^5)$

ISOTOPE	Half Life $T_{1/2}$	Activity dis/s.g.	% of Total Activity
^{60}Co	5.27y	31.66	40.55 %
^{152}Eu - ^{154}Eu	12.2y - 16y	30.15	38.62 "
^{46}Sc	84 days	5.44	6.97 "
^{134}Cs	2.07 y.	5.06	6.48 "
^{65}Zn	245 d	2.29	2.93 "
^{182}Ta	115 d	1.19	1.52 "
^{124}Sb	60 d	0.63	0.81 "
* Annihilation	—	1.65	2.12 "
		TOTAL: 78.07 d/s.g	(100.00)

Nat. Bkg in concrete (^{226}Ra series)

^{214}Bi 1622y 0.84 d/s.g.

(i.e., concrete is 93 times its own natural background)

Activity per cubic foot of concrete

$(12 \times 2.54)^3 = 28317\ cm^3/ft^3$

ρ concrete $= 2.35\ g/cm^3$

$\Big\}\ 6.65 \times 10^4\ g/ft^3$

$$\text{Activity} = \frac{6.65 \times 10^4 \times 78.07}{3.7 \times 10^{10}\ d/s \cdot Ci} = 1.40 \times 10^{-4}\ Ci \text{ or } \boxed{140\ \mu Ci/ft^3}$$

Estimate per "ml" of H_2O: $(78.07)/3.7 \times 10^{10} \times 2.35 = 8.98 \times 10^{-10}$ or $8.98 \times 10^{-4}\ \mu Ci/$ml

* Note: All activities except annihilation radiation (511 kev) were determined from sample with background substracted.

WEIGHT = 2.47 g (BKG. SUBSTRACTED FOR EQUAL TIME: 54,252 seconds)

* Indicated more that one isotope has a peak in this channel 9.71 ± 0.05 kev/ch + 2.5

SOURCE ISOTOPE & $T_{1/2}$	kev FROM ENDF/B-VI	CHANNEL	PEAK AREA GROSS COUNTS	COMPTON UNDER PEAK GROSS COUNTS	PEAK NET COUNTS	e c/d	Total
^{152}Eu–^{154}Eu (2.2y – 16year)	122 ^{152} ^{154}	10	23500	3364	20136	2.50×10^{-2}	805440
	245 & 248 ^{154}	23	4889	2030	2859	1.16×10^{-2}	246465
	344 ^{152}	33	8441	1756	6685	8.10×10^{-3}	825308
	411 ^{154} ^{152}	40	1894	1370	524	6.66×10^{-3}	78797
	488 ^{152}	47.8	1658	1340	318	5.58×10^{-3}	56989
	689 ^{154}	68.8	1366	1230	136	3.90×10^{-3}	34872
	720 & 723.1 ^{154}	72 *	1357	1240	117	3.63×10^{-3}	32231
	779 ^{152}	78	2562	1250	1313	3.35×10^{-3}	391940
	867 & 873 ^{154}	87	1788	1325	463	2.95×10^{-3}	156949
	966 ^{152}	97	2457	1150	1307	2.66×10^{-3}	491353
	1006 ^{154}	101	1130	940	190	2.57×10^{-3}	73930
	~~1092 ^{152}~~	~~110~~	~~1728~~	~~975~~	~~753~~	~~2.45×10^{-3}~~	~~307347~~ delete because of Zi
	1277 ^{154}	129	321	135	186	1.95×10^{-3}	95385
	1412 ^{152}	143	1220	50	1170	1.74×10^{-3}	672913
	1457 ^{152}	146.7	73	32	41	1.67×10^{-3}	24551
	1528 ^{152}	155	59	18	41	1.60×10^{-3}	25625
	1595 ^{154}	162.5	60	17	43	1.52×10^{-3}	28289
						TOTAL =	4.046×10^{6}
^{60}Co (5.27 years)	1173.1	119	4857	589	4268	2.11×10^{-3}	2.023×10^{6}
	1332.4	135	4187	126	4061	1.86×10^{-3}	2.219×10^{6}
						TOTAL =	4.242×10^{6}
^{134}Cs (2.07 year)	563 & 569	56	1580	1300	280	4.72×10^{-3}	5.932×10^{4}
	605	60 *	1612	1293	319	4.36×10^{-3}	7.317 "
	795 & 802	80	2101	1300	801	3.25×10^{-3}	29.646 "
	1040	105 *	870	840	30	2.50×10^{-3}	1.200 "
	1168	119 *	(370)	120	250	2.15×10^{-3}	11.628 "
	1365	139	370	60	310	1.81×10^{-3}	17.127 "
						TOTAL =	6.785×10^{5}
^{65}Zn (245 day)	1115.4	110 *	1728	975	753	2.45×10^{-3}	307347
						TOTAL	3.073×10^{5}

TABLE IV-A. (continued)

ISOTOPE	Rev ENERGY	channel	Gross	B.kg	Net	∈ c/d	Total
^{46}Sc (84 days)	889.4	89 *	2491	1400	1091	2.90×10^{-3}	3.762×10^{5}
	1120.3	113	2330	(1550)	780	2.21×10^{-3}	3.529×10^{5}
						TOTAL	7.291×10^{5}
^{124}Sb (60 day)	602.6	60 *			200	4.40×10^{-3}	45.454×10^{3}
	646	65 *			25	4.17×10^{-3}	5.995 "
	722.8	72 *			25	3.63×10^{-3}	6.887 "
	968	97 *	1120	1100	20	2.66×10^{-3}	7.518 "
	1045	105 *	870	853	17	2.56×10^{-3}	6.800 "
	1691	172	41	28	13	1.43×10^{-3}	9.091 "
	2085	209	25	22	3	1.15×10^{-3}	2.609 "
						TOTAL =	8.435×10^{4}
^{182}Ta (115 day)	100.3	8	4577	4102	475	3.10×10^{-2}	15.322×10^{3}
	152.9&156	13.6	3054	2700	354	1.42×10^{-2}	24.930 "
	198.4	17.5 *	2620	2450	170	1.50×10^{-2}	11.333 "
	223	20.5	2348	2210	138	1.35×10^{-2}	10.222 "
	1121	113 *	870	810	60	2.21×10^{-3}	27.149 "
	1189	119	(710)	650	60	2.09×10^{-3}	28.708 "
	1221.6	123	337	252	85	2.05×10^{-3}	41.463 "
						TOTAL =	1.591×10^{5}
Annihilation	511	50	4100	3100	1,000	4.50×10^{-3}	2.22×10^{5}
						TOTAL	2.22×10^{5}

Additional analysis of "natural background in concrete:

^{214}Bi (1622 years*)

* from ^{226}Ra series

1731	177	36	27	9	1.39×10^{-3}	6.475×10^{3}
1767	180	39	29	10	1.24×10^{-3}	8.064 "
2119	217	23	14	9	1.13×10^{-3}	7.965 "
2204	226	32	23	9	1.08×10^{-3}	8.333 "
2293	234	28	12	16	1.03×10^{-3}	15.534 "
2497	250	36	25	11	9.40×10^{-4}	11.702 "
			Actual about 2x this →		TOTAL	5.807×10^{4}

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE V. TAR-PAPER* SAMPLE FROM FRONT OF EXPOSURE ROOM

* CALLED PHENOLINE COATING

$M = 3.69\ g$ $T = 20{,}183\ s$ $(MT = 7.45 \times 10^4)$

ISOTOPE	HALF-LIFE $T_{1/2}$	Activity dis/s.g.	% of Total Activity
^{65}Zn	245 day	185.2	29.1 %
^{60}Co	5.27 years	363.8	57.2 %
^{152}Eu	12.2 year	86.8	13.7 %
		TOTAL: 635.8 d/s.g	100.0 %

From gross counts, this sample was $423 \div 11 = 38.5$ times more active than the natural background in the Ge(Li) crystal cave. NOTE: It is $635.8/78.1 = 8.1$ times more radioactive than the concrete from the front part of the room.

Activity per square foot of paper: (assuming a 1/8-inch thickness)

$(12 \times 2.54)^2 \times 0.125 \times 2.54 = 295\ cm^3$ per square foot

$\rho = 0.7\ g/cm^3$

Activity $\dfrac{635.8\ d/s \cdot g \times 206.5\ g}{3.7 \times 10^{10}\ d/s \cdot Ci} =$ **3.55 μCi/sq. foot**

TABLE V-A. TAR-PAPER SAMPLE FROM POS #1 END OF ALUMINUM TANK (IN EXPOSURE ROOM).

TIME = 20,183 sec.

423 samples/liter
11 liters

M = 3.6875 grams

20183 000004 002000

	1	2	3	4	5	6	7	8	9
000000									
020183	000004	065213	016877	013805	014030	014273	013571	013285	013616
000010									
028361	011699	011727	011230	011878	010214	010054	009864	009489	009693
000020									
009953	009577	009525	013469	008729	008657	008524	008433	008402	008066
000030									
008237	008377	007927	019884	007346	007546	007144	006983	006859	006789
000040									
007620	006823	006536	007492	006603	006469	006496	006540	006508	006670
000050									
011253	006750	006323	006526	006497	006456	007013	006515	006682	006515
000060									
007831	006514	006433	006741	006702	006907	006732	006904	007032	006989
000070									
006667	006954	007301	007106	007305	007392	007381	007520	009923	007635
000080									
008622	008056	008383	008368	008992	008968	009128	010116	009901	012019
000090									
010301	009768	008668	008394	008263	008354	008177	009248	006260	005804
000100									
005583	005638	004958	004642	004680	004475	004422	004334	004357	004701
000110									
005809	004901	009792	037642	002790	002421	002438	002329	007215	028849
000120									
001547	001386	001187	001140	000822	000787	000644	000546	000530	000822
000130									
000530	000744	000621	000538	000840	027671	000259	000203	000178	000189
000140									
000194	000178	000271	002053	000157	000170	000186	000296	000244	000122
000150									
000132	000137	000141	000132	000135	000193	000148	000144	000137	000121
000160									
000126	000137	000143	000130	000094	000106	000124	000107	000116	000104
000170									
000114	000116	000092	000124	000107	000099	000118	000084	000094	000087
000180									
000094	000080	000089	000100	000093	000115	000099	000102	000088	000080
000190									
000082	000107	000089	000090	000090	000090	000095	000092	000078	000081
000200									
000093	000089	000093	000095	000085	000116	000081	000082	000079	000068
000210									
000090	000064	000065	000062	000069	000081	000053	000076	000058	000074
000220									
000066	000064	000068	000052	000064	000073	000059	000076	000054	000062
000230									
000072	000062	000049	000079	000056	000047	000041	000045	000043	000034
000240									
000033	000027	000023	000019	000027	000016	000012	000013	000012	000016
000250									
000013	000016	000009	000011	000016	000019				

ISOTOPIC ANALYSES OF TAR-PAPER SAMPLE FROM ON TANK WALL AT EXPOSURE-ROOM END AT APPROXIMATELY CORE CENTERLINE.

COUNT TIME = 20,183 sec.; WEIGHT = 3.6875 g

SOURCE ISOTOPE/ $T_{1/2}$	ENERGY* (keV)	CHANNEL #	PEAK GROSS COUNTS	COMPTON BKG. COUNTS	PEAK NET COUNTS	XTAL EFF. (C/d)	TOTAL d
^{65}Zn (245 day)	1115.4	113	37642	3700	33942	2.45×10^{-3}	1.38×10^{7}
						TOTAL=	(1.38×10^{7})
^{60}Co (5.27 year)	1173	119	28249	1700	26549	2.11×10^{-3}	1.26×10^{7}
	1332	135	27671	750	26921	1.86×10^{-3}	1.45×10^{7}
						TOTAL=	(2.71×10^{7})
^{152}Eu (12.2 year)	121.8	10	48361	12600	35761	2.50×10^{-2}	1.43×10^{6}
	244.6	23	13469	9140	4329	1.16×10^{-2}	0.30 "
	344.2	33	19884	7700	12184	8.10×10^{-3}	1.50 "
	411.0	40	7690	6790	900	6.65×10^{-3}	0.14 "
	443.9	43	7492	6500	992	6.60×10^{-3}	0.15 "
	778.6	78	9932	7550	2382	3.35×10^{-3}	0.71 "
	964.1	97	9248	7206	2048	2.66×10^{-3}	0.77 "
	1086.0	110	5809	4800	1009	2.45×10^{-3}	0.41 "
	1407.5	143	2053	210	1843	1.74×10^{-3}	1.06 "
						TOTAL=	(6.97×10^{6})

* Energy is approximately (±1.5%) 9.69 x(CHANNEL+2.4) keV/channel

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE VI. CONCRETE SAMPLES FROM UNDER TANK IN EXPOSURE ROOM

$M = 11.63\ g$ $T = 40{,}000$ sec $(MT = 4.65 \times 10^5)$

ISOTOPE	HALF-LIFE $T_{1/2}$	Activity dis/s·g	% of Total Activity
^{60}Co	5.27 y	63.2	44.5 %
^{152}Eu-^{154}Eu	12.2y – 16y	55.4	39.0 "
^{46}Sc	84 day	10.5	7.4 "
^{65}Zn	245 day	4.0	2.8 "
^{182}Ta	115 day	3.1	2.2 "
^{134}Cs	2.07 y	1.3	0.9 "
^{124}Sb	60 day	0.2	1.2 "
Annihilation	—	2.8	2.0 "
	TOTAL	140.5 d/s·g	(100.0)

Note: This concrete sample is 1.8 times more radioactive than the sample from the front of the room.

Activity per cubic foot of concrete:

Density: 6.65×10^4 g/ft3

$$\text{Activity} = \frac{6.65 \times 10^4 \times 140.5}{3.7 \times 10^{10}\ d/s \cdot Ci} = 2.52 \times 10^{-4}\ Ci \text{ or } \boxed{252\ \mu Ci/ft^3}$$

Estimate per "ml" of H_2O: $1.62 \times 10^{-3}\ \mu Ci/g$

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE VII. LEAD FROM BRICK IN MOVABLE LEAD SHIELD

$M = 2.531$ g $T = 40{,}000$ sec $(MT = 1.01 \times 10^5)$

ISOTOPE	HALF-LIFE ($T_{1/2}$)	ACTIVITY d/s.g	% of TOTAL ACTIVITY
^{124}Sb	60 days	181.8	71.0%
^{110m}Ag	253 day	67.3	26.3%
^{152}Eu-^{154}Eu	12.2y - 16y	1.5	0.6%
Annihilation	—	5.4	2.1%
	TOTAL	256.0 d/s.g	(100.0%)

Activity per pound of lead:
454 g/lb

$$\text{Activity} = \frac{454 \times 256}{3.7 \times 10^{10}} = 3.14 \times 10^{-6} \text{ or } 3.14\ \mu\text{Ci/lb}$$

NOTE: About 44% of this radioactivity will decay by spring 1978 because of the ^{124}Sb contribution. Therefore Eff. Activity = 1.76 μCi/lb.
Futhermore, the top and bottom quarter sections of the lead shield should have activities more like that of the lead curtain (Eff. Act = 0.62 μCi/lb) so a better value for the whole shield is $0.5 \times 1.76 + 0.4 \times 0.62 =$ 1.30 μCi/lb

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE VIII. LEAD FROM CURTAIN

M = 0.602 T = 40,000 sec ($MT = 2.41 \times 10^4$)

ISOTOPE	HALF-LIFE $T_{1/2}$	ACTIVITY (d/s.g)	% of TOTAL ACTIVITY
^{124}Sb	60 days	4.96	62.1%
^{110m}Ag	253 days	2.28	28.5%
^{46}Sc	84 days	0.45	5.6%
^{152}Eu-^{154}Eu	12.2y-16y	0.05	0.7%
Annihilation	———	0.25	3.1%
		7.99 d/s.g	(100.0)

Activity per pound of lead:
(454 g/lb)

$$\text{Activity} = \frac{454 \times 7.99}{3.7 \times 10^{10}} = 0.98 \times 10^{-7} = 1.0\,\mu\text{Ci/lb}$$

NOTE: About 38% of this activity will have decayed by spring 1978.
Therefore: Eff Activity = **0.62 μCi/lb**

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE IX. WOOD FROM FRONT OF EXPOSURE ROOM

$m = 36.58$ g $T = 8967$ sec. $(mT = 3.28 \times 10^5)$

ISOTOPE	HALF-LIFE $T_{1/2}$	ACTIVITY d/s.g	% of TOTAL ACTIVITY
^{152}Eu-^{154}Eu	12.2y-16y	0.588	46.7 %
^{60}Co	5.27 y	0.534	42.4 %
^{46}Sc	84 day	0.055	4.4 "
^{129m}Te	41 day	0.040	3.2 "
^{127m}Te	110 day	0.026	2.1 "
Annihilation	——	0.017	1.2 "
	TOTAL	1.26 d/s.g	(100.0)

Activity per cubic foot of wood

$$28317 \text{ cm}^3/\text{ft}^3 \times 0.42 \text{ g/cm}^3 = 1.189 \times 10^4 \text{ g/ft}^3$$

Activity: $\frac{1.26 \times 1.189 \times 10^4}{3.7 \times 10^{10}} = 4.04 \times 10^{-7}$ or $0.40 \mu Ci/ft^3$

less than 10% of this activity will decay by spring 1978.

NOTE: From WRAMC Health Physics survey of this sample we find there is a β/γ ratio of approximately two.

TABLE X. Summary of radioactive material types, locations, weights, volumes and activity levels at DORF after reactor-core removal.

MATERIAL	LOCATION AT DORF	DENSITY (lbs/ft³)	WEIGHT (lbs)	VOLUME (ft³)	ACTIVITY LEVEL	TOTAL ACTIVITY OF VOLUME (millicuries)
1. CONCRETE	PLUG DOOR TO EXPOSE ROOM	200	87,880	439	———	Negligible
	• (Front 12 inches of door)	"	9,750	49	140 μCi/ft³	6.86
2. "	• WALLS, CEILING & FLOOR OF EXPOSURE ROOM	"	70,920	355	(see separate section for detail)	27.49
3. "	• BELOW TANK IN ROOM	"	1,500	7.5	252 μCi/ft³	1.89
4. LEAD	• MOVABLE SHIELD (51"x72"x2")	708	3,010	4.3	1.30 μCi/lb	3.91
5. "	• CURTAIN ABOVE SHIELD	"	1,847	2.6	0.62 μCi/lb	1.15
6. "	• ON FRONT WALL EXPOSURE RM.	"	10,896	15.4	0.40 μCi/lb	4.36
7. "	• LEAD SHOT IN POOL DOORS	"	40,000	(90)	0.10 μCi/lb	4.00
8. ALUMINUM	• ALUMINUM POOL DOORS	168.5	1,305	7.8	0.10 μCi/ft³	0.01
9. "	• CORE SHROUD	"	168	1.0	75 mCi/ft³	75.00
10. "	• TANK WALL: 5/16" thick with pheneline liner	"	815	6.2	(3.6 μCi/ft²)*	0.70
11. WOOD (dry fir)	• FRONT EXPOSURE ROOM	26	3,328	128	0.40 μCi/ft³	0.05
12. "	• SIDES EXPOSURE ROOM	"	6,656	256	0.25 μCi/ft³	0.06
13. "	• BACK EXPOSURE ROOM	"	4,160	160	0.15 μCi/ft³	0.02
14. "	• FLOOR & CEILING " "	"	20,800	800	0.25 μCi/ft³	0.20
15. STEEL	• SHIELD HOIST	484	2,662	5.5	0.01 μCi/lb	0.03

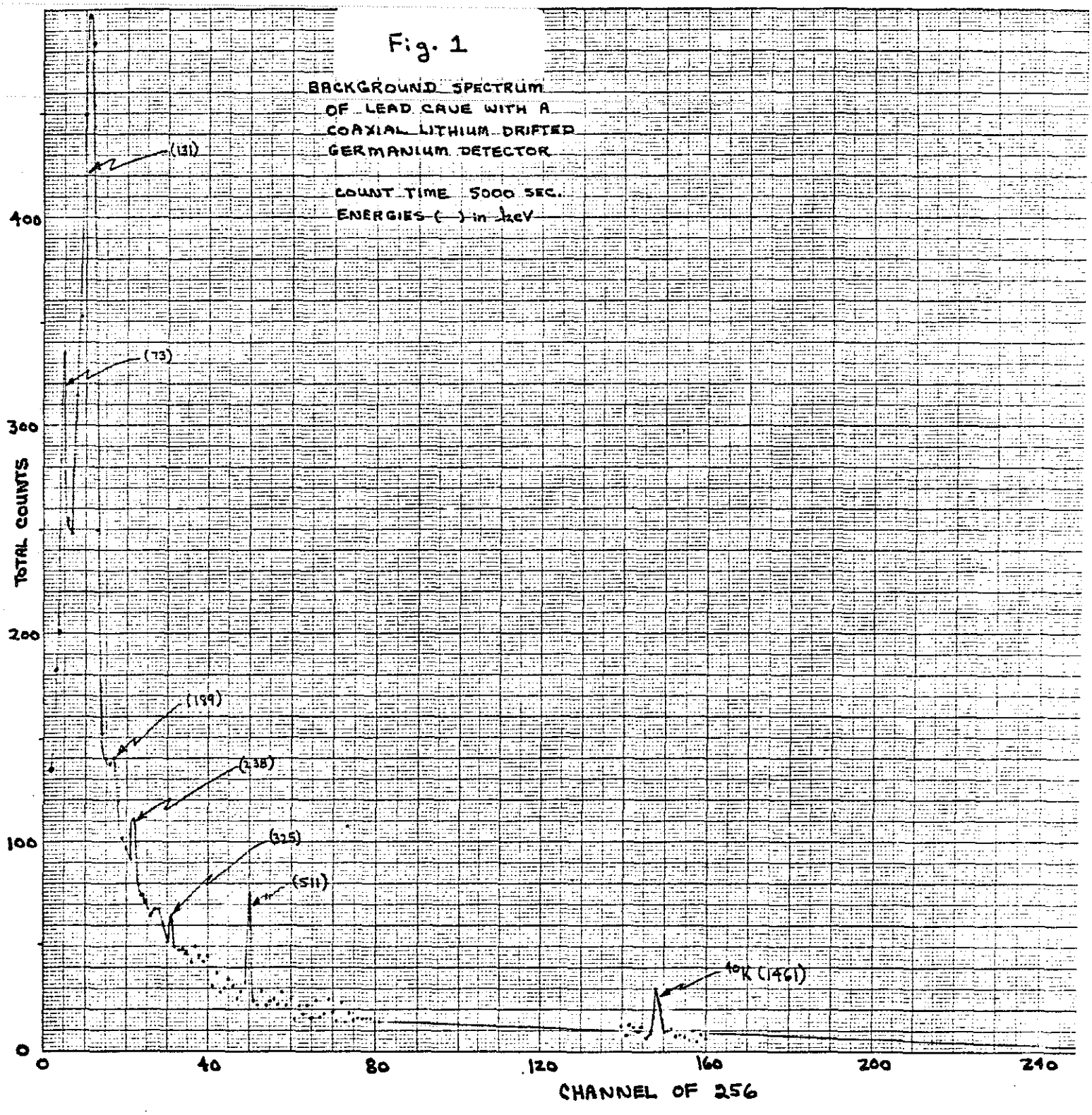


Fig. 1

BACKGROUND SPECTRUM OF LEAD CAVE WITH A COAXIAL LITHIUM DRIFTED GERMANIUM DETECTOR

COUNT TIME 5000 SEC.
ENERGIES ( ) in keV

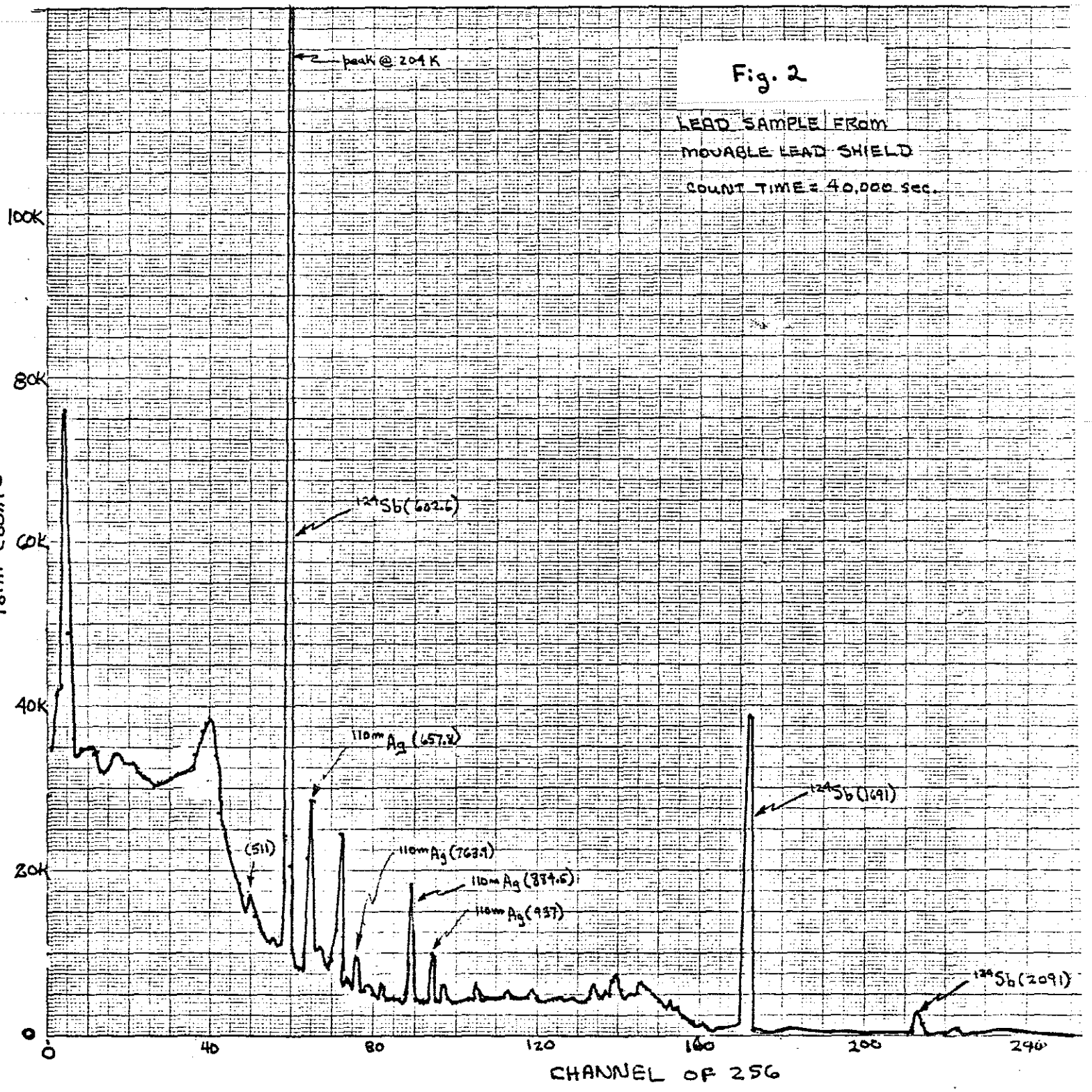


Fig. 2
LEAD SAMPLE FROM
MOVABLE LEAD SHIELD
COUNT TIME = 40,000 sec.
2 — peak @ 204 K
100K
80K
60K
40K
20K
0
Total Counts
124Sb (602.6)
110m Ag (657.8)
(511)
110m Ag (763.9)
110m Ag (884.6)
110m Ag (937)
124Sb (1691)
124Sb (2091)
0
40
80
120
160
200
240
CHANNEL OF 256

Fig. 3

TAR-PAPER SAMPLE FROM TANK WALL AT EXPOSURE ROOM END ℄ CORE ℄ HEIGHTH

COUNT TIME = 20,183 sec.

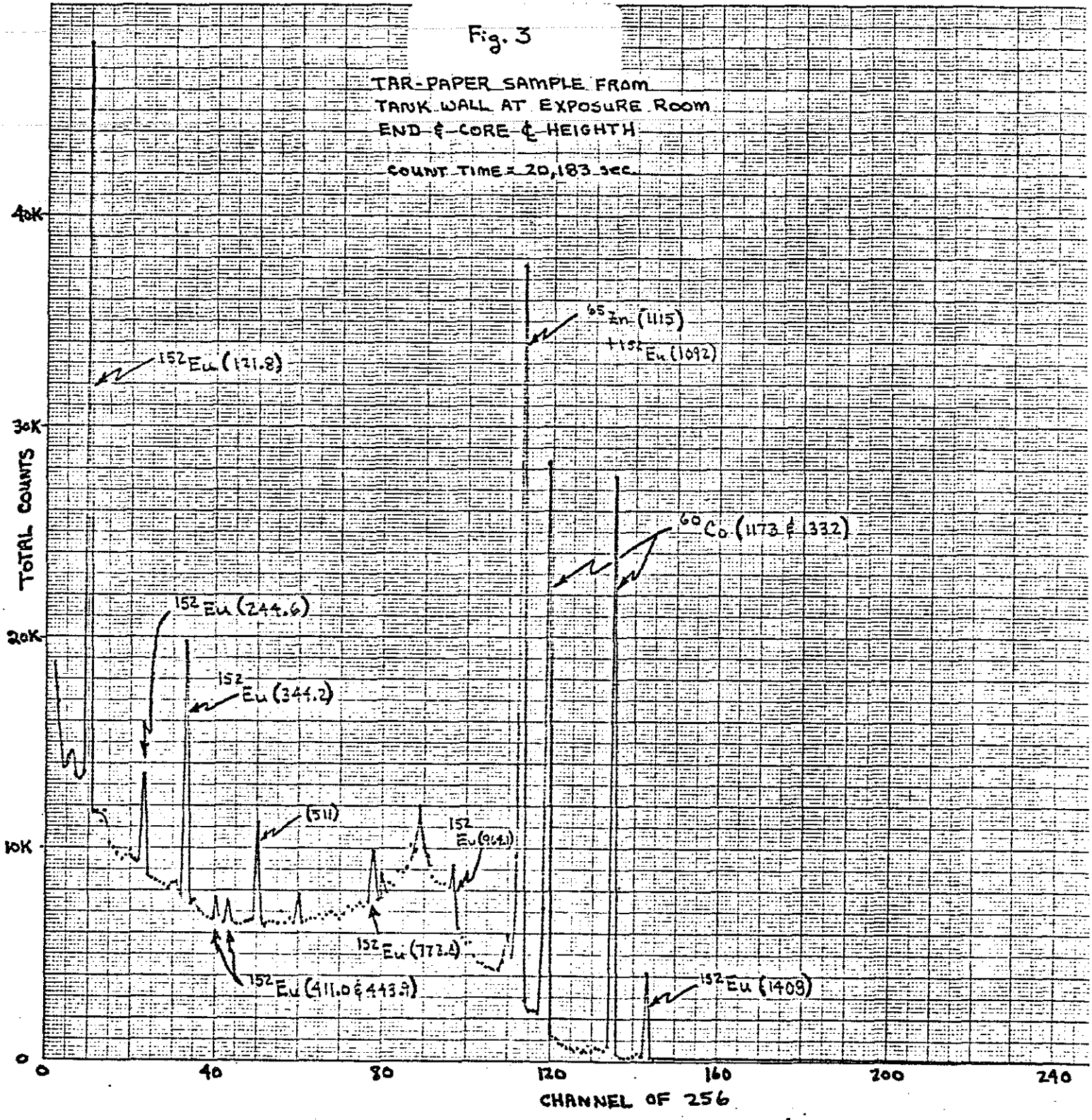
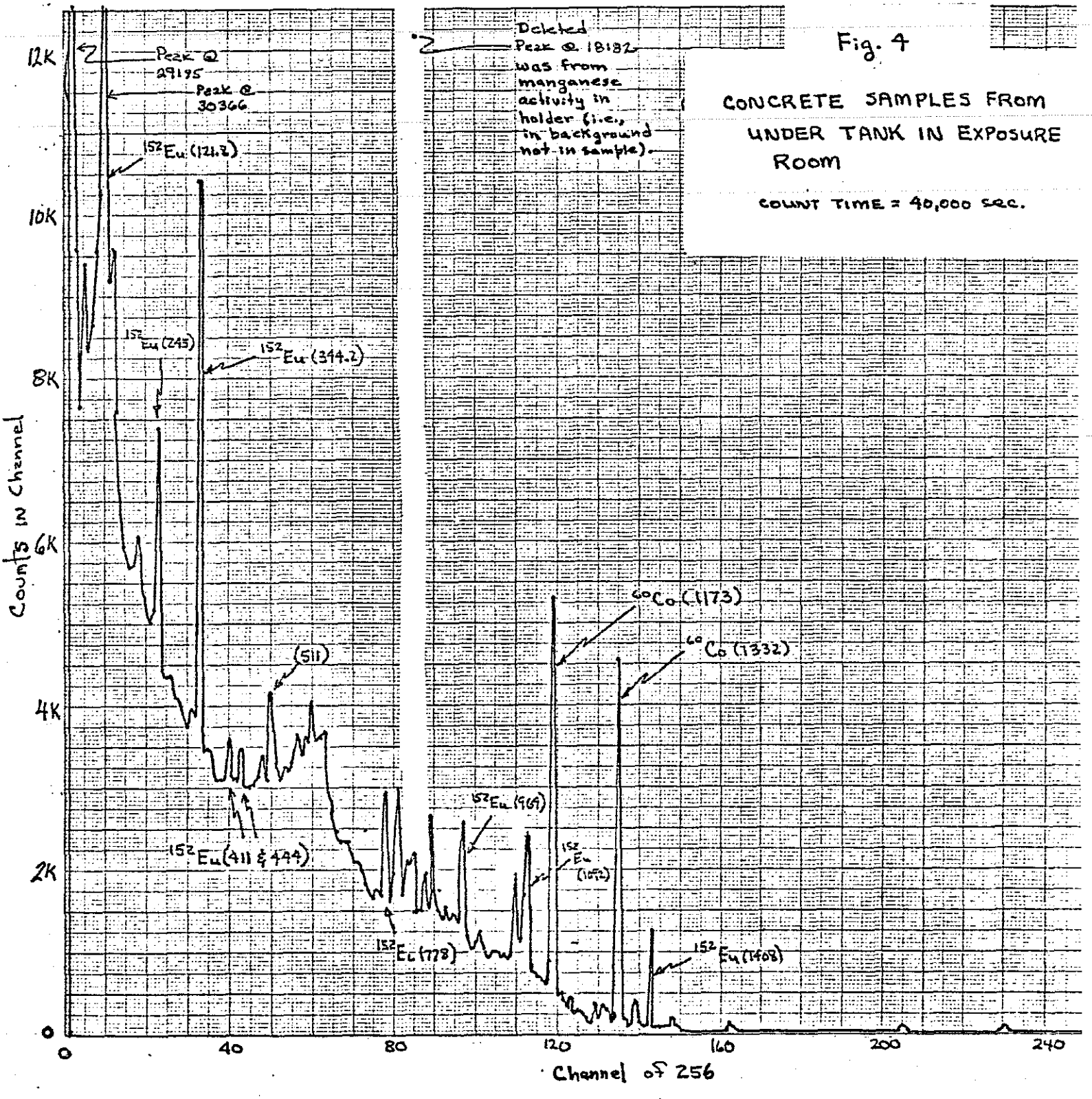


Fig. 4

CONCRETE SAMPLES FROM UNDER TANK IN EXPOSURE ROOM

COUNT TIME = 40,000 sec.



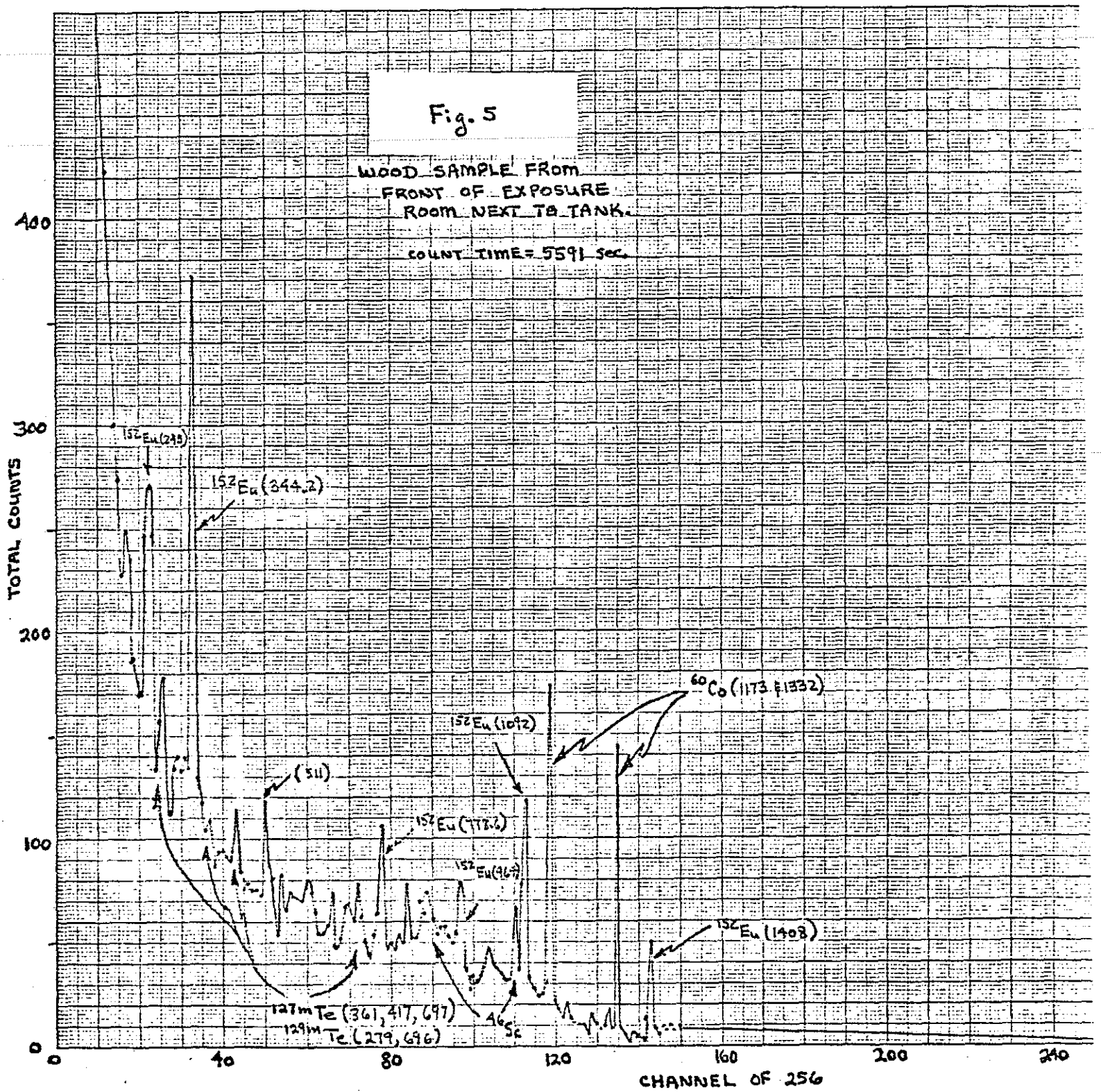


Fig. 5
WOOD SAMPLE FROM FRONT OF EXPOSURE ROOM NEXT TO TANK.
COUNT TIME = 5591 sec.
TOTAL COUNTS
CHANNEL OF 256
0
100
200
300
400
0
40
80
120
160
200
240
152Eu(245)
152Eu(344.2)
60Co(1173 & 1332)
152Eu(1092)
(511)
152Eu(778.2)
152Eu(964)
152Eu(1408)
127mTe(361, 417, 697)
129mTe(279, 696)
46Sc

CONCLUSIONS & RECOMMENDATIONS

The principal conclusion from this study is that once the reactor-grid support structure has been removed there is very little radioactivity remaining at DORF. Unfortunately the levels are definitely above background, but only by factors of several hundred, and the radioactivity is mainly distributed throughout concrete walls and floors. Deep excavations will not be necessary. However, this is of little consequence if one still has to remove several inch-thick layers from large areas. This is the situation in the exposure room. In fact, the exposure-room decontamination is by far the major problem and several possible methods of attack come to mind.

(1) Excavate and remove the three 5000-gallon waste-water holding tanks, cut off part of the tops and use them as shipping containers for the radioactive debris from DORF. For example, the wood has suffered radiation damage and dry rot so that it crumbles rather easily. It is a big volume (1200 ft^3) but relatively light in weight so it can easily be tossed or shoveled into the tanks and they could then be closure welded for shipment. There will also be much dust, dirt, paper and small concrete chips of radioactive waste, all of which could be put into the tanks.

(2) Mechanically cut, DO NOT CUT WITH A TORCH, the aluminum because of the radioactive "tar-paper" liner which could easily catch on fire and produce contaminated smoke. However by reference to the excavation-of-concrete details in this report, the places where the aluminum liner will be radioactive are easily identified. It does not appear that the liner will produce a problem in other than these areas.

(3) Thought should be given to the possibility of transferring some of the lead to AFRRI or APRF because its radioactivity is really not a serious hazard and these facilities need it for shielding in neutron fields. This could save a few dollars on transportation and disposal costs.

(4) Survey activities are going to be a problem because there just isn't much activity to survey right now. For example, depending on what is going to be done with the exposure room, it may not be necessary to excavate concrete from the rear wall of the room. In any event, thought should be given to how much the survey reading "from the rear wall only", before excavation, must be decreased by material removal to provide an "acceptable" survey level. In view of the expense to breakup and ship concrete, it is prudent to be practical about sealing up or burying very small, but detectable, amounts of radioactivity.

(5) Almost all of the materials exhibit one or two predominant and characteristic photopeaks. Therefore, survey activities could be determined by a sodium-iodide scintillation detector. It is suggested that a portable detector with a 3/4-inch-thick cylindrical lead shield around the sides would be practical. Calibration could be accomplished in a crude, but adequate, manner by measuring the response of a variety of sources simultaneously positioned over a square-meter plane area behind about 1/4-inch thick aluminum. This approximates the following situation. The dose rate to tissue in rads per hour in an infinite medium, of density ρ, uniformly contaminated by a gamma emitter, of energy E (MeV), is

$$2.12\ EC/\rho \qquad (3)$$

where C is in microcuries per cm^3. At the surface, the dose rate is about one half of this and for air a one-centimeter-from-the-surface survey is an adequate representation of the surface rate. By then surveying the "calibration setup at one meter" and correcting for $1/R^2$ to one centimeter, one can estimate the rads per hour efficiency of the scintillation detector. A variety of sources, repositioned should be used and the results averaged.