Met-Ed / GPU

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Writer's Direct Dial Number

December 15, 1980
TLL 666

TMI Program Office
ATTN: Mr. Lake Barrett, Deputy Director
c/o Three Mile Island Nuclear Station
Middletown, Pennsylvania 17057

Three Mile Island Nuclear Station Unit II (TMI-2)
Operating License No. DPR-73
Docket No. 50-320

METHODOLOGY FOR CALCULATION OF INTEGRATED DOSE

Dear Sir:

Forwarded herewith is the initial report on integrated dose calculations for the EPICOR-II prefilters. This document details the calculational methods employed by GPU as well as calculations performed to date on Prefilter 29.

Our transmittal to you, dated December 3, 1980 (TLL-634), provided our initial evaluation of the potential corrosion effects to the EPICOR-II resin liners. This report supplements that transmittal such that radiation effects to resin integrity might be evaluated. The calculations performed were benchmarked to radiation surveys taken on the liner when removed from services. Other data, as described in the report, include influent and effluent concentrations, which could not be reported by gamma analyses due to limitations in the counting technique. The calculational techniques defined in this transmittal require the distinction between parent and daughter radionuclides which exist in a state of equilibrium in order to accurately determine the type and quantity of energy transfer per unit time. A typical example of this phenomenon is the Cesium 137 - Barium 137m equilibrium state. The curie loadings in this document differ from previous transmittals to you because of the inclusion of the daughter radioisotopes.

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We will continue to update this report as calculations are performed and more information as well as the chemical/metallurgical evaluations which are being performed by GPU, become available.

Should additional information be required, please contact me or my staff.

Sincerely,

/s/ G. K. Hovey

G. K. Hovey,
Vice-President and
Director, TMI-2

GKH:JAD:djb
Enclosure
cc: B. J. Snyder

METHODOLOGY AND CALCULATION OF INTEGRATED DOSE TO EPICOR-II PREFILTER 29

METHODOLOGY AND CALCULATION

OF

INTEGRATED DOSE

TO

EPICOR-II PREFILTERS

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REVISION 1

DECEMBER 15, 1980

METHODOLOGY AND CALCULATION OF INTEGRATED DOSE TO EPICOR-II PREFILTER 29

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RADIOLOGICAL ANALYSIS GROUP

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1.0 Purpose and Summary

1.1 Purpose

The purpose of this report is to describe the methodology of calculating the integrated dose to EPICOR-II ion exchanger material; specifically, in the prefilters of that system since the prefilters have been used to remove the most highly concentrated radionuclides of the contaminated waste water from TMI-2. Results of calculations for Prefilter 29 are included in order that the postulated effects of radiolytic decomposition might be evaluated for the actual resins and chemical additives used. The effects of radiolytic decompostion are being evaluated by the Materials Technology Section of GPU.

1.2 Summary

EPICOR-II became operational on October 22, 1979, to process intermediate level waste water,i.e., waste containing leas than 100 μCi/ml. It was recognized at the time of operation that radiolytic effects from fission products could have a direct impact on resin stability and the degradation impact on the integrity of the carbon steel liner itself. Operational limitations were set during the month of July, 1979, and were agreed upon by all parties involved. (Ref.1) Specifically, curie loading limitations were set at 1300 curies for the prefilters. However, radiation surveys taken on the initial spent liners during changeout indicated that curie loading was taking place in a relatively narrow band of the resin. In April, 1980, Radiological Analysis Group was requested to study

the impact of this occurence. Included in this report is a realistic analysis of energy absorption by the resin, i.e., neither overly conservative nor overly optimistic. Conservatism vs. non-conservatism is discussed in the text. In cases where actual information and/or data was unavailable, logical assumptions were made based on accepted experimental data from the industry.

Comparison of radiation survey data to shielding calculations indicate that approximately 80% of the total curies deposited in a prefilter takes place in a relatively narrow band of the resin in the liner, i.e., the cation layer. This specific activity results in a localized integrated dose of approximately 1.0×10^8 RADS within 1 year of removal from service, for those prefilters loaded to the maximum curie loading. The calculations in this document are for Prefilter 29 only, and are based on information available. Updates to this calculation as well as integrated dose to other prefilters will be made as more detailed information becomes available.

2.0 Source Term Generation

Calculation of an energy absorption rate is a simple and straightforward procedure once the volumetric source term, hereafter referred to as S_v, is defined. Depending upon the degree of accuracy desired, the generation of S_v varies from an estimation to a detailed fission product transport/chemical balance calculation. The source term generated for this calculation includes all available data with respect to the fission products identified to be present in water processed by EPICOR-II as well as fission product transport calculations for those isotopes known to be present but not identified because of limitations in the counting techniques.

2.1 Fission Product Activity In The Core

The fission product activity in the Unit 2 core has been calculated by the digital computer code ORIGEN (Version 2) which B&W has modified and named LOR-2. The fission product inventory was calculated using the appropriate power history and boron concentration history for Unit 2. The initial conditions were 83,000 Kg Uranium with an average enrichment of 2.63 wt.% U-235, and final burnup of 3175 MWD/MTU (95 EFPD). This data was used as input to the computer code RADTRAN, (Ref.2) a time dependent fission product transport code, to determine the fission product release from the core. The fission products were decayed to the time of cladding failure (~130 min.) and then allowed to leach out of the fuel as a function of time. The escape rate coefficients represent the fraction of the activity in the fuel that is released, per unit time, from the fuel

matrix. Values of these coefficients are derived from experimental data for most elements. (Ref. 3-7) The values of the escape rate coefficients for elements of lesser importance were estimated based on the chemical similarity with elements for which the escape rate coefficient was experimentally determined. The escape rate coefficients used in this analysis are given in Table 2.1.

2.2 Fission Product Activity In Coolant

The general rate equation for the inventory of a radioactive nuclide, N, in the coolant is:

$$\frac{dNc}{dt} = \alpha N_f + f\lambda' N_c' - \lambda N_c - \beta N_c - \gamma N_c$$

Where:

N_c = inventory of radioactive nuclide in coolant

N_f = inventory of same radioactive nuclide in the fuel

α = escape rate coefficient of N_c (sec^{-1})

f = fraction of precursor which decays to N_c

$\lambda' N_c'$ = activity of precursor which decay to N_c

λ = decay constant of N_c (sec^{-1})

β = removal rate coefficient of N_c by intentional removal from primary system (sec^{-1}) determined by:

$$\frac{R_L}{M_c}$$

Where:

R_L = letdown rate (Kg/sec)

M_c = mass of primary system (Kg)

γ = removal rate coefficient of N_c by discharge out of the PORV determined by:

$$\frac{R_p}{M_c}$$

where:

R_p = discharge rate out of PORV (Kg/sec)

M_c = mass of primary system (Kg)

The sequence of events was used to determine the times at which the PORV was open, and the Moody Critical Flow Tables were used to determine the actual flow out of the PORV.

Letdown flow began at 5 minutes into the accident. The assumption was made that letdown flow was directed to the Reactor Coolant Bleed Holdup Tanks. This assumption is based on the knowledge that this action can be initiated by an operator in the control room without having an auxiliary operator enter the Auxiliary Building, and normal valve alignment is for the letdown to be directed to the RC Bleed Holdup Tanks.

Prior to the accident, an entry in the auxiliary operator's logbook dated March 27, 1979 indicated that the Miscellaneous Waste Holdup Tank (MWHT) contained approximately 15,300 gal. Usable volume of this tank is ~19,600 gallons. The same entry noted that the Auxiliary Building Sump Tank was full (~3000 gal)

and that the Auxiliary Building sump was empty. Sample data from the Reactor Coolant Bleed Tanks were used to determine the amount of dilution due to water in the tanks prior to core damage, and all isotopes were adjusted accordingly.

It is recognized that water transfers were made during the month of April, 1979, into and out of the bleed tanks. However, these transfers were between tanks in the Auxiliary Building, which were contaminated with essentially the same ratios of fission products, varying only in concentration. It is therefore felt that the concentrations shown in this document are not significantly different from what was actually in the bleed tanks. Such transfers would have introduced oxygenated water into the bleed tanks along with other possible chemicals.

TABLE 2.1

ESCAPE RATE COEFFICIENTS

Element	Coefficient (sec^{-1})
Kr	6.5×10^{-8}
Xe	6.5×10^{-8}
Sr	1.0×10^{-11}
Br	1.3×10^{-3}
I	1.3×10^{-3}
Cs	1.3×10^{-3}
Rb	1.3×10^{-3}
Cd	1.0×10^{-9}
In	1.0×10^{-9}
Sn	1.0×10^{-9}
Mo	2.0×10^{-9}
Nb	2.0×10^{-9}
Tc	2.0×10^{-9}
Ru	2.0×10^{-9}
Rh	2.0×10^{-9}
Ba	1.0×10^{-11}
Y	1.6×10^{-12}
Ce	1.6×10^{-12}
Pr	1.6×10^{-12}
Zr	1.6×10^{-12}
Nd	1.6×10^{-12}
Sm	1.6×10^{-12}

TABLE 2.2

RCBT 'B' ISOTOPIC INVENTORY AS OF MAY 3, 1980

Nuclide	ORNL Reported Activity (μci/ml)	EXXON* Nuclear (μci/ml)	Calculated Activity (μCi/ml)
Cs-137	37.57	36.47	35.1
Cs-134	6.80	6.75	6.60
Ba-137m			33.34(+1)
Sr-90	.30		.4
Sr-89	.17		0.17
Y-90			.4
Ru-106			1.58(-3)
Ru-103			6.83(-5)
Rh-106			1.58(-3)
Rh-103m			6.83(-5)
Sb-125			5.10(-3)
Nb-95			2.90(-3)
Zr-95			1.46(-3)
Co-60			1.31(-4)
Co-58			5.56(-5)
Mn-54			3.19(-4)
Ag-110m			4.14(-5)
Ce-144			3.48(-3)
Ce-141			9.78(-6)
Sn-113			4.11(-5)
In-113m			4.11(-5)
Pr-144			3.48(-3)
Pu	.012ppb		
U	13.0 ppb		

*Preliminary Results, as of November 25, 1980

3.0 Material Balance and Curie Deposition Calculation

Concentration products were calculated for possible chemical compounds using elemental concentrations obtained from Oak Ridge National Laboratory analyses of Reactor Coolant Bleed Tank (RCBT) 'B' sample (Ref. 8). These concentration products were compared to the corresponding solubility products, available in the literature, and very few of the possible compounds showed any possibility for precipitation. Since the systems connected to the primary system are kept oxygen free to alleviate corrosion problems associated with oxidation, the assumption was made that oxide formation was negligible, as was radioactive-oxyanion formation. Thus, the nuclides present in RCBT 'B' were assumed to be primarily in cationic form and were favored for cation exchange.

3.1 Curie Deposition

The data obtained from effluent samples taken during processing was used to determine radionuclide effluent concentrations. These effluent sample results were averaged to determine the mean isotopic concentration during processing. The difference was taken between the influent and effluent concentrations of known isotopes to determine an approximation of the radionuclides deposited per ml of processed water. Total curies deposited was calculated by multiplying by the total volume of water processed. Table 3.1 is a tabulation of the curies deposited in the cation region of PF-29 by the above method. Also shown is the activity in the chemically bound water after dewatering.

3.2 Determination of Specific Activity

Attachment I containes the radiation survey data for those prefilters that were surveyed, together with the band of principle loading. The instrument was fixed at 9" from the liner, and radiation readings were recorded at 6" vertical intervals. Cation exchange resins of the highly acidic sulfonated polystyrene type have swelling ranges from 3-to-15% (3% for highly cross-linked resins to 15% for low cross-linked resins). If a low degree of cross linkage is assumed, there is greater swelling capability and thus a lower specific activity after exchange. However, a low degree of cross linkage results in less radiation stability. Alternatively, if a high degree of cross linkage exists, a smaller swelled volume occurs and results in higher specific activity after exchange but enhanced radiation stability. This may be graphically demonstrated by Figures 3.1, 3.2 and 3.3. These figures are not meant to be interpreted quantitatively, they are only provided to help the reader understand the relationships between these characteristics. On a dry-weight basis the exchange capacity of a resin is inversely proportional to its cross linkage (Ref. 10).

For the calculation of specific activity loading in PF-29, an average swelling over the total resin volume was assumed to be about 25%. This average swelling of 25% was used to find the volume of cation in which distribution of the cationic radionuclides took place. The actual distribution of the radionuclides could conceivably take place in a smaller volume. It should be noted that this assumption is not particularly

conservative in one sense, i.e., it results in lower specific activity [illegible]e calculated loading was compared to the radiation survey taken on PF-29 when it was removed from service. Comparison of survey data to deposited curies indicate that 80% of the principle gamma emitting nuclides were distributed in the layer identified as the band of principle curie loading. It is assumed that this is a cation layer with even distribution. Further, the assumption was made that 40% water retention took place in the resin volume, due to chemical/mechanical bonding. The volumetric source term generated was then used as input to the computer code ANISNBW (Ref.9), a one dimensional discrete ordinates transport code, to calculate the gamma flux.

3.3 Calculation of Gamma Flux

The gamma source term as calculated above was used as input to the ANISNBW code as volumetric source strength in 12 energy groups in units of photons/cm^3·sec. Further, material properties were used which approximate a mixture of wetted resins. Also, the physical dimensions of the liner and its location in the Long Term Storage Module was modeled to account for those photons being scattered back into the liner from the wall of the storage cell. The contribution from liners in adjacent cells was neglected. The weight percent compositions of the dewatered cation as well as the energy group source terms used are shown in Tables 3.2 and 3.3 respectively. It was assumed that a sulfonated polystyrene resin (Na-form) was used. This assumption has some effect on the calculation of gamma flux, in that a higher density material would increase energy absorption, and thus increases the absorbed dose.

A total of 51 intervals in the source volume (from the centerline to the edge of the liner) were used to increase the accuracy of the calculation. The geometry of the liner in the cell is shown in Figure 3.4. ANISNEW was then used to calculate the total photon flux at each of the 51 intervals. The code uses energy dependent scattering and absorption cross sections for each element in the material specified according to its weight percentage. The output of ANISNEW is a photon flux at each interval broken down into discrete energy groups.

TABLE 3.1

CALCULATED CURIE DEPOSITION IN PF-29
(5/9/80)

Isotope	Cation Deposition (Ci)	Activity In Chemically Bound Water (Ci)
Co-58	5.84 E-4	7.12 E-6
Co-60	2.05 E-3	1.68 E-5
Sr-89	1.49 E+1	7.77 E-2
Sr-90	4.88 E+1	2.56 E-1
Y-90	4.88 E+1	2.56 E-1
Mn-54	6.70 E-3	4.08 E-5
Zr-95	3.14 E-2	1.87 E-4
Nb-95	6.31 E-2	3.71 E-4
Ru-103	3.71 E-4	8.74 E-6
Rh-103m*	3.71 E-4	8.74 E-6
Ru-106	4.91 E-3	2.02 E-4
Rh-106	4.91 E-3	2.02 E-4
Ag-110m*	0	5.30 E-6
Sn-113	0	5.26 E-6
In-113m*	0	5.26 E-6
Sb-125	5.84 E-2	6.53 E-4
Te-125m*	1.34 E-2	1.50 E-4
Cs-134	1.91 E+2	9.97 E-1
Cs-137	8.64 E+2	4.49 E0
Ba-137m*	8.16 E+2	4.26 E0
Ce-141	1.31 E-4	1.25 E-6
Ce-144	4.64 E-2	4.45 E-4
Pr-144	4.64 E-2	4.45 E-4

*Note: International Commission on Radiation Units and Measurements Report, ICRU REPORT 19, July 1, 1971 specifically defines an isomeric transition as a nuclear transformation, and is measured in units of curies.

TABLE 3.2

WEIGHT % COMPOSITION OF DEWATERED RESIN

Element	Weight Percentage
H	7.4
O	44.8
C	37.2
S	6.2
Na	4.4

Assumption: Resin used is sulfonated polystyrene (Na-form)

Density: 0.705 gm/cm^3

TABLE 3.3

ENERGY GROUP SOURCE TERM

Group	Average Energy (MeV)	Photons/cm^3·sec
29	2.25	2.47×10^1
30	1.83	4.05×10^2
31	1.495	5.33×10^5
32	1.165	4.42×10^5
33	.9	1.43×10^6
34	.7	8.53×10^7
35	.5	1.27×10^6
36	.35	5.13×10^1
37	.25	1.05×10^1
38	.15	2.28×10^3
39	.075	1.84×10^2
40	.03	3.16×10^3

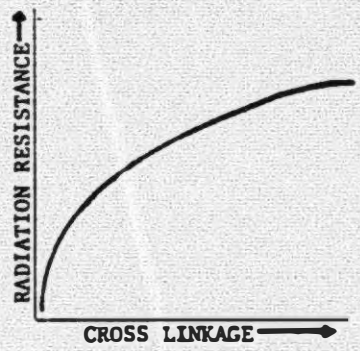


Figure 3.1

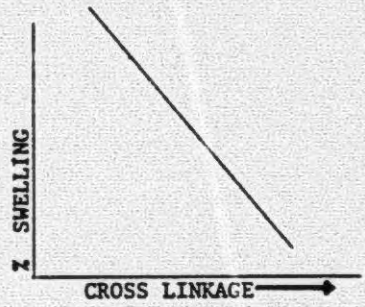


Figure 3.2

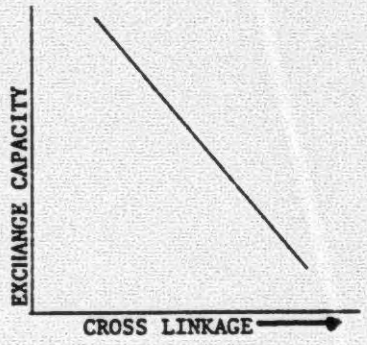
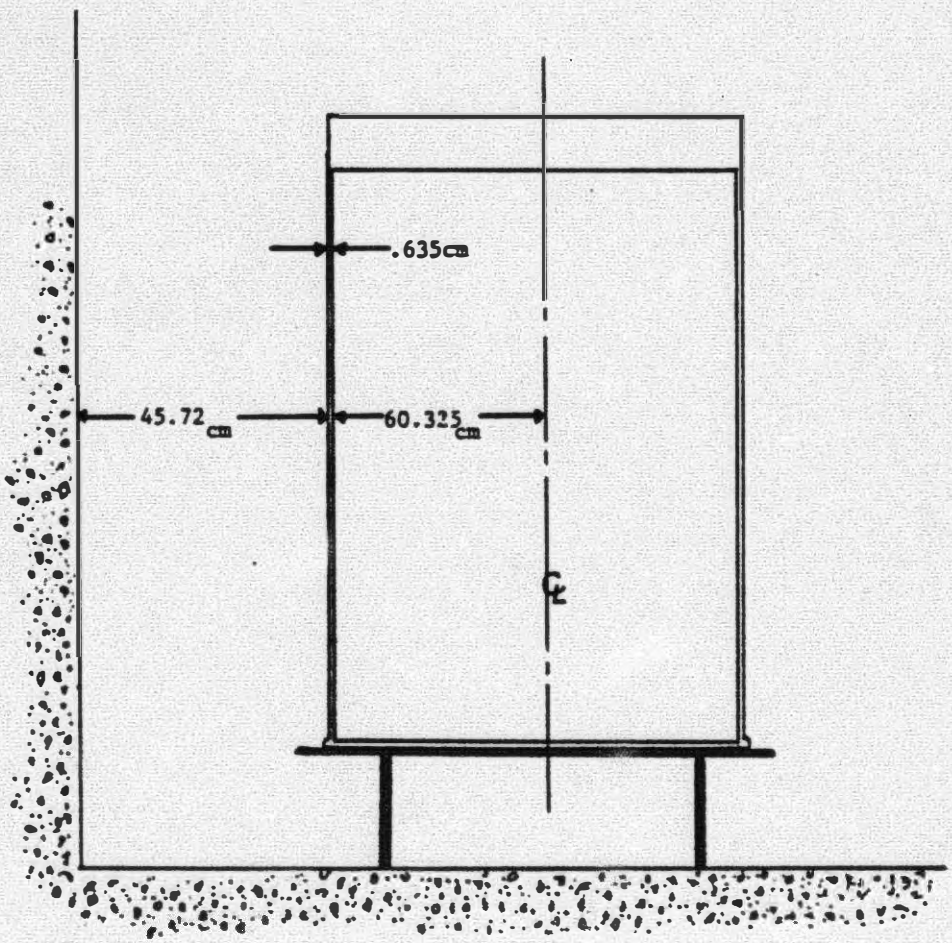


Figure 3.3

FIGURE 3.4

DIMENSIONS OF LINER IN STORAGE CELL



4.0 Energy Absorption Calculations

4.1 Gamma Energy Absorption

The photon flux calculated by ANISNBW at each interval was multiplied by the appropriate energy absorption coefficient at that particular energy and summed over all energies to determine the energy deposition rate at that interval. This calculation was performed by the computer code INTDOSE, developed for this analysis. Expressed mathematically,

$$\dot{D} = \sum_{i=1}^{n} 1.602 \times 10^{-8} \left[\mu_{e/\rho}\right] \phi_i E_\gamma$$

where $\dot{D}$ is the dose rate in rads/sec at the interval, and n = number of energy groups. A more detailed explanation is given below. The gamma dose rate as a function of radius in the cation bed is shown in Figure 4.1.

4.2 Beta Energy Absorption

The dose rate from beta emitting nuclides calculated to be in the cation bed was calculated essentially the same as that for the gamma flux, with the assumption that the beta particle could not escape the media in which it was located. The average energy of the beta particle was calculated by assuming 1/3 of the maximum energy of the beta particle. Expressed mathematically, the average energy of the beta is given by:

$$\bar{E}_\beta = \frac{E_{\beta\ max}}{3}$$

where,

$E_{\beta max}$ = maximum energy of Beta particle (Ref. 11)

The above energy balance assumes 2/3 of the remaining energy of disintegration is attributable to the neutrino. A detailed description of beta energy absorption is given below, as well as the detailed calculation results for Prefilter 29.

4.3 Energy Absorption from Transuranics

Energy absorption from transuranics is not included in this report at this time due to lack of quantitative laboratory analyses. When these analyses are completed, this calculation will be incorporated in this report by revision.

The equation to be used is essentially that for beta energy absorption, with the exception that $\overline{E}_\beta$ is replaced by E_T, where E_T is the energy of the emitted particle. In the case of beta emitting transuranics, the same equation is used, again assuming 1/3 of E_{max} as the energy of the particle.

GAMMA ENERGY DEPOSITION RATE:

ENERGY FLUX IS GIVEN BY: ϕE_γ

ENERGY DEPOSITION RATE IS GIVEN BY:

$$\left[\frac{\mu_e}{\rho}\right] \phi E_\gamma$$

Where: μ_e/ρ Energy Absorption coefficient (cm^2/gm)

ϕ Photon Flux ($\gamma/cm^2 \cdot sec$)

E_γ Energy of Photon (MeV)

ABSORBED DOSE RATE IS GIVEN BY:

$$RAD/sec_\gamma = 1.602 \times 10^{-8} \frac{\mu_e}{\rho} \phi E_\gamma$$

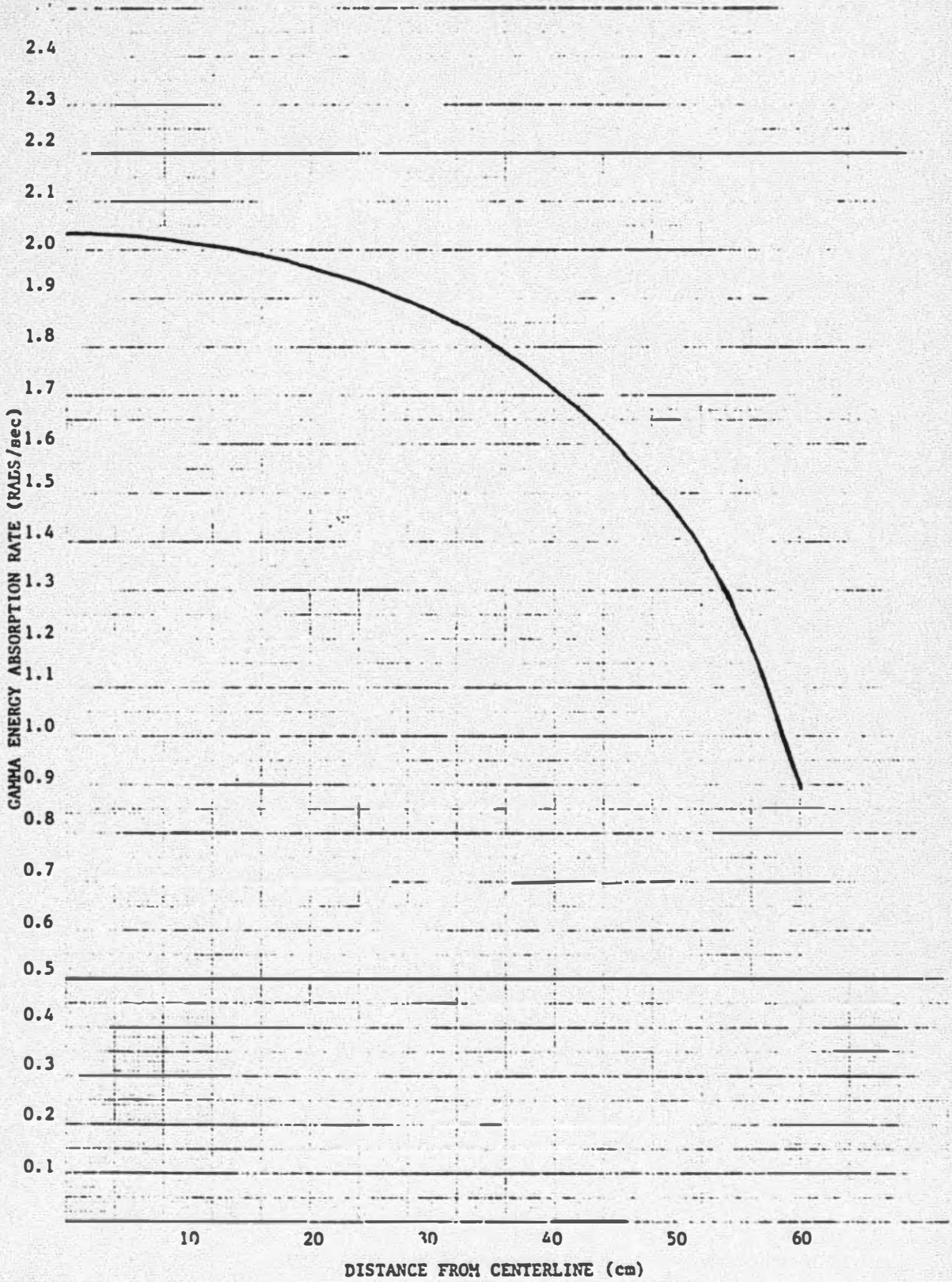
UNIT BALANCE: $\frac{cm^2}{gm} \frac{\gamma}{cm^2 \cdot sec} MeV \frac{RAD \cdot gm}{100\ erg} 1.602 \times 10^{-6} \frac{erg}{MeV}$

Note that μ_e is $(\mu_t - \Sigma_s)$

i.e., scattering is not included

$$RAD/sec_\gamma = 1.602 \times 10^{-8} \left[\frac{\mu_e}{\rho}\right] \phi E_\gamma$$

GAMMA ENERGY ABSORPTION RATE vs RADIUS



BETA ENERGY DEPOSITION RATE:

ENERGY FLUX IS GIVEN BY:

$$3.7 \times 10^{10} \frac{S_v \bar{E}_\beta}{\mu}$$

ENERGY DEPOSITION RATE IS GIVEN BY:

$$k \left[\frac{\mu_e}{\rho}\right] 3.7 \times 10^{10} \frac{S_v \bar{E}_\beta}{\mu}$$

Where:

k	unit balanace constant determined below
μ_e	energy absorption coefficient (assumed = 1) $(1/cm)$
ρ	density of media (g/cm^3)
S_v	volumetric source term (ci/cm^3)
$\bar{E}_\beta$	average energy of beta praticle MeV

Unit Balance: $\left[\frac{RAD\ gm}{100\ erg}\right] 1.602 \times 10^{-6} \left[\frac{erg}{MeV}\right] \left[\frac{1}{cm}\right] \left[\frac{cm^3}{.705\ gm}\right] \left[\frac{3.7 \times 10^{10}}{sec \cdot Ci}\right] \left[\frac{Ci}{cm^3}\right] [cm] [MeV]$

$$RAD/sec_\beta = 8.4 \times 10^2\ S_v\ \bar{E}_\beta$$

and the infinite β dose is given by:

$$\frac{\dot{D}_\beta\ (RAD/sec)}{\lambda_i\ (1/sec)}$$

Where:

λ_i Decay constant of isotope (1/sec)

PREFILTER 29

BETA DOSE RATE (CATION BED)

Isotope	λ(1/sec)	S_v(Ci/cc)	$\bar{E}_\beta$(MeV)	$\dot{D}$ (RADS/sec)	D_∞ (RADS)
Co-58	1.12×10^{-7}	1.36×10^{-9}	.43	4.9×10^{-7}	4.39
Co-60	4.18×10^{-9}	4.80×10^{-9}	.497	2.0×10^{-6}	4.79×10^{2}
Sr-89	1.55×10^{-7}	3.44×10^{-5}	.496	1.43×10^{-2}	9.25×10^{4}
Sr-90	7.83×10^{-10}	1.13×10^{-4}	.182	1.72×10^{-2}	2.20×10^{7}
Y-90	3.01×10^{-6}	1.13×10^{-4}	.763	7.24×10^{-2}	2.41×10^{4}
Zr-95	1.26×10^{-7}	7.26×10^{-8}	.373	2.28×10^{-5}	1.81×10^{2}
Nb-95	2.29×10^{-7}	1.46×10^{-7}	.308	3.78×10^{-5}	1.65×10^{2}
Ru-103	2.02×10^{-7}	8.73×10^{-10}	.242	1.77×10^{-7}	8.77×10^{-1}
Ru-106	2.20×10^{-8}	1.20×10^{-8}	.013	1.32×10^{-7}	6.00
Rh-106	2.31×10^{-3}	1.20×10^{-8}	1.18	1.19×10^{-5}	4.42×10^{2}
Ag-110m	3.17×10^{-8}	1.22×10^{-11}	.5	5.12×10^{-9}	1.61×10^{-1}
Sb-125	8.13×10^{-9}	1.36×10^{-7}	.208	2.37×10^{-5}	2.92×10^{3}
Cs-134	1.07×10^{-8}	4.41×10^{-4}	.485	1.79×10^{-1}	1.68×10^{7}
Cs-137	8.25×10^{-10}	2.00×10^{-3}	.392	6.58×10^{-1}	7.98×10^{8}
Ce-144	2.77×10^{-8}	1.08×10^{-7}	.109	9.89×10^{-4}	3.57×10^{4}
Ce-141	2.51×10^{-7}	3.04×10^{-10}	.194	4.95×10^{-8}	1.97×10^{-1}
Pr-144	6.60×10^{-4}	1.08×10^{-7}	.999	9.06×10^{-5}	3.27×10^{3}
			TOTALS	9.42×10^{-1}	8.36×10^{8}

5.0 Summary of Calculations

5.1 Total Dose Rate Comparison

The total rate is determined by summing the individual dose rates at each interval, i.e. Figure 4.1. Thus, at the centerline, the dose rate is:

$$\dot{D}_{Total} = \dot{D}_{\gamma} + \dot{D}_{\beta} + \dot{D}_{TRU} = (2.03 + 0.94 + 0)\ \text{RAD/sec}$$
$$= 2.97\ \text{RAD/sec}$$

which is approximately 1100 Rads/hr. The beta dose rate is comparable to that calculated by Georgia Tech, i.e., .59 vs .94 for this calculation, which is attributable to source term difference. However, the gamma dose rates differ considerably, 0.3 for Georgia Tech vs. 2.03 for this calculation. Aside from the difference in the source terms, this calculation took into account liner geometry in the cell, density of media, as well as scattering into lower energy groups which increases the probability for energy absorption. As can be seen by Figure 4.1, the maximum gamma dose rate occurs at the centerline, and decreases as a function of distance away from the centerline, as one would expect.

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8. Letter to R.J. McGoey, dated April 2, 1980 from J.A. Carter, RE: ERD-79-058, Task 7, RCBT B Sample

9. ANISNBW, A One Dimensional Discrete Ordinates Transport Code, NPGD-TM-491, Rev. 1, May, 1979.

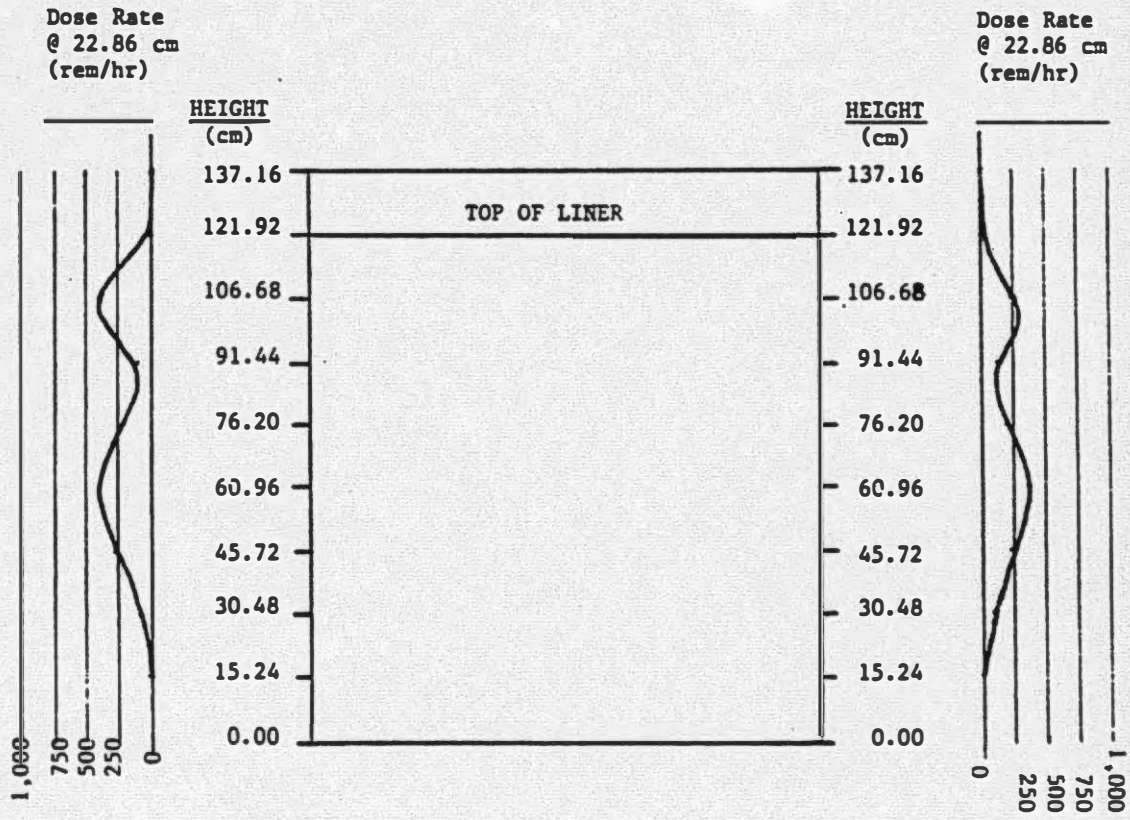
10. Gangwer, T.E., Goldstein, M. and Pillay, K.K.S., Radiation Effects on Ion Exchange Materials, BNL 50781, April, 1978.

11. Table of Isotopes, 7th Edition, Lederer, Shriley, et.al, Wiley and Sons, Inc., 1978.

ATTACHMENT I

Prefilter- 28

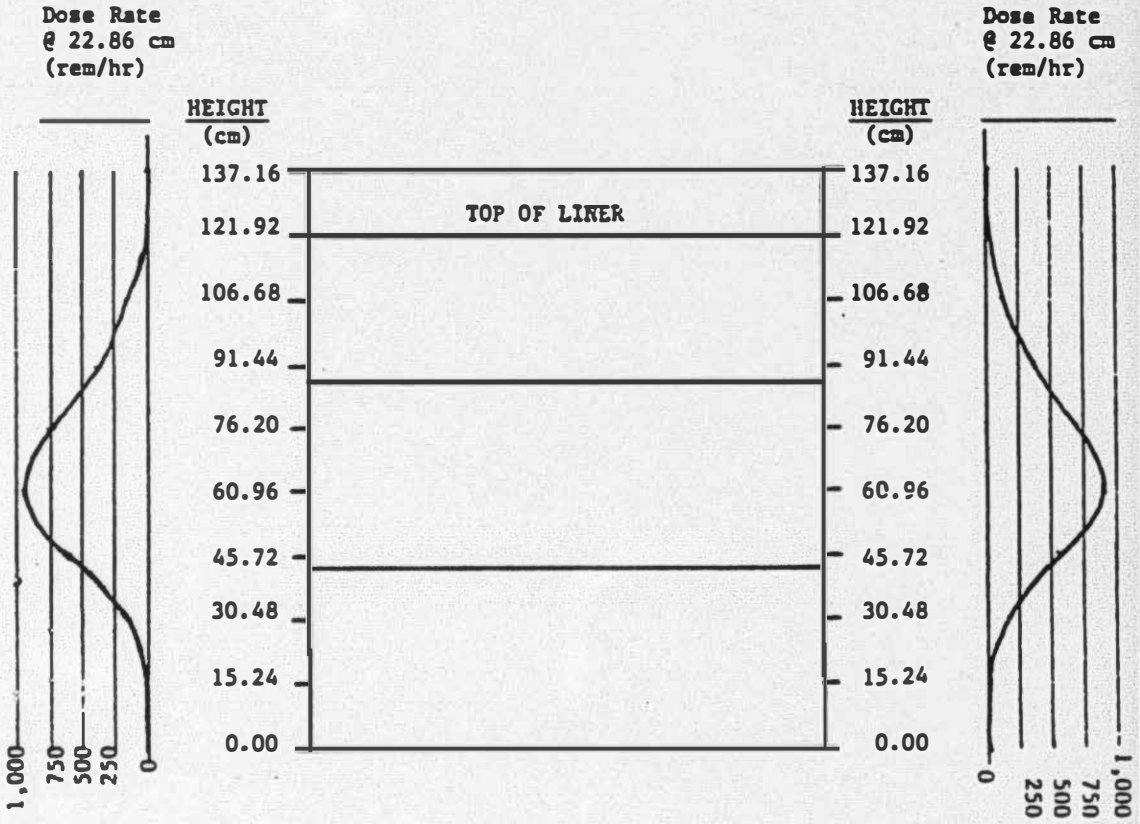
(Survey Date-5/2/80)



Four-Foot (121.92 cm) diameter liner drawn to scale

Prefilter-29

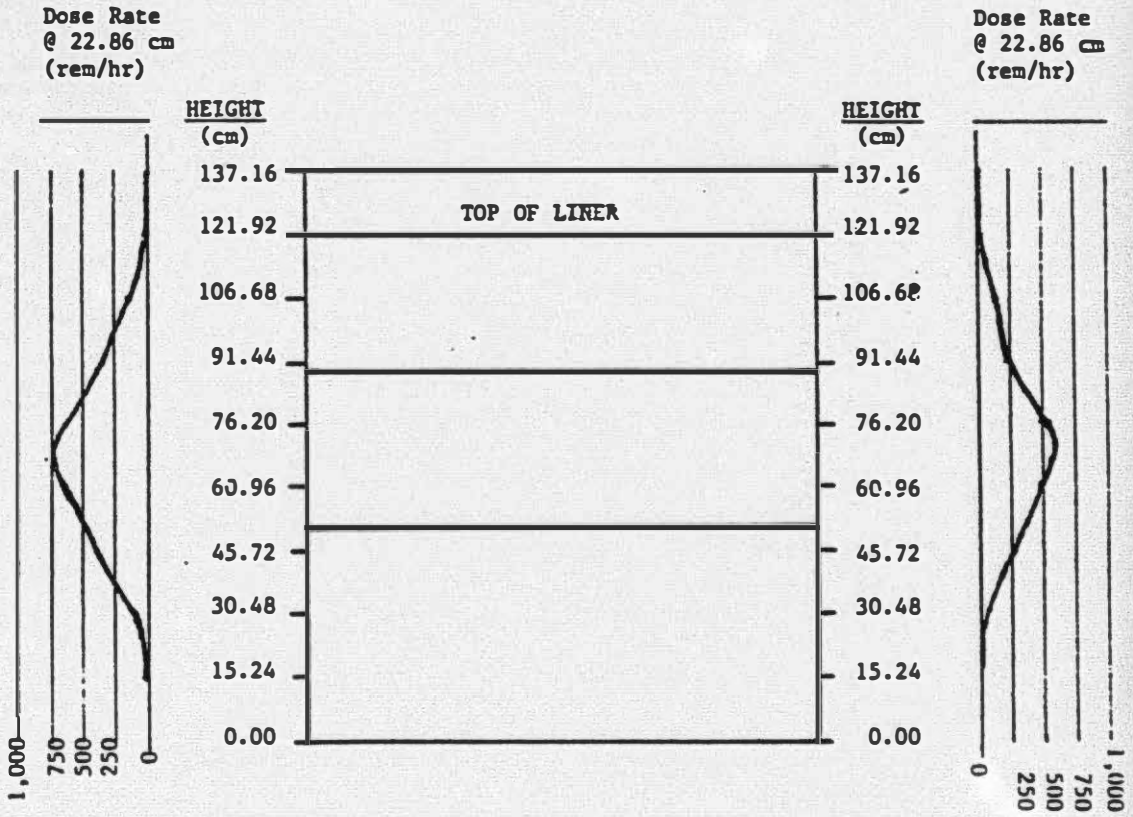
(Survey Date-5/9/80)



Four-Foot (121.92 cm) diameter liner drawn to scale

Prefilter-30

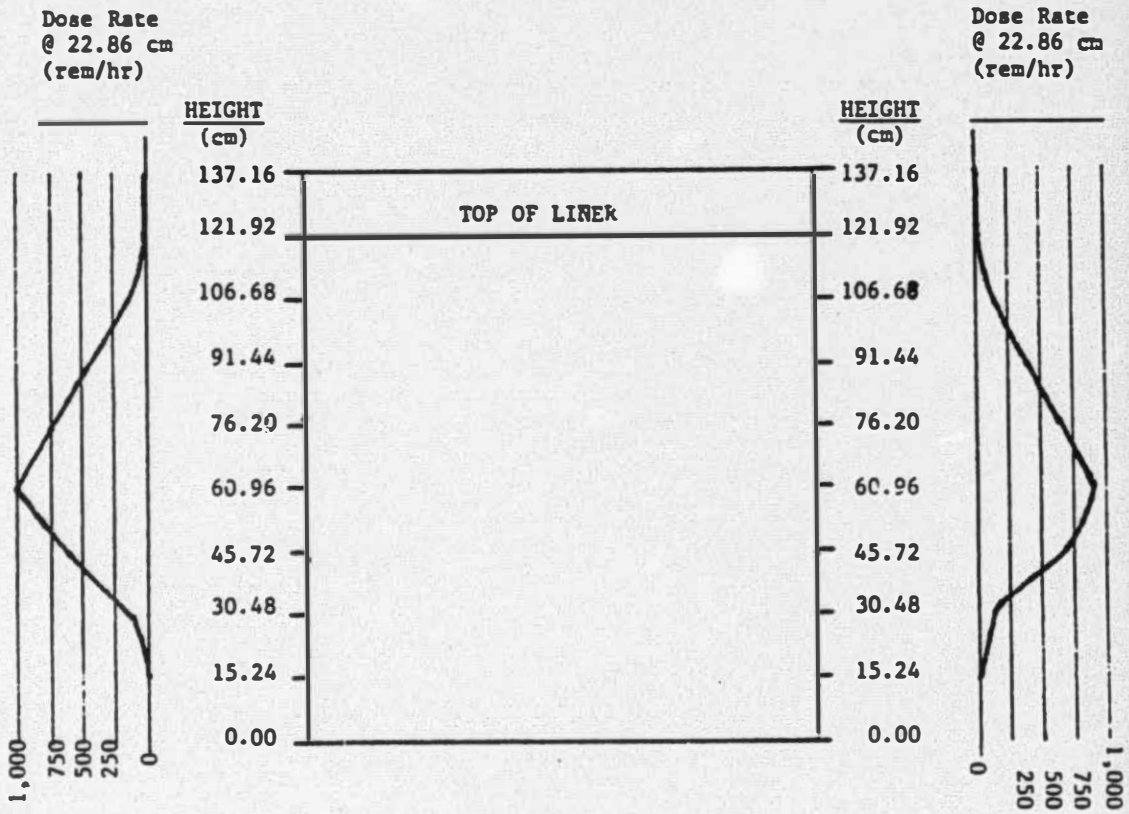
(Survey Date-5/13/80)



Four-Foot (121.92 cm) diameter liner drawn to scale

Prefilter-31

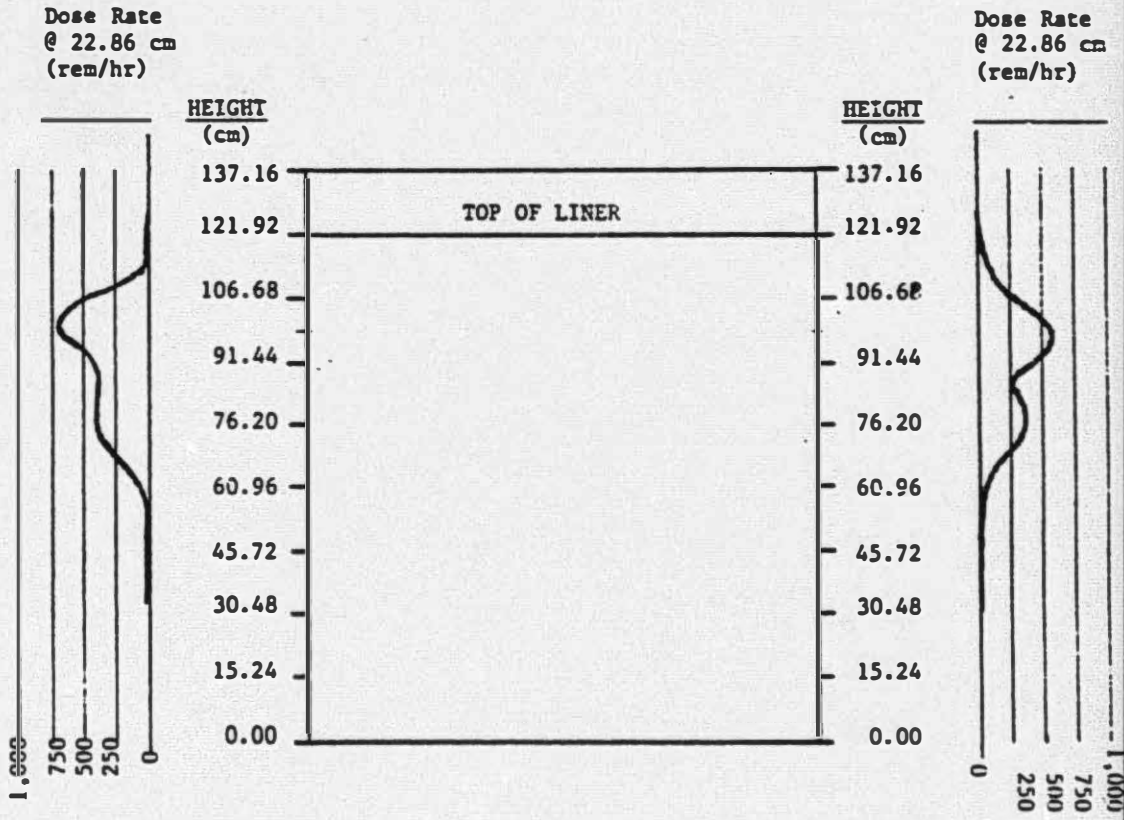
(Survey Date-5/14/80)



Four-Foot (121.92 cm) diameter liner drawn to scale

Prefilter-38

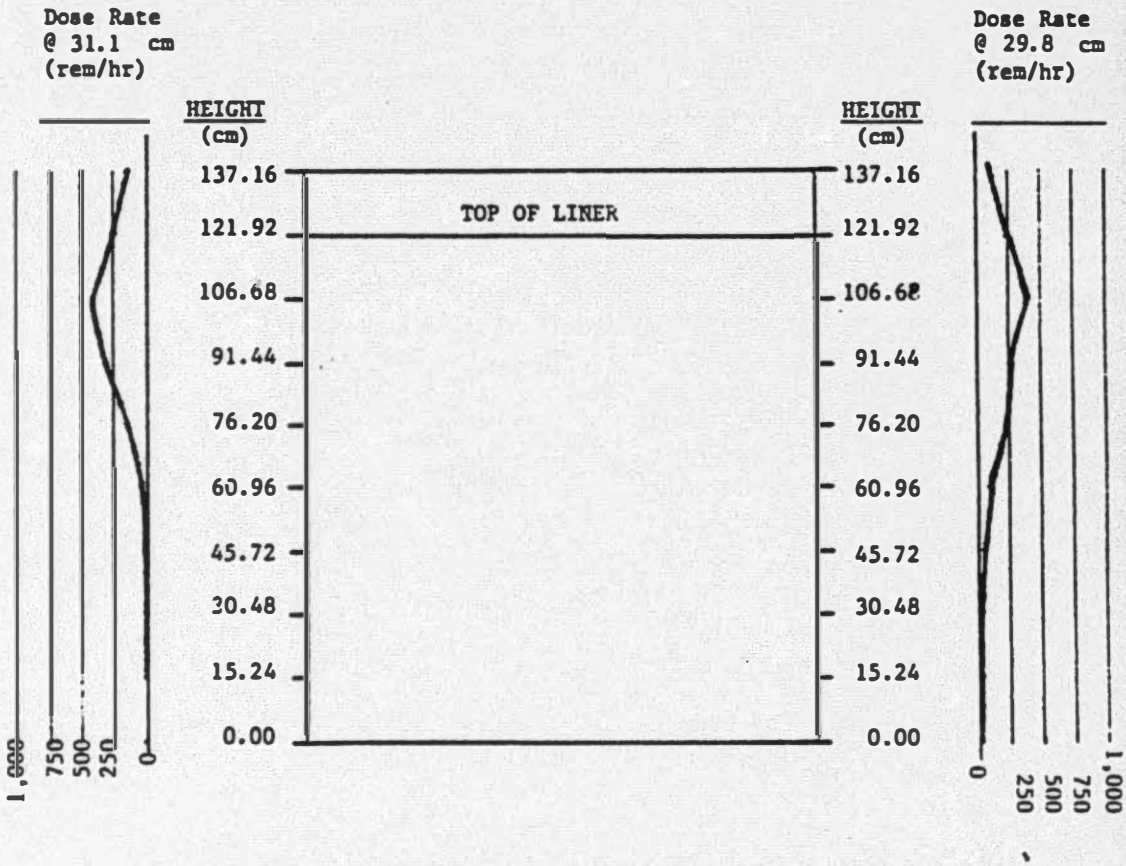
(Survey Date-5/27/89)



Four-Foot (121.92 cm) diameter liner drawn to scale

Prefilter-40

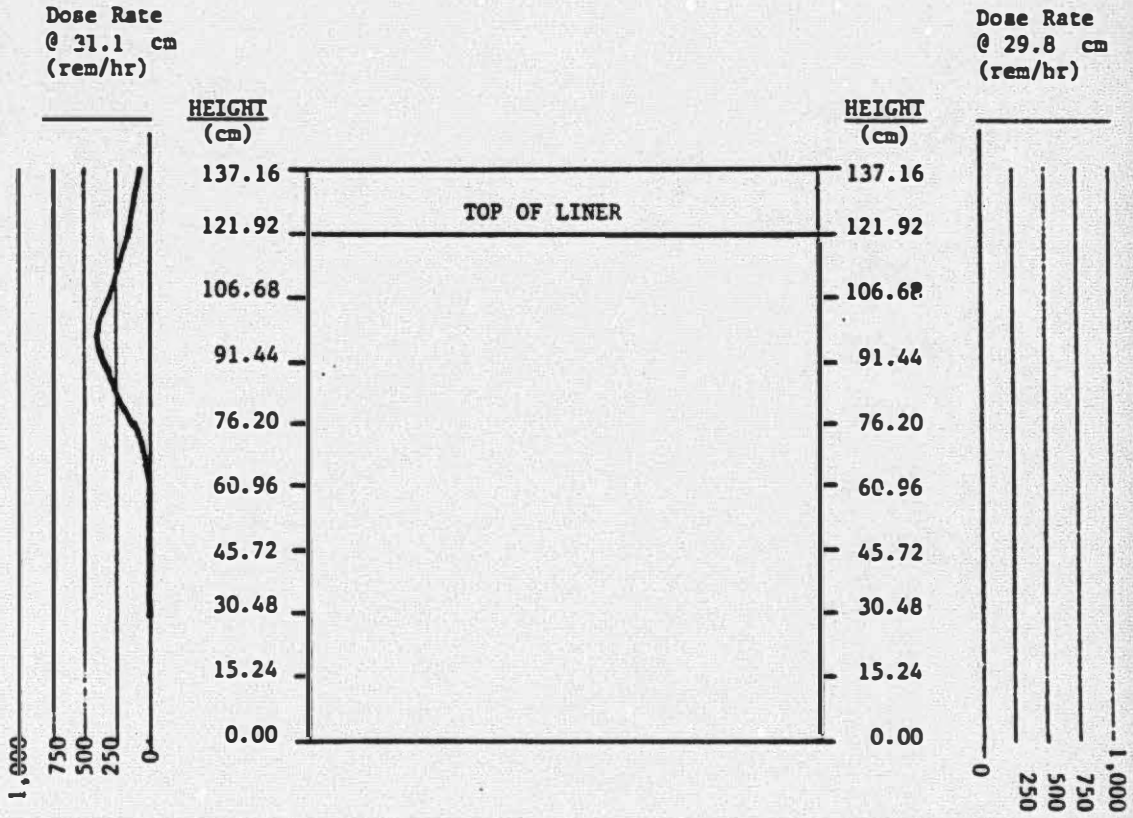
(Survey Date-5/31/80)



Four-Foot (121.92 cm) diameter liner drawn to scale

Prefilter-41

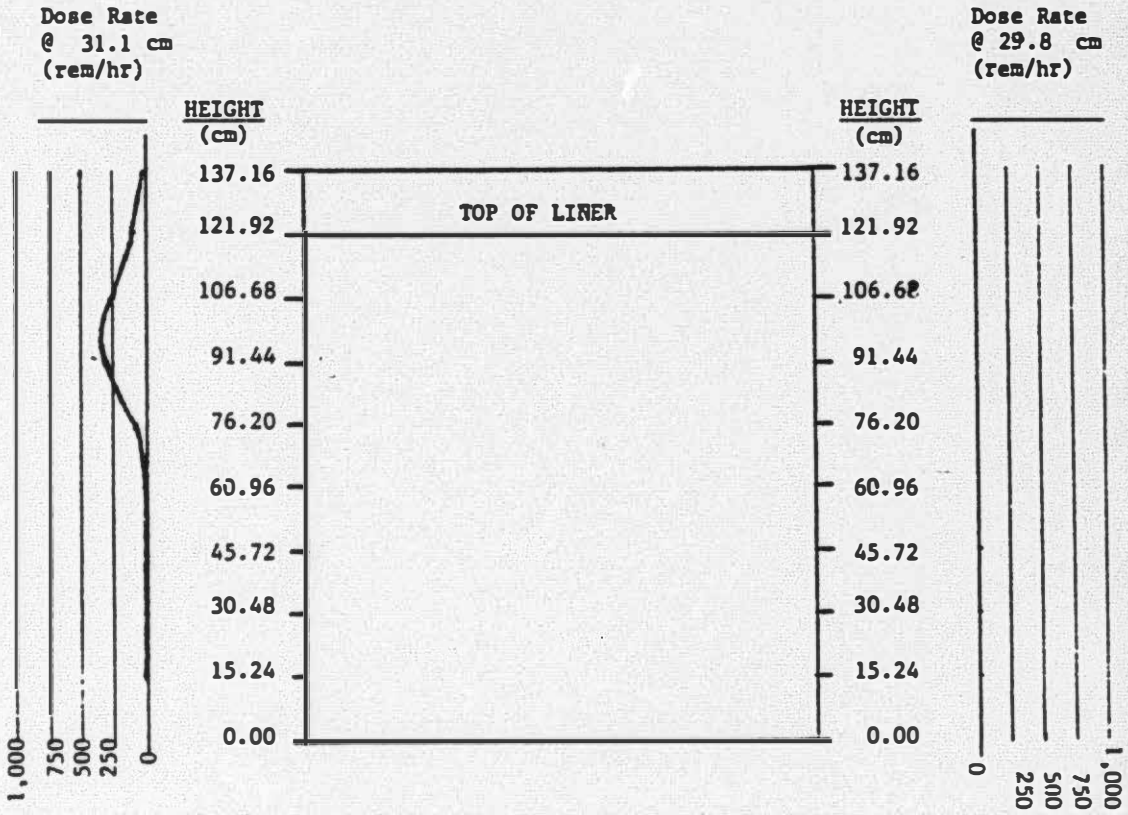
(Survey Date-6/2/80)



Four-Foot (121.92 cm) diameter liner drawn to scale

Prefilter-42

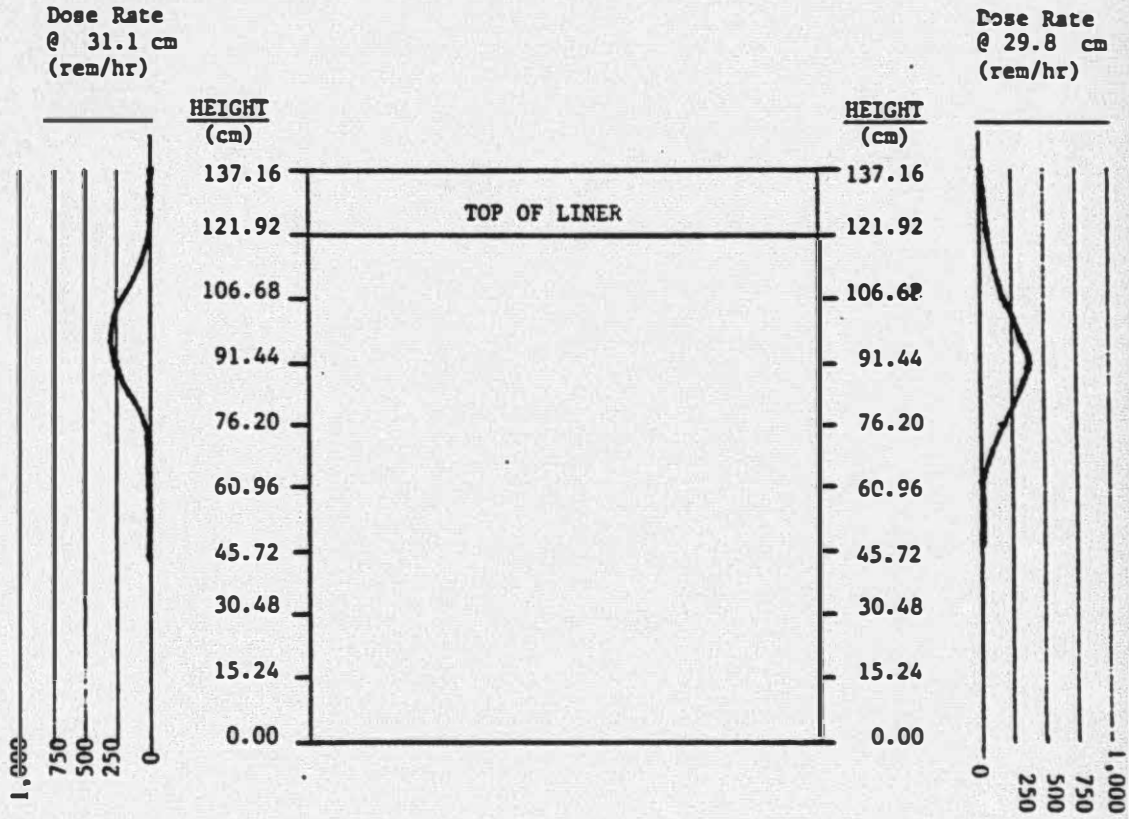
(Survey Date-6/4/80)



Four-Foot (121.92 cm) diameter liner drawn to scale

Prefilter-43

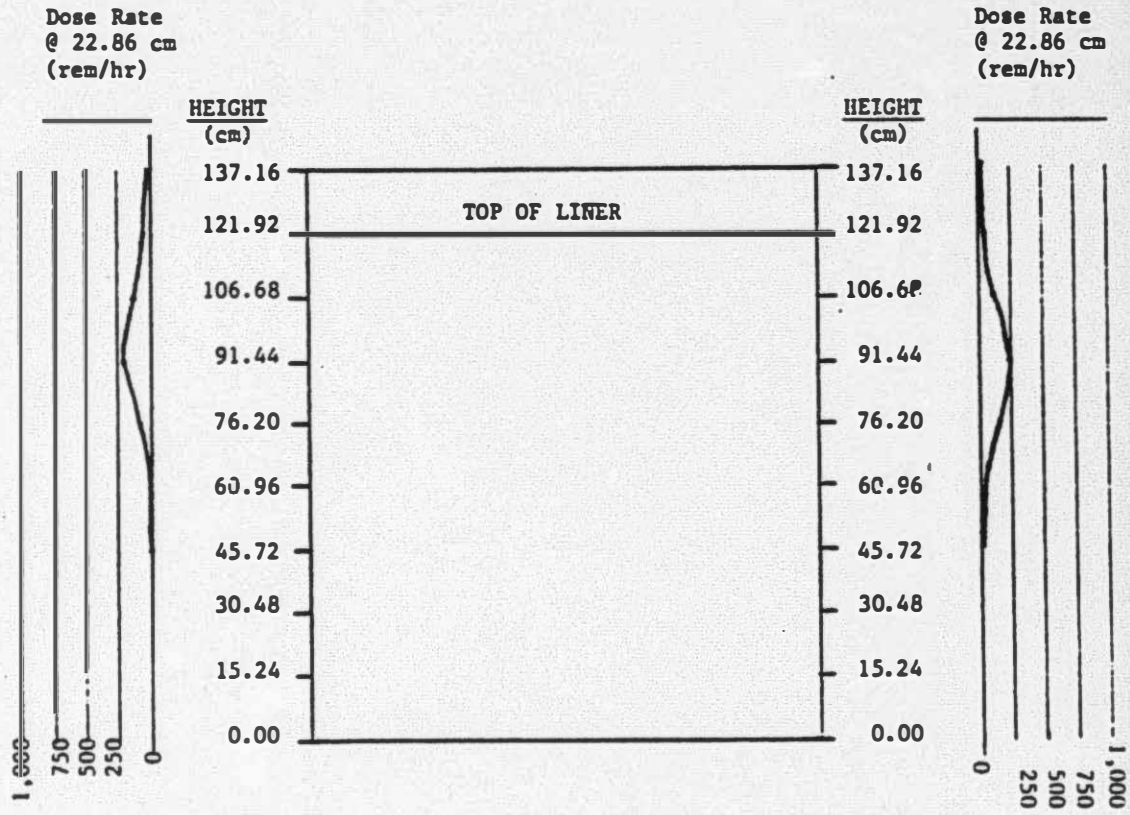
(Survey Date-6/6/80)



Four-Foot (121.92 cm) diameter liner drawn to scale

Prefilter- 44

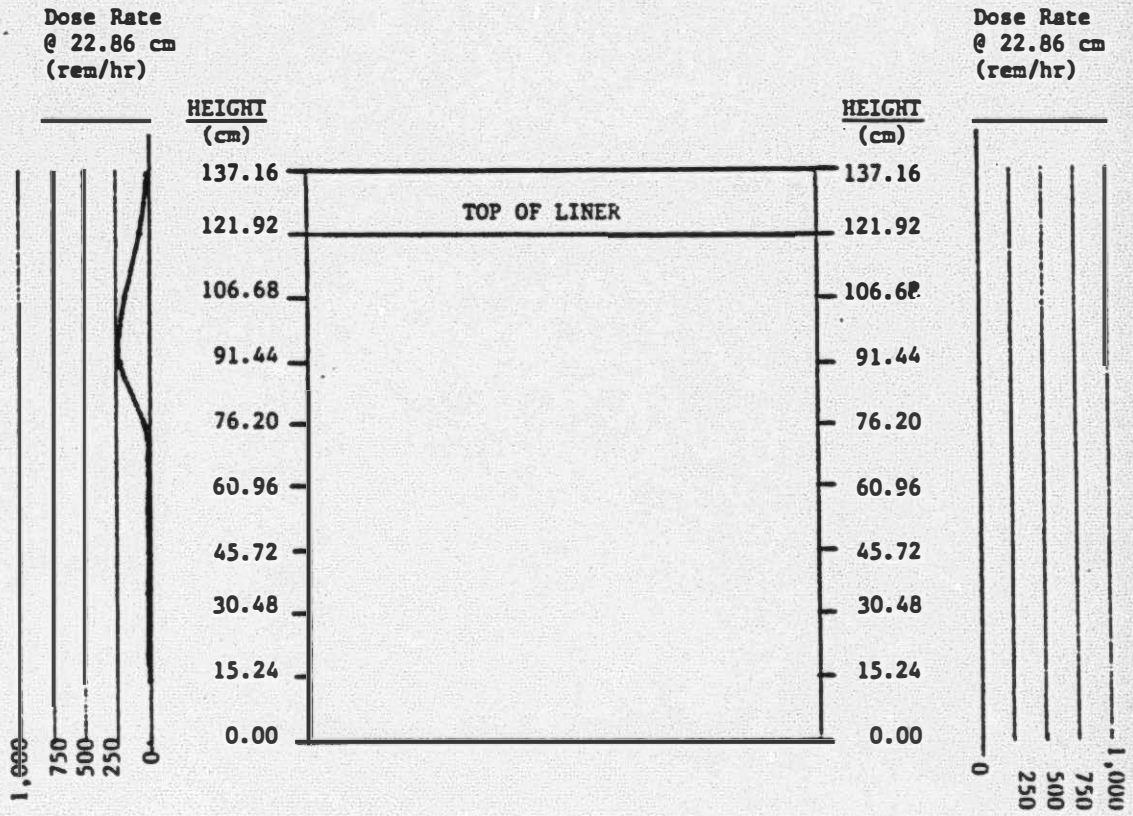
(Survey Date- 6/27/80)



Four-Foot (121.92 cm) diameter liner drawn to scale

Prefilter- 45

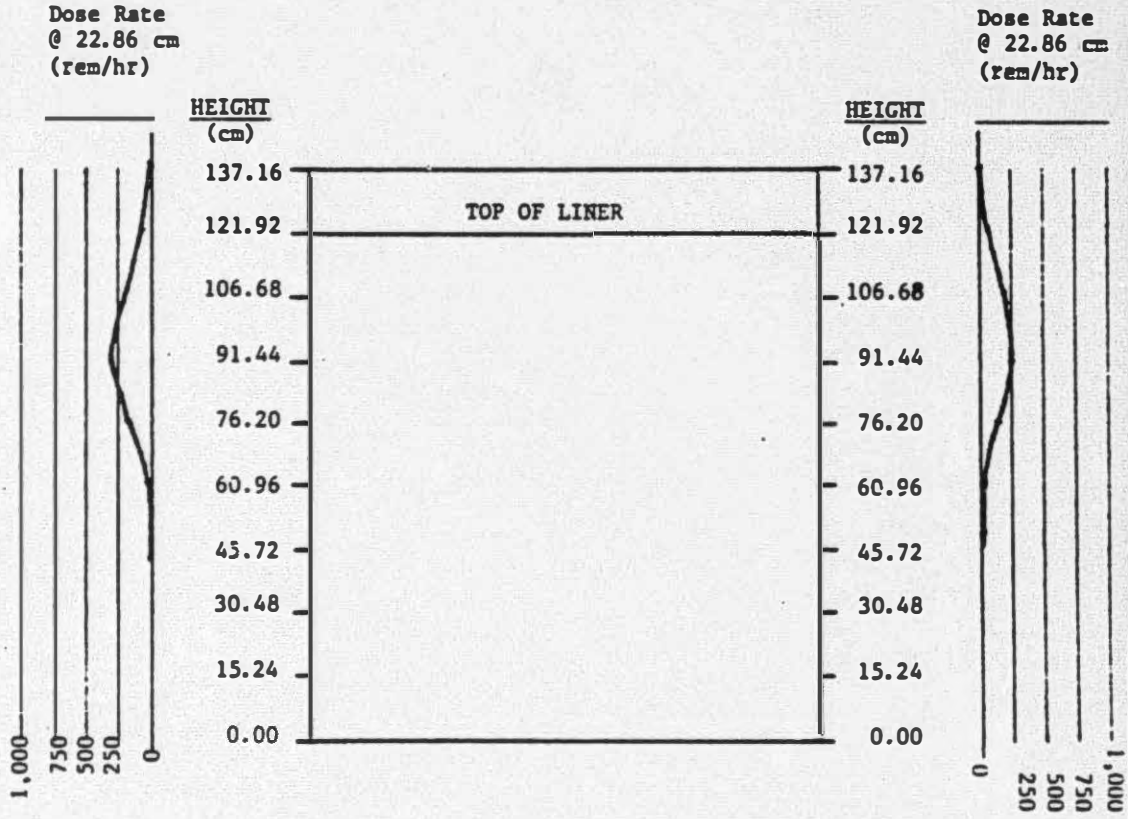
(Survey Date-6/30/80)



Four-Foot (121.92 cm) diameter liner drawn to scale

Prefilter-46

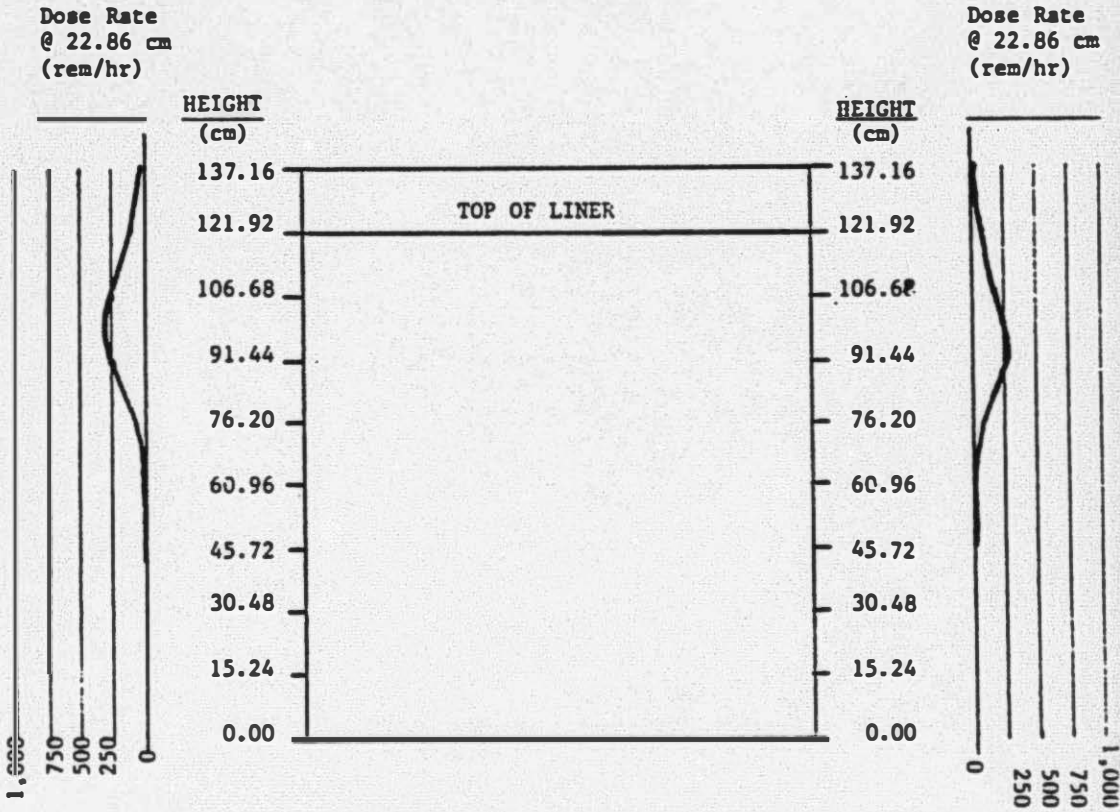
(Survey Date-7/2/80)



Four-Foot (121.92 cm) diameter liner drawn to scale

Prefilter-47

(Survey Date-7/14/80)



Four-Foot (121.92 cm) diameter liner drawn to scale