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HEALTH AND SAFETY BRANCH
PROGRESS REPORT
1968

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NATIONAL REACTOR TESTING STATION
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HEALTH AND SAFETY BRANCH PROGRESS REPORT 1968

J. W. McCaslin

EDITOR

IDAHO NUCLEAR CORPORATION
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PHILLIPS PETROLEUM COMPANY

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ABSTRACT

The Idaho Nuclear Corporation Health and Safety Annual Progress Report for 1968 is comprised of the major accomplishments of the various sections of the Health and Safety Division. It includes significant studies, policies, experiments, projects, pamphlets, and reports produced during the year.

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HEALTH AND SAFETY BRANCH
PROGRESS REPORT

I. REACTOR HEALTH PHYSICS

1. DIRECTOR OF REACTOR HEALTH PHYSICS

1.1 Gamma Radiation Damage of Structural Concrete Immersed in Water (J. F. Sommers)

1.11 Introduction. In late 1955, the question arose of possible damage to the walls of the Materials Testing Reactor (MTR) canal due to gamma irradiation from used fuel stored in the canal. Therefore, the Reactor Health Physics Section obtained permission to start a gamma irradiation program on structural concrete samples to test them for loss of strength due to intense and prolonged gamma irradiation. A batch of 3/4-inch aggregate concrete specimens were prepared from a single "pour". In early 1958, 24 of the specimens were placed into the gamma irradiation facility as test samples and 24 control samples were suspended in the gamma facility canal, but out of the irradiation zone. An eight-year testing program was planned whereby eight control samples would be removed from the canal, inspected, and tested by compression to destruction at specific gamma irradiation intervals of 10^{10}, 10^{11}, and 2×10^{11}R.

In September of 1958, four test samples with 10^{10}R exposures and four control samples were compression tested to destruction with no significant difference in strength being evident between the control and test specimens. In late 1958 and early 1959, the leak rate in the MTR canal increased so drastically that concern over the integrity of the canal concrete became a major worry. Therefore, four more test samples with 10^{10} and 1.2×10^{10}R exposures and four control samples of concrete were compression tested to destruction, again, with no evidence of loss of strength due to gamma irradiation. No significant deterioration of the samples was evident at this time. The MTR canal was lined with stainless steel to stop the leaks, and concern over the structural integrity of the concrete subsided.

In April of 1967, eight more test samples and eight control samples were removed from the gamma irradiation facility and visually inspected. Four of the test specimens had been irradiated to 10^{11}R and four to 2×10^{11}R. Extensive surface deterioration was evident on some of the test samples; the deterioration was obviously progressive with increased gamma exposure. There were some anomalies in the extent and severity of the surface deterioration that appeared to be a function of position in the sample holder cans. Other anomalies were evident, such as foreign deposits on some surfaces of the test samples and a lack of deterioration at one spot on one sample which otherwise exhibited severe surface deterioration. Small samples were obtained of the control and test specimens and of the debris from the deteriorated surfaces for emission spectrographic analysis. All of the samples were then tested to destruction for compressive strength.

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The remaining eight test samples are still being irradiated, and plans at this time are to take four of them to total exposures of 3×10^{11}R and four to 4×10^{11}R. This will take until about 1975 to complete. However, the apparent loss of strength with gamma radiation exposures in the range of 10^{11}R makes issuing of this interim report advisable.

1.12 Methods. The molded three-inch diameter by four-inch long (nominal concrete samples were cast on January 23, 1956, using 3/4-inch maximum diameter aggregate and No. 28 mix having the following quantities per yard of concrete: (a) 517 pounds of cement, (b) 2,040 pounds of 3/4-inch aggregate, (c) 1,073 pounds of sand, (d) 240 pounds of water, and (e) 120 cc of "Protex". All cylinders were stored in the curing room of the AEC-IDO Concrete Testing Laboratory for 14 days. At the end of seven days, two cylinders were broken in compression at 2,350 and 2,940 psi (uncorrected from nominal size of samples).

Requests for compression tests of the irradiated samples and controls called for use of the ASTM standard method of test for "Compressive Strength of Molded Concrete Cylinders", ASTM designation C39-49. The cylinders were broken by different laboratories. Variations from stated conditions in Test C39-49 are evident in a comparison of the absolute test results. However, all test sample data in Table I are compared to control specimen data obtained at the same place and time. Thus, the percent loss of strength of the test specimens is probably valid.

Qualitative emission spectrographic analyses were performed on selected samples of test and control specimens by the Analytical Chemistry Branch of Idaho Nuclear Corporation at the Idaho Chemical Processing Plant (ICPP).

The concrete cylinders were enclosed in open top, 24-SO, 20-gauge sheet aluminum cans, 11 inches high and 3-1/2 inches in diameter, two samples to a can. A 3/16-inch diameter aluminum rod was welded to the top of the cans to allow suspension of the samples in the irradiation facility. The top samples in the cans were designated as "A" samples; the bottom specimens were designated as "B" samples. Each test can was lettered serially to keep identity on all test samples. A matching set of control specimens also was suspended in the gamma facility canal, but out of the gamma irradiation zone. Test samples are identified as T-1A and T-1B through T-12A and T12B, while control specimens are identified as C-1A and C-1B through C-12A and C-12B.

The gamma facility canal water in which the specimens were submerged is maintained at 60 to 70°F, is demineralized to a water purity of 1 ppm total solids, and has a pH of 6 to 7 ± 0.5. Internal temperatures were calculated to have never exceeded 75°F. The average gamma source intensity varied from 2.4×10^6 to 3.9×10^6 R/hr.

1.13 Test Results. The irradiated specimens removed from the gamma facility in March of 1967 exhibited extensive surface deterioration which appeared to be progressive with the quantity of incident radiation. The deterioration also was a function of the position of the specimens in the irradiation holder cans (see Figures 1 through 8). Specimens removed previously did not show any surface deterioration. Control samples removed from the facility at the same time (March, 1967) showed no evidence of surface deterioration except on the tops of the "A" specimens (compare Figures 9 and 10).

TABLE I

DESTRUCTIVE COMPRESSION TEST ON STRUCTURAL CONCRETE SAMPLES SUBJECTED TO GAMMA IRRADIATION

Date Tested	Sample Number[a] and Position[b]	Incident γ Radiation Exposure (R)	Compression Strength, - Breaking (psi)	Max [c]-Min and Average Loss of Strength (psi)	% Loss of Compressive Strength
9-23-58	C-2A	0.0	4,880	-	-
9-23-58	C-1A	0.0	5,520	-	-
9-23-58	T-2A	1.05×10^{10}	4,960	+ 560	+ 10.1
9-23-58	T-1A	1.2×10^{10}	5,920	- 1,040	- 21.3
9-23-58	Average-T(1&2)A	1.125×10^{10}	5,440	- 240	- 4.61
9-23-58	C-1B	0.0	5,050	-	-
9-23-58	C-2B	0.0	6,480	-	-
9-23-58	T-1B	1.2×10^{10}	5,480	+ 1,000	+ 15.4
9-23-58	T-2B	1.05×10^{10}	6,640	- 1,590	- 31.5
9-23-58	Average-T(1&2)B	1.125×10^{10}	6,060	- 295	- 5.11
4-22-59	C-4A	0.0	1,630	-	-
4-22-59	C-3A	0.0	1,910	-	-
4-22-59	T-4A	1.0×10^{10}	1,855	+ 55	+ 2.88
4-22-59	T-3A	1.0×10^{10}	1,930	- 300	- 18.4
4-22-59	Average-T(3&4)A	1.0×10^{10}	1,892	- 122	- 6.89
4-22-59	C-4B	0.0	1,440	-	-
4-22-59	C-3B	0.0	1,490	-	-
4-22-59	T-3B	1.0×10^{10}	1,380	+ 110	+ 7.38
4-22-59	T-4B	1.0×10^{10}	1,530	- 90	- 6.25
4-22-59	Average-T(3&4)B	1.0×10^{10}	1,455	+ 10	+ 0.68
12-1-67	C-5A	0.0	3,420	-	-
12-1-67	C-6A	0.0	3,870	-	-
12-1-67	T-5A	1.0×10^{11}	1,830	+ 2,040	+ 51.7
12-1-67	T-6A	1.0×10^{11}	2,260	+ 1,160	+ 33.9
12-1-67	Average-T(5&6)A	1.0×10^{11}	2,045	+ 1,600	+ 43.8
12-1-67	C-6B	0.0	3,320	-	-
12-1-67	C-5B	0.0	4,000	-	-
12-1-67	T-6B	1.0×10^{11}	2,410	+ 1,590	+ 39.8
12-1-67	T-5B	1.0×10^{11}	2,520	+ 800	+ 24.1
12-1-67	Average-T(5&6)B	1.0×10^{11}	2,465	+ 1,195	+ 32.6
12-1-67	C-9A	-	3,100	-	-
12-1-67	C-10A	-	4,020	-	-
12-1-67	T-9A	2.0×10^{11}	1,520	+ 2,500	+ 62.1
12-1-67	T-10A	2.0×10^{11}	1,580	+ 1,520	+ 49.1
12-1-67	Average-T(9&10)A	2.0×10^{11}	1,550	+ 2,010	+ 56.5
12-1-67	C-10B	-	3,270	-	-
12-1-67	C-9B	-	3,540	-	-
12-1-67	T-10B	2.0×10^{11}	2,130	+ 1,410	+ 39.8
12-1-67	T-9B	2.0×10^{11}	2,910	+ 360	+ 11.0
12-1-67	Average-T(9&10)B	2.0×10^{11}	2,520	+ 885	+ 26.0

[a] Control samples are indicated by a "C" prefix, test samples by a "T" prefix.

[b] Position of the samples in the top of the irradiation holder cans is indicated by the suffix "A." The bottom position in the cans is indicated by the suffix "B."

[c] Maximum loss of strength is obtained by subtracting minimum strength of a test sample from the maximum strength of a companion control sample. Minimum loss of strength is obtained by the opposite method. Percent loss of strength is obtained by dividing the loss as indicated above by the control sample strength.

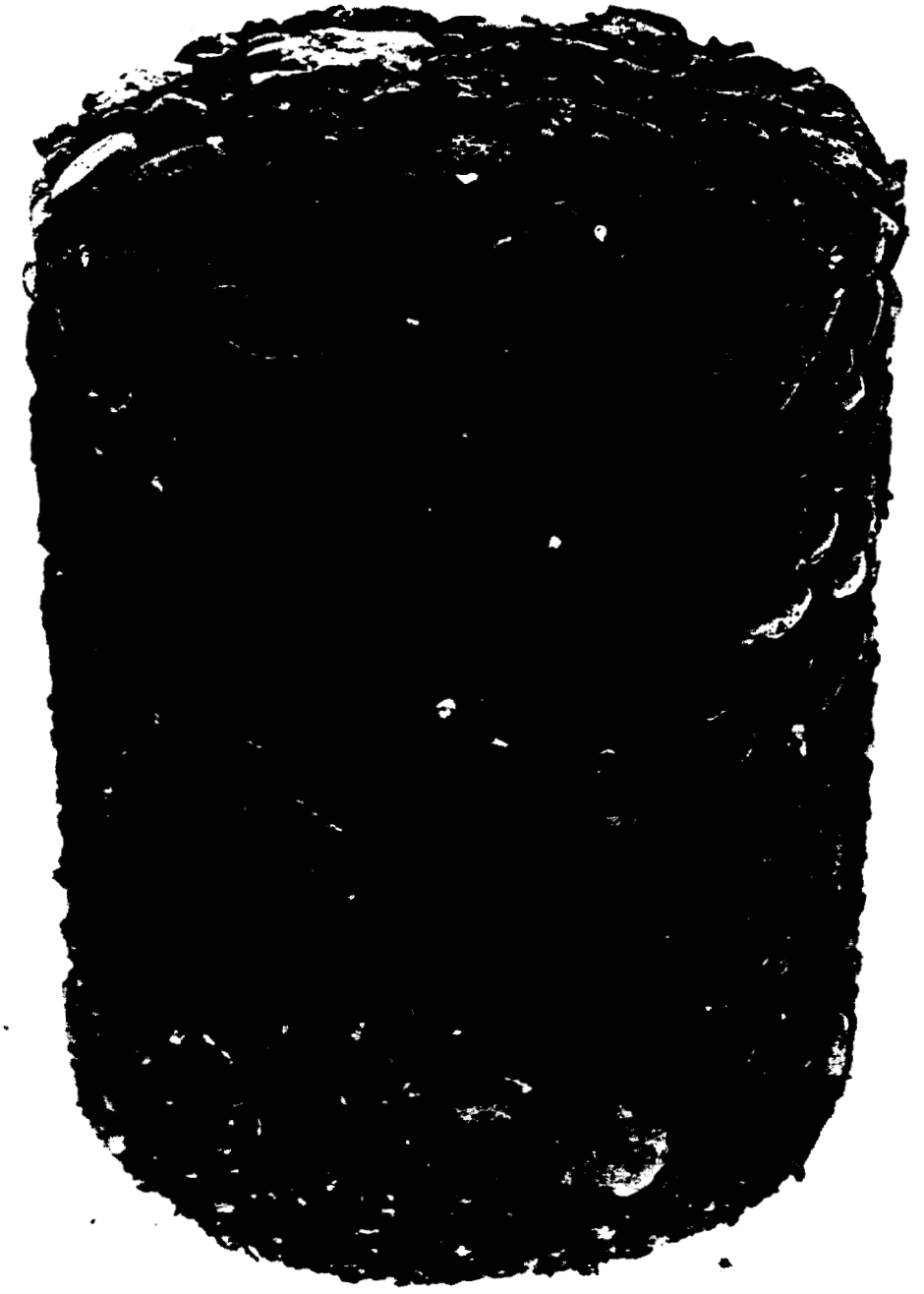

Fig. 1 Concrete cylinder -- Test Sample 5A.

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Fig. 2 Concrete cylinder -- Test Sample 5B.

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Fig. 3 Concrete cylinder -- Test Sample 6A.

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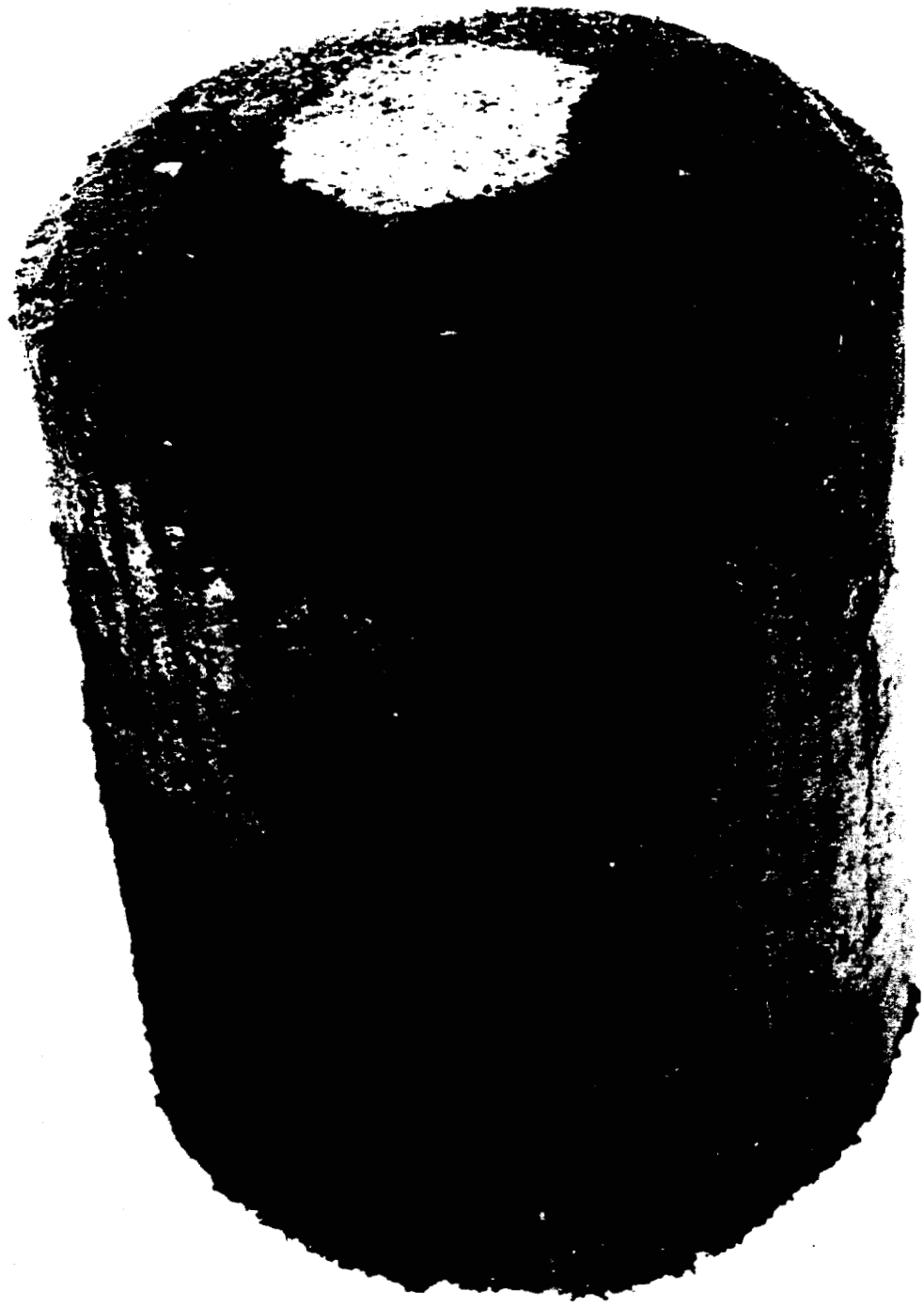

Fig. 4 Concrete cylinder -- Test Sample 6B.

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Fig. 5 Concrete cylinder -- Test Sample 9A.

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Fig. 6 Concrete cylinder -- Test Sample 9B.

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Fig. 7 Concrete cylinder -- Test Sample 10A.

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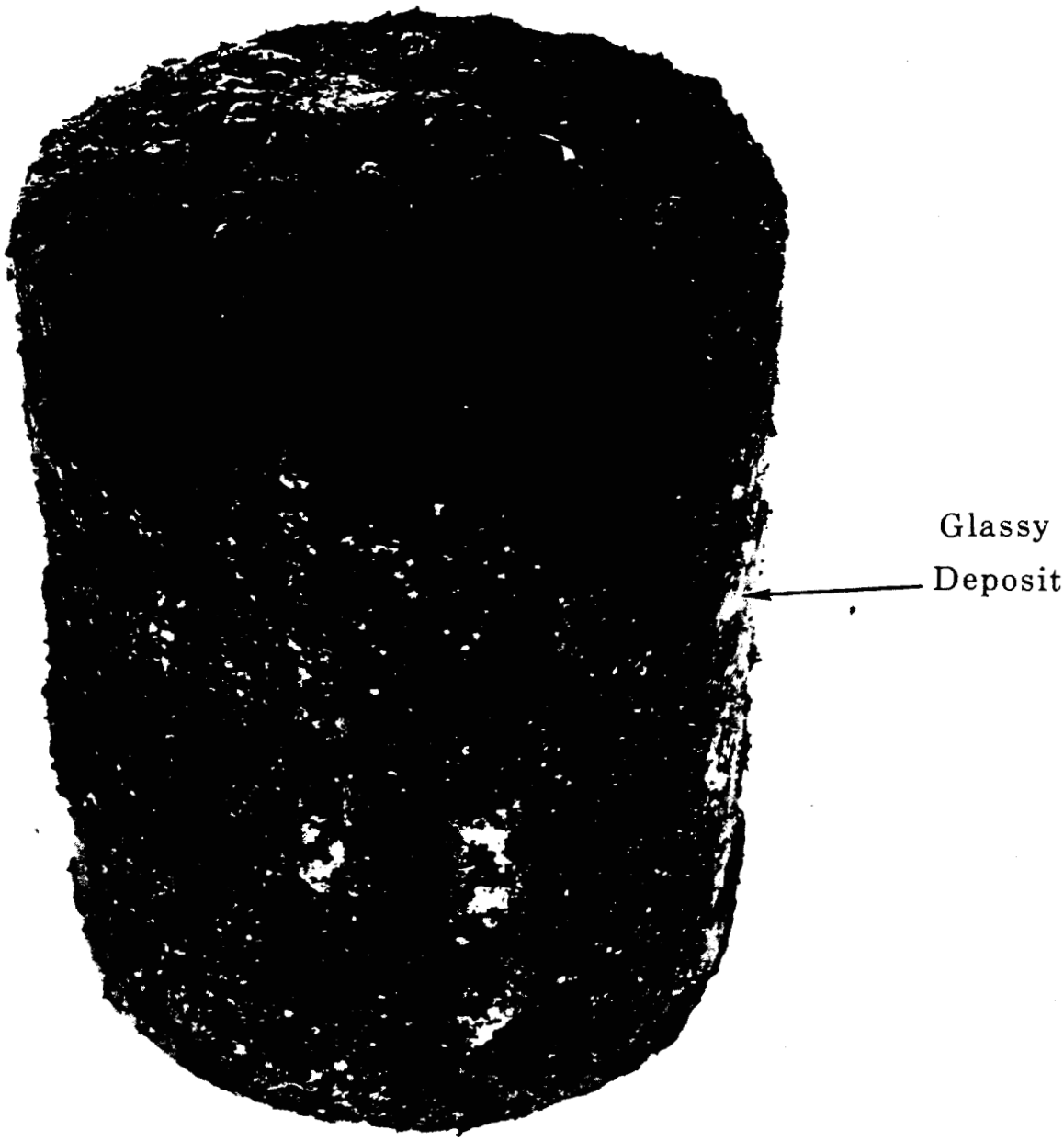

Fig. 8 Concrete cylinder -- Test Sample 10B.

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Fig. 9 Concrete cylinder -- Control Sample 6A.

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Fig. 10 Concrete cylinder -- Control Sample 6B.

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Notice the relative lack of deterioration of the "B" test specimens in relation to the "A" test specimens. Two other anomalies are evident: (a) the 1 x 1-1/2-inch spot on sample T-9A (Figure 5) that shows no visual surface deterioration and (b) the foreign glassy deposits in patches on the surface of test sample T-10B (Figure 8).

Table I indicates the results of the compression tests on concrete specimens in September, 1958, April, 1959, and December, 1967. The average percent losses of strength versus gamma exposures from Table I data are plotted in Figures 11 and 12 for the "A" and "B" test specimens, respectively. The log-linear graphs are assumed functions, but they dramatically illustrate the apparent loss of strength of the concrete samples due to gamma irradiation. The difference in the slope of the graphs illustrates the effect of position of the test samples in the holder cans. The limits of variation shown on the graphs are the maximum and minimum percent loss of strength data indicated in Table I.

The test specimens were reddish-brown to light yellowish-tan in color while the control specimens had a "pasty" grayish coloration. Sand from the deteriorated mortar of the "A" specimens settled out and adherred to the sides of the "B" test specimens. Much of this debris was a dark reddish-brown color.

Spectrographic emission analyses were obtained on mortar samples from control specimens, test specimen T-9A, on the loose decomposition debris from T-9B, on the glassy deposits from sample T-10B, and on a combination of glassy deposit and loose debris from T-10B. The only definite results show that: (a) there was no significant difference in the irradiated and unirradiated mortars, (b) the loose debris from the decomposed mortar was high in iron, silicon, lead, and molybdenum in relation to the nondecomposed mortar, and (c) the glassy deposits on specimen T-10B were high in molybdenum, lead, silicon, tin, zirconium, and nickel. In particular, the nickel content of the glassy deposits was up over a factor of 100 in relation to the mortar samples.

On June 28, 1956, some measurements were taken with an ion chamber at the gamma facility canal to estimate the exposure rate to the gamma facility and MTR canal walls. The rate at the canal wall as measured by the ion chamber was 1.43×10^6 R/hr, with fuel elements aligned at 12 inches from the walls. The exposure rate in the concrete walls at various depths of concrete was calculated (shown in Figure 13). Fifteen years at this rate would account for a total exposure of 1.9×10^{11} R on the inside surface of the MTR canal walls in the area of the fuel storage racks.

In 1958, R. G. Clark published a report[1] listing in summary various studies of radiation damage to concrete. The studies were primarily concerned with neutron radiation damage. Gamma radiation exposures were incidental to the neutron exposures and usually were very low or were not measured quantitatively. Clark's conclusions were "Data are available for conditions of flux up to 2×10^{19} nvt (thermal) and temperature to 120°C. Under these conditions, radiation damage to concrete is apparently insignificant. All effects on concrete due to radiation per se were too slight to reliably measure because of the gross effects from the increased temperature during exposure". Clark

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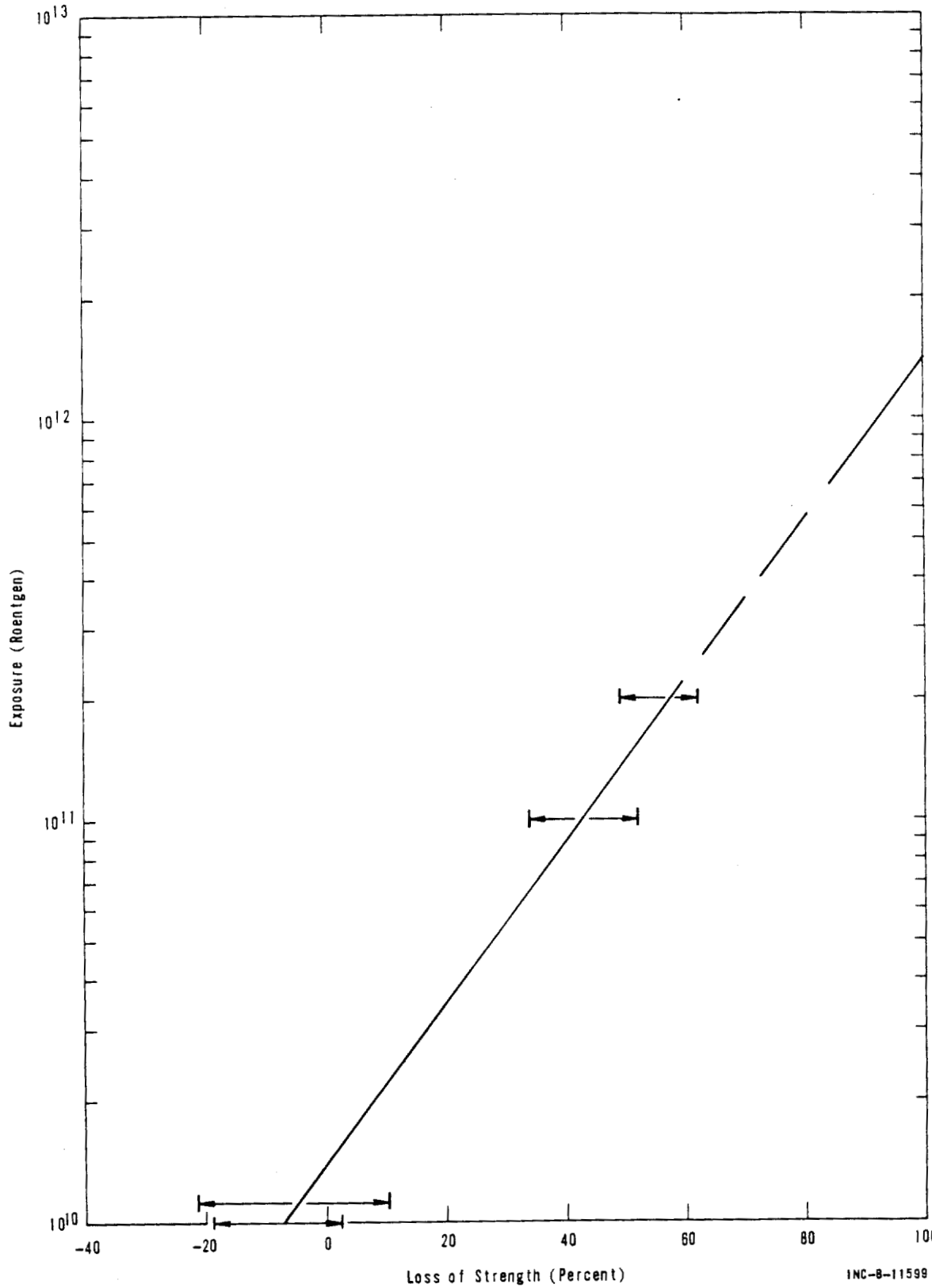

Fig. 11 "A" specimens percent loss of strength versus Roentgen of gamma radiation.

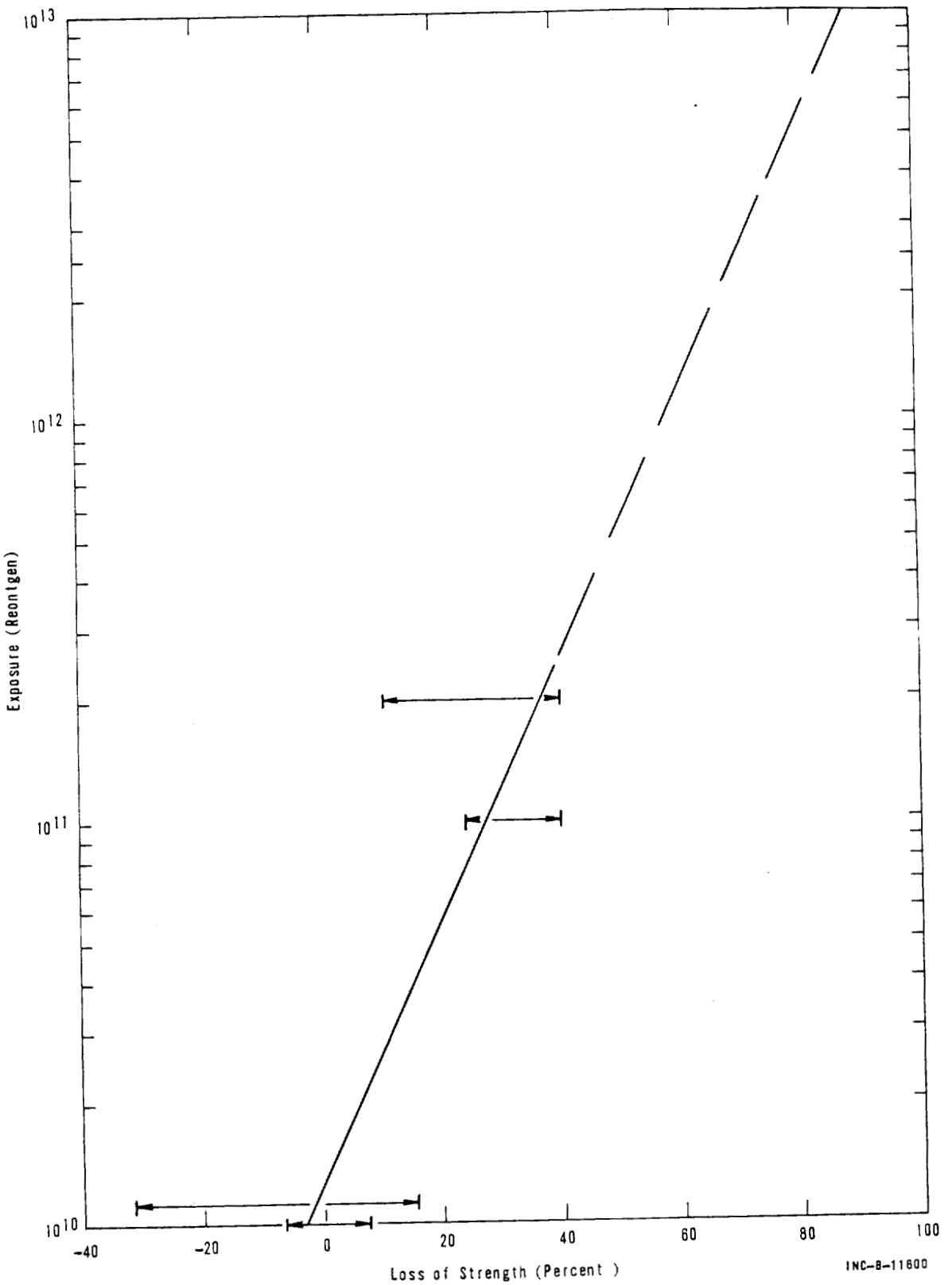

Fig. 12 "B" specimens percent loss of strength versus Roentgen of gamma radiation.

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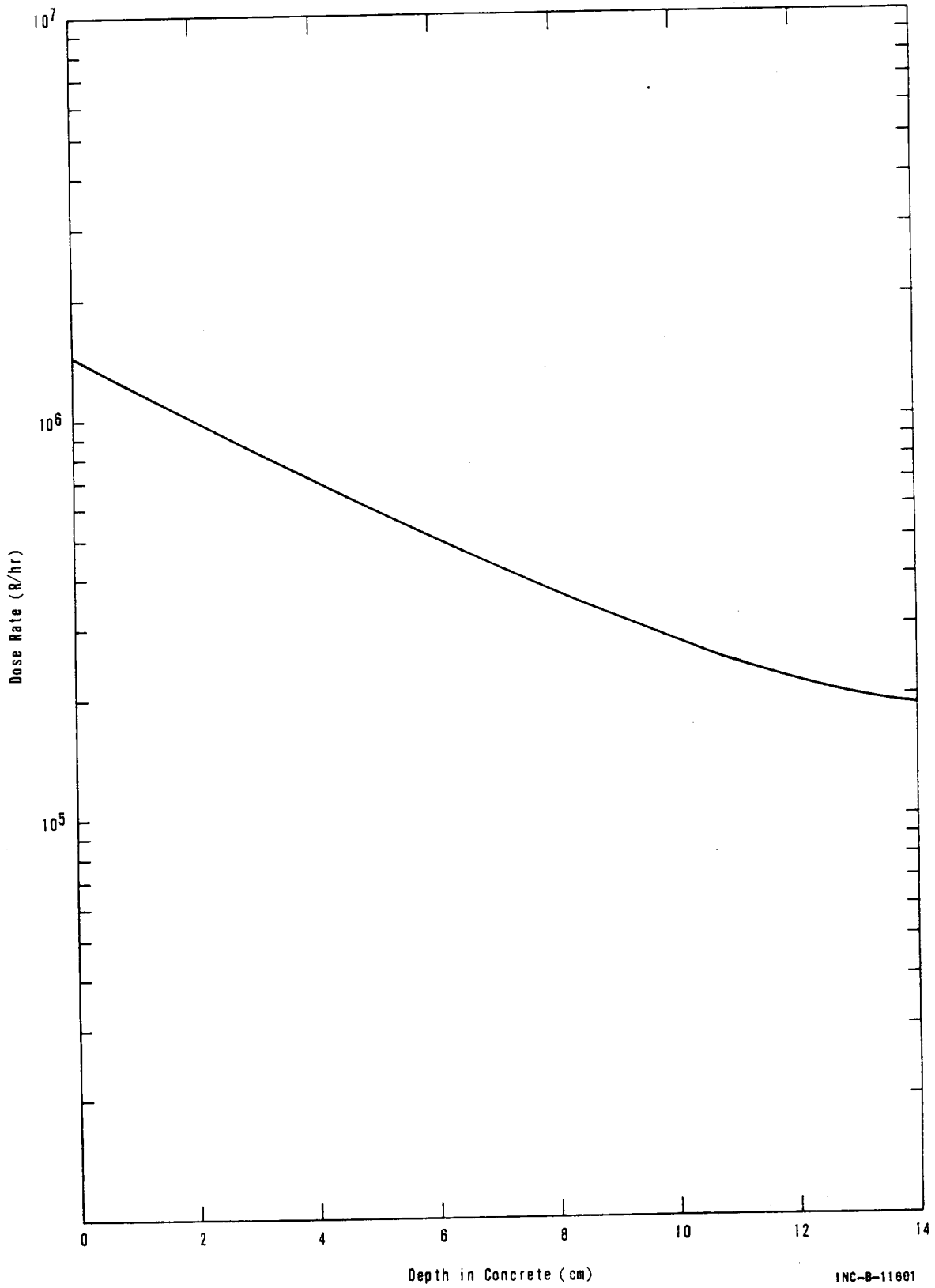

Fig. 13 Gamma dose rate versus depth in concrete.

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presented data on one experiment by Price, Horton, and Spinney [2] where quantized gamma plus neutron radiation exposures was graphed versus percent loss of strength of the concrete samples. Figure 14 compares the loss of strength versus exposure of our "B" test samples to the data for the Price, Horton, and Spinney samples. Casual inspection indicates that neutron exposures cause an added 32% loss of strength for any given gamma exposure. However, this difference may be caused by any number of variables that are unaccounted for such as composition of the cement, type and size of the aggregate, and/or the irradiation environment. Of interest is the fact that the Price et al specimens were irradiated at 50°C for up to two years while our samples were irradiated at 16 to 24°C (calculated) for up to 10 years. If the elevated concrete temperatures are the major cause of loss of strength as Clark indicates, what is the primary factor for loss of strength in our samples? In any event, we can be assured that radiation is a necessary adjunct for the loss of strength we observed, if not the basic cause.

1.14 Conclusions. The compressive strength of the concrete samples has been adversely affected by the gamma irradiation exposures. It also appears that this loss of strength is affected by the position of the test samples in the holder cans. It is believed that the ability of water to circulate around the top samples in the holders contributed to the loss of strength of those concrete samples positioned in the top of the cans. A simple log-linear extrapolation of the test data indicates that the loss of compressive strength in three-inch diameter cylinders of concrete, submerged in circulating water, will approach 100% after exposure to 10^{12} R of gamma radiation. There is no way to determine from the available data whether the loss of strength is due solely to deterioration of the concrete from the surface, or if the structural defects also are generated and integral within the body of the concrete structure.

Although there does not appear to be a problem at this time, it is suggested that the structural soundness of the MTR and gamma facility canal walls be checked periodically to assure that the integrity of the canals is maintained.

1.2 Reactor Health Physics Methods Used to Calculate Airborne Waste Effluent Activities (J. F. Sommers)

A new method was developed to determine airborne waste effluent activities at TRA for reporting purposes. A description of the method follows:

1.21 General

(1) Obtain grab samples as often as necessary to keep the stack monitor calibration factors current and accurate.

(2) Grab samples will be filters for particulate and fission gas determinations, and gas bottles for Ar-41 and possible foam fission gases.

(3) Count the grab samples soon after removal under standardized conditions and calculate isotopic activities. In the reactor areas, the predominant activities will be Cs-138, Rb-88, Rb-89, their parents, and Ar-41.

(4) The curies per day for each type of activity released may be calculated using the following general formula (a). Symbols used in all calculations are listed in 1.3.

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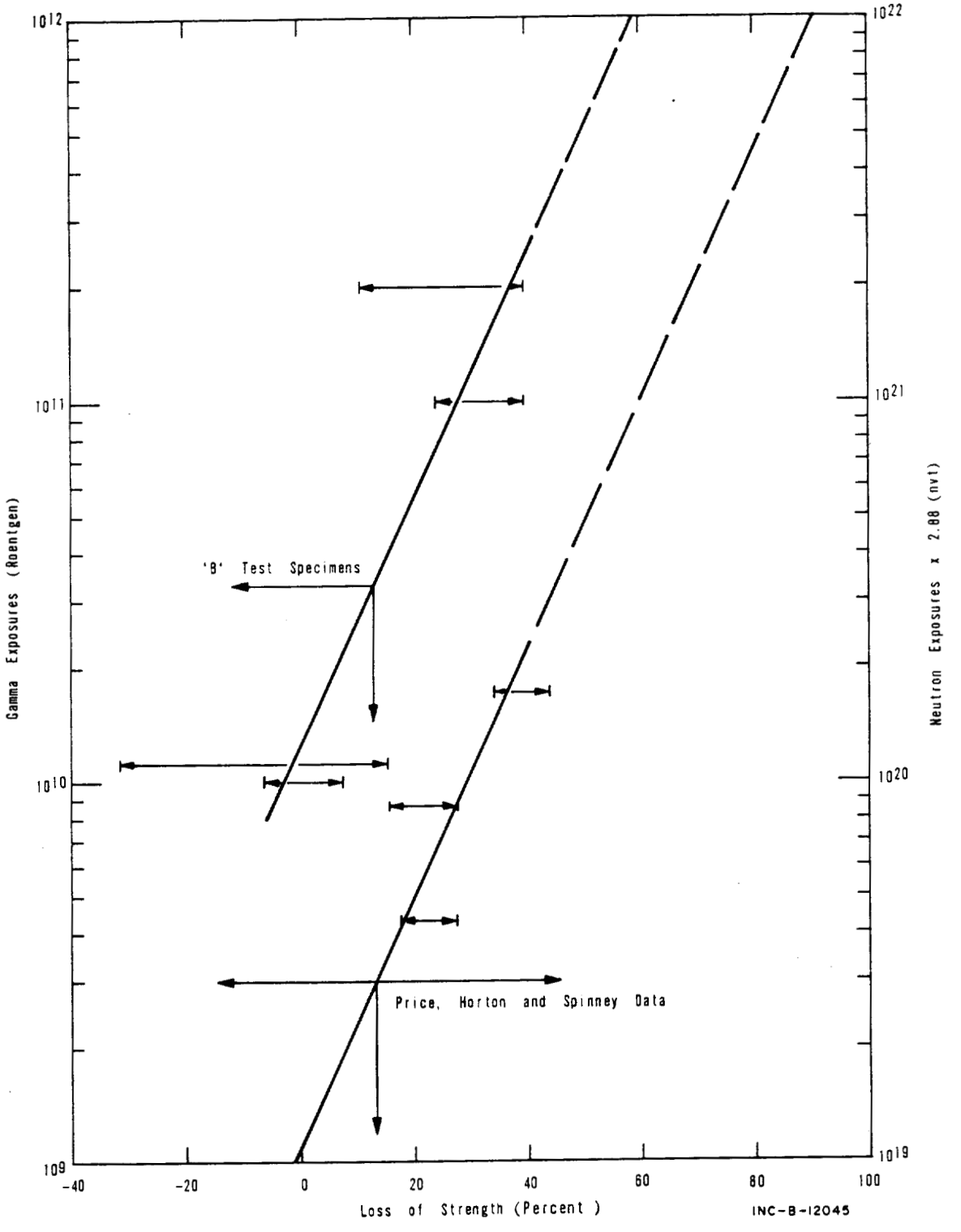

Fig. 14 Comparison of percent loss of strength versus gamma or gamma plus neutron exposures.

(a) $A_i = 7.75 \times 10^{-10} (a_o)_i R/rt_1$ (Ci/d).

Note that the stack discharge rate is in terms of scfm while the sampling rate is in actual cfm.

1.22 Corrections to the General Formula

(1) The general formula should be corrected from sample analysis for growth and/or decay of activities when the isotopes being monitored have half-lives less than an order of magnitude more than the sampling and counting intervals. For fission gas daughters, three decay and one growth correction may be necessary.

(2) The decay of particulate activities on a filter sample during the counting interval may be corrected by

(b) $a_2 = \lambda_2 a_3 t_3/(1 - e^{-\lambda_2 t_3})$ (dis/min).

(3) The correction for decay between the end of the sampling period and the start of the counting period is made using

(c) $a_1 = a_2 e^{\lambda_2 t_2}$ (dis/min).

(4) The effective particulate activity that arrived at the sampler during the sampling period can be expressed as

(d) $a_o = \lambda_2 a_1 t_1/(1 - e^{-\lambda_2 t_1})$ (dis/min) or

$b_o = 1.59 \times 10^{-11} \lambda_2 a_1/r(1 - e^{-\lambda_2 t_1})$ (μCi/cc).

(5) Corrections for the differences in transit times from the point of production of parent activity in the stack gas to its daughter's collection at the sampler or release out the top of the stack can be made using

(e) $a_i = a_o \sigma_2/\sigma_1$ (dis/min) or as shown in formula (l).

(6) Formula (a) must be corrected for (e) so

(f) $A_i = 7.75 \times 10^{-10} a_i R/r_{t_1}$ (Ci/day).

(7) An instrument calibration factor can then be determined from the count rate of the stack monitor at the time sampling was done, the stack gas emission rate and analyses for the various isotopes in the stack gas that affect the reading of the monitor. Thus

(g) $f = \sum_{i=1}^{n} (A_i) / cR \left(\frac{\text{Ci/day}}{(\text{scfm})\ (\text{c/min})}\right)$.

Note the sampler flow rate must remain constant for this formula to work correctly.

(8) The stack emission rate can then be determined at any time (t) from the count-rate reading of the stack monitor and a determination of the stack discharge rate in scfm. Thus

$$\text{(h)}\quad A_t = f\, c_t\, R_t \text{ (Ci/day).}$$

1.23 Determination of Parent Gas Specific Activities

(1) The specific activities of the parent gases may be calculated from the quantities of daughter activities collected on filter samples.

(2) The time rate of change of the daughter isotope on the filter is a function of the decay rate of the parent isotope in the effluent stream and the time for production of the parent into the gas effluent to passage through the sample filter. Thus

$$\text{(i)}\quad a_1 = g\sigma_1 \left(1 - e^{-\lambda_2 t_1}\right) \text{ (dis/min).}$$

(3) The production rate (g) of the parent atoms into the effluent gas can thus be determined by

$$\text{(j)}\quad g = a_1/\sigma_1 \left(1 - e^{-\lambda_2 t_1}\right) \text{ (atoms/min).}$$

(4) The parent gas will have some decay in its passage to the top of the stack so the specific activity of a parent isotope in the stack gas at the top of the stack is

$$\text{(k)}\quad k_i = 1.59 \times 10^{-11}\, \lambda_1 a_1 e^{-\lambda_1 \tau_2}/r\sigma_1 \left(1 - e^{-\lambda_2 t_1}\right) \text{ (}\mu\text{Ci/cc).}$$

(5) In the same terms (μc/cc), the specific activity of the daughter isotope in the stack gas arriving at the top of the stack would be

$$\text{(l)}\quad b_i = 1.59 \times 10^{-11}\, \lambda_2 a_1 \sigma_2/r\sigma_1 \left(1 - e^{-\lambda_2 t_1}\right) \text{ (}\mu\text{Ci/cc).}$$

(6) The ratio of the parent to daughter activities would be

$$\text{(m)}\quad B = k_i/b_i = \frac{\lambda_1}{\lambda_2} \frac{e^{-\lambda_1 \tau_2}}{\left(1 - e^{-\lambda_1 \tau_2}\right)} = \frac{T_2}{T_1} \frac{e^{-\lambda_1 \tau_2}}{\sigma_2}.$$

(7) Thus the specific activity of the parent in the stack gas can be determined by

$$\text{(n)}\quad k_i = Bb_i = 1.59 \times 10^{-11}\, B\, \lambda_2 a_1 \sigma_2/r\sigma_1 \left(1 - e^{-\lambda_2 t_1}\right) \text{ (}\mu\text{Ci/cc).}$$

(8) Notice that if B is >1, there is a good possibility that the specific activity of k can be determined by direct measurement on gas samples. However,

if B is small, the integrating effect of using a filter will make the indirect method of formula (n) better because of more sensitivity and probably improved accuracy.

1.24 Determination of the Delay Time in System

(1) Where k_i and b_i can be determined directly, τ_2 can be calculated and checked for conformance with the value used in calculating k_i and/or b_i for other isotopes. Thus

$$\text{(o)} \quad \tau_2 = \frac{1}{\lambda_1} \ln \left(\frac{\lambda_1}{\lambda_2 B} + 1\right) \text{ (min).}$$

(2) Check the value of τ_2 frequently to assure that the value used is a good average value.

1.3 Symbols Used in Stack Effluent Formulas

A_i = (Ci/day) of isotope (i) discharge from stack.

$(a_o)_i$ = (dis/min) (effective) of isotope (i) arriving at the sampling filter during sampling period t_1.

R_a = 1.2 R (cfm) discharge rate from stack.

R = (scfm) discharge rate from stack.

r = (cfm) sampling rate -- a constant.

t_1 = (min) sampling time.

a_1 = (dis/min) of isotope (i) on sample filter at the end of t_1.

a_2 = (dis/min) of isotope (i) on sample filter at the end of delay time (t_2) from sampling to counting.

a_3 = (dis/min) (average) of isotope (i) on filter during counting period (t_3).

t_2 = (min) delay from sampling to counting.

t_3 = (min) counting time.

λ_2 = (min^{-1}) decay constant of isotope (i).

λ_1 = (min^{-1}) decay constant of isotope (j) which is the parent of isotope (i).

τ_1 = (min) time for the parent to flow from production point to sample filter.

τ_2 = (min) time for the parent to flow from production point to the top of the stack.

σ_1 = $(1 - e^{-\lambda_1 \tau_1})$ (a fractional factor).

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$\sigma_2 = (1 - e^{-\lambda_1 \tau_2})$ (a fractional factor).

a_i = (dis/min) (effective) of isotope (i) arriving at the top of the stack during sampling interval (t_1) in ($r \times t_1$) ft^3 of air.

f = (Ci-min/count-day-scfm) a calibration factor.

c = (counts/min) (average) from stack monitor during a sampling period (t_1).

A_t = (Ci/day) total discharge rate from stack as determined by use of the calibration factor (f), the stack monitor reading c_t, and stack flow rate R_t.

g = (atoms/min) production rate of the parent isotope into stack sample stream.

k_i = (μCi/cc) of parent gas in effluent at the top of the stack.

b_i = (μCi/cc) of daughter activity released in the effluent at the top of the stack.

B = k_i/b_i (a ratio).

2. MTR HEALTH PHYSICS

2.1 Preparation of ^{233}Pa Sample at the ARA Hot Cells (J. H. Metcalf, D. Majors)

The hot cells at the Auxiliary Reactor Area were extensively modified for work with high specific activity alpha-emitting materials. The primary purpose for the modification was to provide facilities for the preparation of a ^{233}Pa sample. Major modifications to the ARA facility included changing the cell ventilation system and adding isolation boxes to the interior of the largest of the two cells (hereinafter designated Cell No. 1). The small cell, Cell No. 2, was used as a secondary or transfer cell. Samples were brought into Cell No. 2 in casks, unloaded remotely, and transferred into the isolation boxes via a transfer tray between the two cells. Samples were removed from the boxes by reversing this procedure. (Refer to Figure 15 for the general layout of the isolation boxes within the cells.)

The cell ventilation system was modified so that air passed from each isolation box through two absolute prefilters (in parallel), a two-inch charcoal filter, and an absolute filter before being exhausted to the atmosphere. Air from the cells was passed through roughing filters and an absolute filter before release (see Figure 16).

Two separate exhaust stacks were used in this system -- one discharged the air from the isolation boxes and Cell No. 1, and air from Cell No. 2 was released through the other. Air from both stacks passed through a common sample line to a Tracerlab MAP-1B continuous particulate monitor (with $\beta\gamma$ and α detecting units), to a makeshift iodine monitor, thence to a Tracerlab

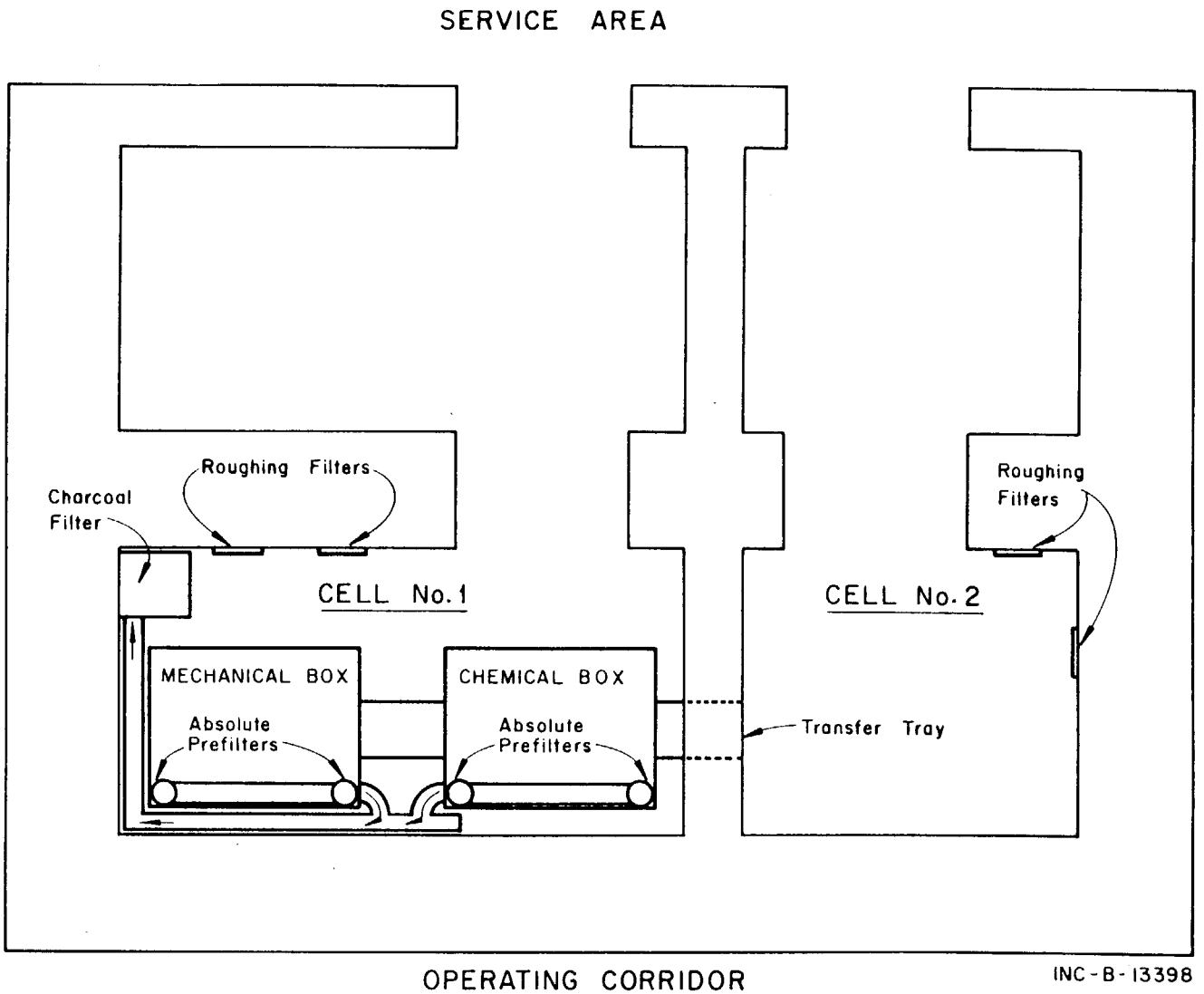

Fig. 15 General layout of ARA hot cells.

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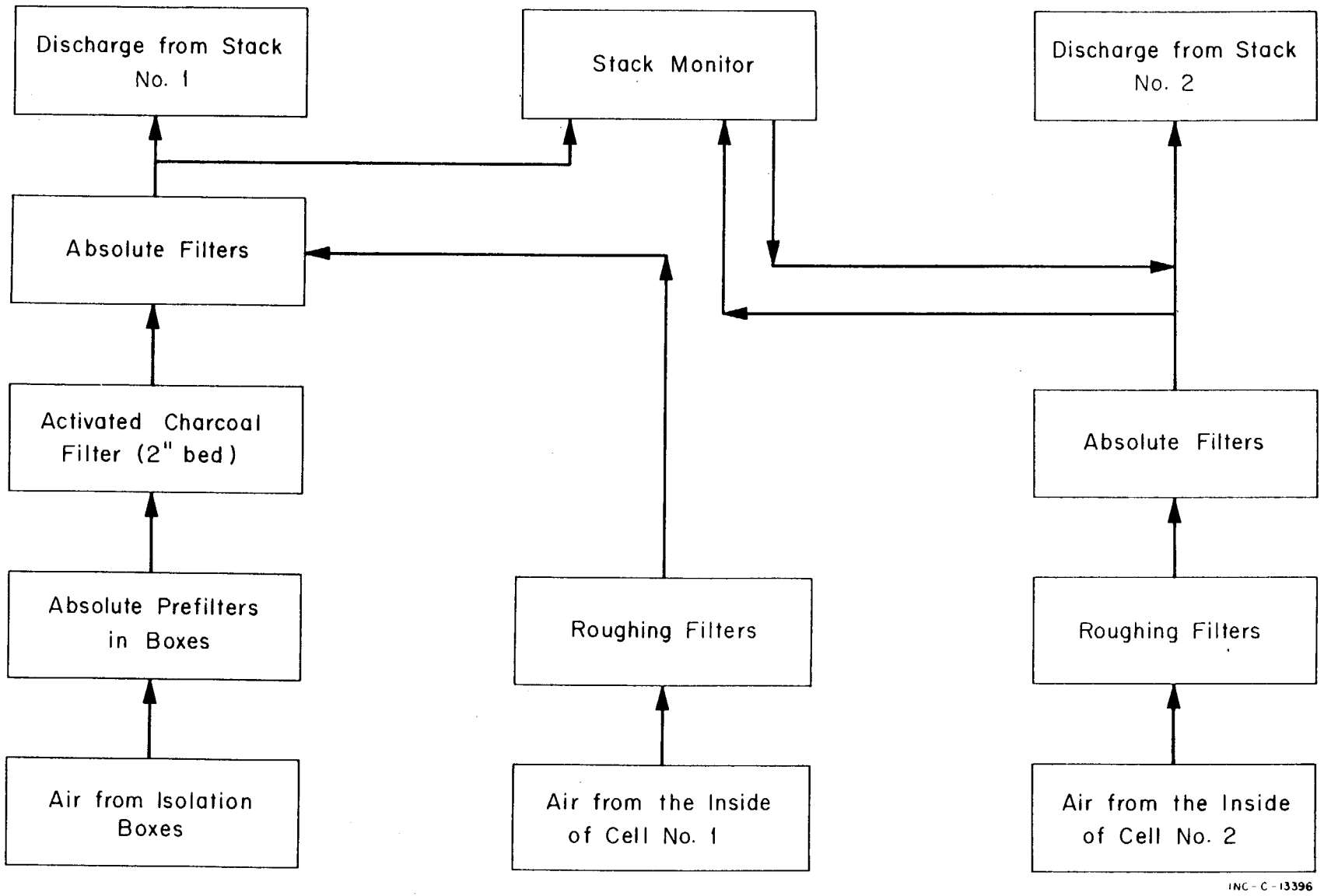

Fig. 16 Block diagram of the cell ventilation system.

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MG-1A gas sampler, then back to the stack. The iodine monitor was fabricated of three-inch black iron pipe and was shielded with lead shot. The filter medium was a Cesco charcoal cartridge, and an indication of the activity collected was given continuously by a GM detector. The iodine detection system as orginally installed was too efficient, with the result that the instrument was pegged (by radioactive noble gases) throughout the time when the majority of the iodine was expected to be evolved. Total iodine released from the stack was estimated by gamma analysis of the charcoal filter.

The base material used for the preparation of the ^{233}Pa sample was ^{232}Th metal. Twelve capsules containing a total of 816 grams of thorium metal were irradiated in the MTR for three cycles. These samples were removed from the reactor and transferred to the ARA Hot Cell. The capsules were transferred from the cask to the mechanical isolation box, where they were opened and the thorium metal removed. Some radioactive noble gases were released during this step in the procedure.

After all the capsules had been opened there were 48 pieces of thorium. One piece of this metal pegged the in-cell monitor (greater than 10^5 R/hr at approximately four inches). Radiation emanating from the 48 pieces of metal was therefore estimated to be greater than 5×10^6 R/hr at four inches.

Dissolution of the thorium started after some minor problems with the manipulators. The metal was dissolved using a mixture of nitric and hydrofluoric acids. Iodine was expected to be evolved and released during this procedure, but very little iodine and large amounts of noble gases were detected. The dissolution process continued throughout the rest of the day and work was stopped at about midnight.

After the thorium metal had been dissolved in the nitric acid, hydrofluoric acid mixture, potassium permanganate, and manganese nitrate were added to precipitate manganese dioxide. The protactinium was expected to be carried along with the precipitate. The precipitate was filtered and then dissolved in sulfuric acid. This procedure was repeated two more times. At this point the protactinium present in a sulfuric acid solution was pure.

Iodic acid was added to the solution to precipitate protactinium as an iodate of unknown composition. The precipitate was filtered, dried, and then thermally decomposed to form protactinium oxide. The oxide was mixed with a small amount of aluminum powder and pressed. Then the sample was sandwiched between two nickel foils and placed in a holder which was used primarily for heat transfer. The completed assembly was loaded into a shipping cask and flown to its destination. Two samples were also prepared for the MTR Fast Neutron Chopper.

The first indication of a problem with the ventilation system occurred during the manganese precipitation. At this time approximately 2 a.a.u.'s[a] of activity, primarily ^{131}I, were detected in the service area behind the cells and in the chemistry lab. Airborne iodine (^{131}I and ^{132}I) was detected in the

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[a] 1 a.a.u. = 3×10^{-9} μCi/cc.

building periodically throughout the rest of the time the sample was being prepared. Charcoal respirators were worn by all personnel when air activity was present.

Investigation later revealed that the air activity problem was because of the design of the cell ventilation system. One of the design requirements was that the flow through the door to Cell No. 2, when open, would be 70 ft/min. This was accomplished by using a two-speed motor on the blower ventilating this cell. A limit switch on the cell door shifted the motor to high speed as soon as the door started to open. The motor stayed on high speed until the door was almost completely closed. This increase in flow caused iodine to be drawn out of the chemical isolation box into Cell No. 2 and out the stack. Thus a path was available whereby iodine could bypass the charcoal filter. No trouble was experienced with radioactive nuclides other than iodine being drawn out of the isolation boxes.

Smoke bombs were used to determine the path of the plume after it left the stack. These tests showed that the plume was being carried to the ground as a result of the building downwash effect. As the plume passed over the edge of the building, activity contained therein was drawn into the building through the air supply system. Each time wind direction was such that the plume passed over the building air intakes, air activity was experienced in the building.

Although airborne iodine activity was present intermittently in the hot cell building throughout sample preparation, no internal exposures resulted. Each time that the presence of iodine was indicated by constant air monitors in the building, urine samples were requested from all personnel. These were analyzed for iodine on a gamma spectrometer, and all samples were negative. Also, all personnel involved were counted with the TRA thyroid counter following completion of the project. Again the results were negative.

All openings to the cells have been taped, the exhaust stacks have been capped, and all equipment has been turned off and placed in a standby condition. No cleanup has been attempted inside the isolation boxes or in the cells.

2.2 Biomedical Application of Neutrons (B. R. Baldwin)

The manager of the Nuclear Technology Branch, R. M. Brugger, requested advice and services of MTR Health Physics Section for a special project in the field of biomedical application of neutrons. A model of a human skull was constructed, filled with tissue-equivalent solution, and placed in a neutron beam. The neutron flux and gamma ray intensity were measured as a function of position. Recommendations to Dr. Brugger on tissue-equivalent materials and dosimetry were incorporated in the study.

2.3 Monitoring Neutrons Filtered Through Scandium (B. R. Baldwin)

The Nuclear Technology Branch is performing experiments with neutrons which have been filtered through scandium. The flux is approximately 10^5 n/cm^2/sec of 2 keV neutrons and 10^5 n/cm^2/sec of 0.5 MeV neutrons. The 2 keV neutrons present an unusual monitoring problem since neither thermal nor fast neutron instrumentation responds adequately. Instrumentation techniques are still being investigated.

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2.4 MTR Beam Hole Resurvey (J. H. Metcalf)

A resurvey of the MTR beam hole facilities was completed (Metc-15-68) characterizing radiation hazards associated with them and limitations provided by, or placed upon, the associated experiments.

3. TAN HEALTH PHYSICS

(R. S. Peterson)

3.1 Health Physics

The TAN Hot Shop, RML, and HCA work during the calendar year 1968 was made up of many small jobs rather than any one large project. The RML major cell modification was completed and the cell went hot in February. Jobs completed include 693 radioactive shipments either sent out or received. The types of work include disassembly, repair and, reassembly of radioactive systems; sample preparation and examination; destructive disassembly; machining, cutting, and related cell work.

The TAN Machine Shop is doing increased amounts of work on radioactive systems. To date, contamination has been well confined on these jobs by close cooperation; Health Physics surveillance and the equipment has been readily decontaminated at the completion of each job. Radiation exposures have been maintained to well below the administrative guide levels.

3.2 Decontamination

The major items which were decontaminated, chemically cleaned, and/or sandblasted include: 107 casks, 10 monitors, 16 pumps, 5 tanks, 10 CPP filter housings, 9 manipulators, and 153 miscellaneous large items. Radiation exposure to the decontamination technicians was reduced by transferring most of the low level decontamination work to a section of the chemical cleaning room -- thereby removing the technician from the high background of the decontamination room.

The work load in the decontamination area has increased over previous years and most of the increase is attributed to additional work being sent in by other contractors.

4. ETR HEALTH PHYSICS

4.1 Decontamination of Reactor Tank (D. K. Jenson)

The decontamination of the upper reactor tank surfaces was attempted on the first day of reactor shutdown on April 4, 1968. The decontamination solution was a mixture of oxalic acid and Fome-Add (Fome-Add is manufactured by DuBois). The mixture was applied with the use of the foam gun.

Radiation surveys were made of the reactor tank before and after the application of foam. Radiation surveys indicated a decontamination factor of

approximately two for a single application. This small reduction in the radiation field has not warranted additional attempts.

4.2 Report Review (T. C. Roy)

A review of the report "Patterns in Radiography Incidents and Over-exposures" by E. N. Cramer and R. L. Scott was made by T. C. Roy, Health and Safety Engineer, and K. C. Carroll, Quality Assurance Supervisor. This review may be found in Roy-12-68.

Suggestions were made with regard to a Radiation Safety Training Program, source control, and administrative control of radiographic operations.

4.21 It was suggested that a Radiation Safety Training Program should incorporate the following:

(1) All personnel who are radiographers or assistant radiographers should attend a professional radiography training school. This school should have a course structure approved by the AEC Division of Licensing and Regulations.

(2) There should be emphasis on the limitations and reading of portable survey instruments.

(3) The calculations of possible exposure doses to personnel from several simulated accident conditions could be covered.

(4) A discussion of the biological effects on man from large exposure doses should be given. Films showing the effect of partial body and whole body exposure of man-to-gamma radiation could be shown. The effects of a massive exposure to radiation of the hands and a whole body exposure of about 5000 rem, for example, would be very effective.

(5) A report similar to the one being reviewed, but containing only those incidents involving biological change, should be given to all radiographers. A very vivid description should be presented concerning the biological damage which took place in these incidents.

4.22 The following source control changes were suggested:

(1) A small radiation detector should be supplied within the radiographic case and the associated readout meter installed exterior to the case. The radiographer could measure the radiation with the source in the "IN" position. After the exposure and returning the source to the "IN" position, the meter reading could be compared with the initial reading. If the meter readings compared, the source has then been returned to its proper position.

(2) An unattached source in a guide tube is much more possible with equipment in which the source drive cable is routinely detached from the source capsule for transport and storage. The problem arises when the drive mechanism is attached to the shield without the source capsule being attached to the drive cable, the source capsule can be driven out but not retrieved by the drive cable. This potential hazard is created every time the source is connected for use. A structural design change to preclude the connecting of the drive

mechanism without the drive cable being attached to the source capsule may be necessary to reduce this type of accident.

(3) The radiographer should set up for a radiography exposure but perform a radiation survey with the source capsule driven out and retrieved by the drive cable before beginning the actual radiography exposure.

(4) Audible portable alarms readily usable in field operations would help overcome the problem of radiographers who are hurrying to meet schedules or with preoccupied minds entering cubicles without a monitoring instrument. A chirper alarm would be one such instrument.

4.23 Certain recommendations were made with respect to AEC and agreement state licensing and control.

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II. HEALTH AND SAFETY RESEARCH

1. SKIN ADSORPTION OF IODINE

(J. L. Clark, C. E. Nichols, H. V. Piltingsrud)

This project is intended to evaluate the hazard to the human body resulting from the permeation of radioactive iodine or iodine compounds through the skin. The project effort will establish the rate at which the radioactive iodine or iodine compounds move through the skin, ie, the permeation rate, and find those factors which influence the movement through the skin. Factors to be investigated are the chemical form of the iodine, the exposure period, the concentration of the iodine species, the temperature and humidity of the air during the exposure, and the type of skin. In addition, the iodine permeability of various materials used in protective equipment and sample packaging will be measured.

In order to make the permeability measurements, a section of excised skin is exposed to a constant concentration of iodine and the amount which penetrates through the skin section is periodically collected and measured. These measurements establish how long it takes iodine to penetrate the skin and establish the equilibrium penetration rate for iodine.

During the past year, a number of changes and improvements in the apparatus and technique have been made which resulted in a more efficient apparatus. In the previous apparatus, an initial charge of radioactive iodine was introduced into the exposure chamber at the start of a measurement. Considerable difficulty had been encountered in keeping a constant concentration of radioactive iodine in the metal chamber, even though the walls were coated with fluoroplastic. Deposition of iodine on the walls, especially at low iodine concentrations, caused a rapidly decreasing iodine level. A new all-glass system, using a constant flow of radioactive iodine, was built. With this system, reasonably stable iodine concentrations can be maintained over the skin surface for periods of one or two hours. Figure 17 is a sketch of this apparatus.

The initial measurements were made on the epidermal layer of excised human skin. Other workers had demonstrated that this epidermal layer, composed mostly of dead cells, was the skin's principal penetration barrier. Materials passing this layer were rapidly adsorbed into the body's circulation system. Penetration measurements made using separated epidermal layers were found to correspond with measurements made in vivo. Since skin is a rather effective barrier, it was desirable to use iodine of very high specific activities to provide enough iodine activity passing through the skin for accurate measurements of the permeability. These in vitro measurements eliminated the risk of a high human exposure in the initial permeation work. Figure 18 is a typical measurement of the iodine penetration of a polyvinyl chloride film.

An improved sampling device was used to measure the purity of the radioiodine in the test stream. Other workers have shown that when elemental iodine is released into the air at low concentrations it is partially converted to organic iodides and other forms. The new sampler provides a method for determining how much organic iodide and how much elemental iodine are

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present in the test stream. It is a modification of a sampler developed for the LOFT project by Idaho Nuclear Corporation's Analytical Chemistry Branch. It consists of three selective adsorber layers. The first, iodide loaded hydrous zirconium oxide, traps elemental iodine. The second, silver loaded 13X molecular sieve, retains the organic iodides. The final layer, iodide impregnated charcoal, serves as a backup for the other two layers.

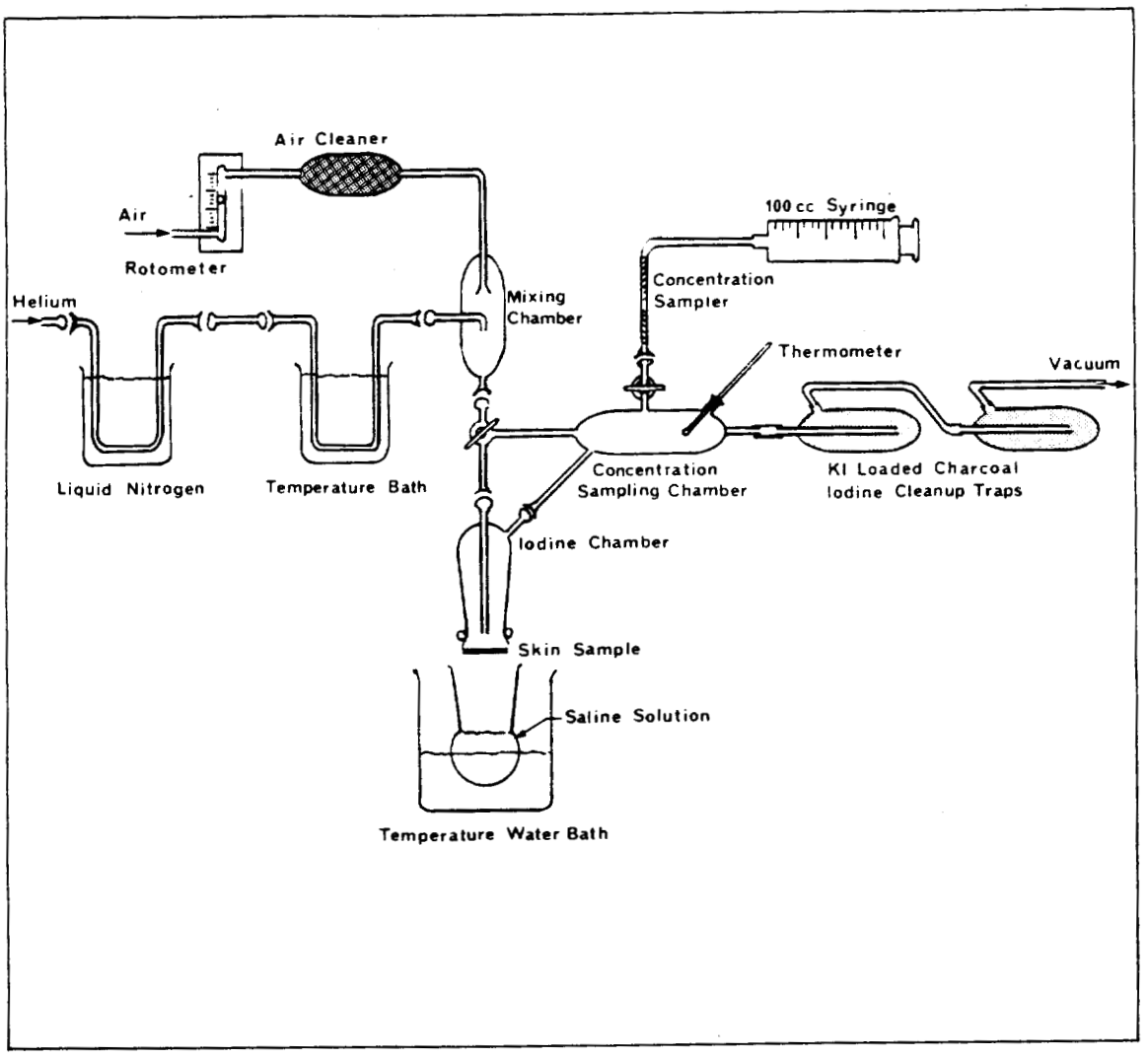

Fig. 17 Iodine exposure apparatus for excised skin.

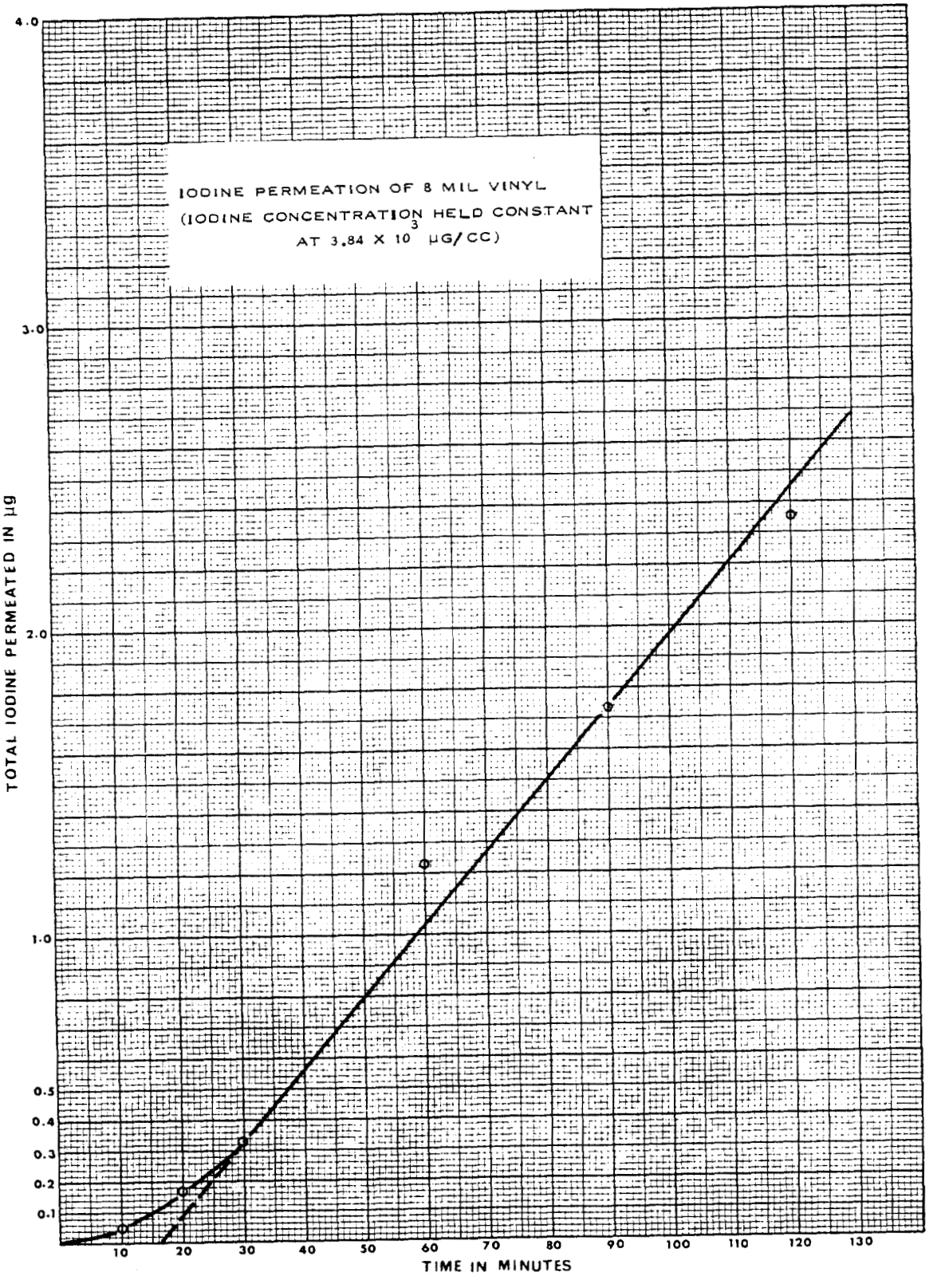

Fig. 18 Iodine permeation of 8 ml vinyl.

2. COMPUTER APPLICATIONS

(F. O. Cartan, D. K. Jenson, H. K. Peterson)

The usefulness of a time sharing computer terminal to simplify the preparation of routine Health and Safety Branch reports is being investigated. Possible applications involve preparation of Safety Analysis and Reactor Waste Reports and the analysis of various monitoring systems. The advantages would be faster results, fewer errors, and more sophisticated calculations.

A small amount of computer time was purchased from Computer Update, Inc., a Salt Lake based commercial time sharing service. A remote terminal belonging to the Nuclear Technology Branch is being used for evaluation trials. This terminal provides immediate access to a computer using FORTRAN, AID, BASIC, and other common programming aids. The evaluation is in progress.

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III. HEALTH AND SAFETY DESIGN REVIEW

(D. A. Davis)

1. DESIGN REVIEW

Health and Safety design reviews accomplished during 1968 include 74 proposals to the operating committee, 29 Title I reviews, 14 Title II reviews, 44 design criteria, 51 construction designs, and 8 miscellaneous reviews. The projects may or may not have been carried to completion; some have, some are pending, and others were dropped. Of those reviews, only projects with special Health and Safety significance are listed below. A very brief indication of the interesting aspects is included.

ARA-627 Ventilation -- Control of ammonia fumes.

ARA Microprobe Facility -- New facility.

ARA GCRF Instrument Lab -- New facility.

CFA-690 Exposure Chamber -- HEPA filter and charcoal adsorber design.

CFA Special Services Building -- New facility.

CFA-688 Communications Addition -- Fire detection system included.

CFA-689 Machine Exhaust Modification -- Control of ammonia fumes.

CFA-617, -618 Fire Detection System -- Protection of sleeping quarters.

CFA-656 Dynamic Corrosion Loop -- High pressures and temperatures.

CFA Standard Calibration Lab -- New facility.

CFA-664 Steam Cleaner -- Propane installation.

CFA Photo Laboratory -- New facility.

CFA-684 Shield Enclosure -- Direct radiation and contamination control.

CFA-684-E Storm Shed -- Corrects hazardous arrangement.

CFA-688 RIOT Station Modification -- Fire protection considerations.

CPP-620 Roof Hatch -- Corrects hazardous arrangement.

CPP-627 RAF Cave Modification -- Radioactive contamination and radiation control.

CPP-637 Ventilation Modifications -- Upgrade control of toxic materials.

CPP-602 Product Denitration -- Radioactive contamination control.

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CPP Tank Farm Valve Box -- Radiation and radioactive contamination control.

CPP-640 LETS -- Ventilation, radioactive contamination, radiation control, hazard analysis.

CPP-640 Electrolitic Dissolver -- Radiation, radioactive contamination control, process hazards, hazard analysis.

CPP-627 New Fume Hood -- Control of toxic materials.

CPP-608 Modification -- Improved fire protection.

CPP-627 H and V Modification -- Upgrade control of toxic materials.

CPP-602 Perchloric Hoods -- Ventilation improvements.

CPP-602 Fume Hood Ventilation -- Increased face velocities.

CPP-637 Glass Shop -- New facility.

CPP-633 Inbed Combustion -- New process.

CPP G-Cell Feed Clarification -- Radiation control.

CPP PEW -- Ion exchange hazard control.

TAN RML Fire Protection Upgrading -- Cell fire protection.

TAN-607 LOFT Sample Processing System -- Radiation and radioactive contamination control.

TAN Evaporator Modification -- Radioactive materials and waste management.

TRA Fuel Inspection Building -- New facility.

TRA ATR Primary Pump H and V -- Ventilation and noise considerations.

TRA-656 Valve Test Facility -- Pressure testing.

TRA ARMF-I Modification -- Hazard analysis.

TRA ATR HVE-4 Modification -- Noise control.

TRA-606 Warehouse -- New facility.

TRA-606 RIOT Station -- Fire protection considerations.

TRA-604, -647 Battery Room Ventilation -- Flammable and toxic gas control.

TRA ETR KAPL H-10 Modification -- Noise control.

TRA-657 Hot Cell Addition -- Radioactive contamination control, materials handling, fire protection.

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2. FIRE AND SAFETY EQUIPMENT SAFETY INSPECTION PROGRAM

During 1968, 7,026 inspections of 3,530 pieces of equipment or systems were conducted. The work was accomplished by two full-time inspectors and one summer employee. A total of 4,256 manhours were invested to perform the prescribed number of inspections necessary to assure reliable emergency equipment.

Early in 1968, equipment inspection frequencies were adjusted on the basis of previous experience with the equipment being serviced. An estimated eight man-weeks were gained by judicious reductions in inspection procedures and frequency. As a consequence, it was possible to complete our inspection schedule and adsorb the added work load occasioned by the addition of new buildings and equipment, without increasing manpower. It also was possible to inaugurate a reinspection program for operational checks of equipment previously found defective.

During the year a study was made to determine the various methods and costs for having carbon dioxide extinguisher cylinders and breathing air cylinders hydrostatically tested. It was determined that the most economical method was to have the AEC fire department do this work. Subsequently, the fire department obtained the necessary equipment and we are now delivering cylinders to them for testing.

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IV. INDUSTRIAL SAFETY

1. TRA SAFETY

(L. L. Berry)

1.1 TRA Evacuation System

In 1966, a study was begun to consider the adequacy of the TRA evacuation system in relation to the anticipated operation of ATR. As a result of that study, numerous changes and improvements were made in both the procedures and hardware of the evacuation system.

Criteria for situations requiring mandatory evacuation of TRA were developed and incorporated into various pertinent operational documents.

Major changes were made in the hardware associated with the evacuation system. Two staging areas were constructed at TRA, one southwest and one east of the perimeter fences. Two additional buses were assigned to the area, giving emergency transportation for the off-shifts in each staging area.

A 40-foot weather tower was installed at the southwest staging area; work is currently under way to install weather instruments on the tower and to connect the system into readout instruments in reactor control rooms and Health Physics field offices.

Fifteen high-intensity neon directional signs were installed to direct personnel either to the "West" or the "East" staging area depending upon the velocity and wind direction at the time an evacuation occurs. The power for the system is drawn from the ETR fail free power system. The system was designed and installed to comply with IEEE single failure analysis criteria.

The alarm system was evaluated using the IEEE single failure criteria and resulted in the design of a new alarm system. Thirty-eight additional sirens were added to the system to provide coverage in areas which were not adequately covered. The power to the sirens will be removed from the commercial diesel buses and will be tied into the ETR fail free power system.

The lock system on all perimeter fence truck gates was modified to use security wire seals instead of padlocks; culverts were installed at three points to provide level crossing of area ditches; mechanical blocks were installed on all evacuation switches to prevent inadvertent sounding of an alert; identifying hard hats were provided for the evacuation director; and a training program was developed to inform all personnel of changes in the evacuation procedures. Detailed information on procedures and hardware was provided to Operational and Health Physics personnel.

V. INDUSTRIAL HYGIENE

(D. D. Coward, P. B. Anderson, W. Hanson)

1. NONRADIOACTIVE AIR POLLUTION STUDIES

A detailed review was made of proposed Idaho State air pollution regulations. Comments were made to ID for possible inclusion in their brief to the State Air Pollution Control Board prior to final adoption of such regulations.

A complete review was made of the boiler stack sampling procedures used in the nonradioactive air pollution study. As a result a considerably revised set of methods and procedures was developed to obtain more accurate results. The first complete sampling program of every applicable Idaho Nuclear boiler installation was initiated and completed in 1968.

2. RADIOACTIVE WASTE MANAGEMENT REPORT

Industrial Hygiene coordinated the preparation of the first comprehensive report on Idaho Nuclear Radioactive Waste Management Plans and Practices. This report was prepared with considerable assistance from area Health Physics personnel. A formal version of this report is being prepared.

3. SOLVENT EVALUATION STUDY

To promote safer solvent usage in all Idaho Nuclear areas a solvent evaluation study was initiated. Manufacturers of numerous solvents and cleaning agents used in IN areas were contacted. Some manufacturer's representatives viewed the various areas in which solvents are being used. A report on this study will be issued in 1969.

4. ANNUAL INDUSTRIAL HYGIENE SURVEYS

Annual Industrial Hygiene reviews of each Idaho Nuclear area were made on a quarterly basis; 1968 was the first year that every IN area was reviewed.

5. CAFETERIA INSPECTION PROGRAM

An inspection program of all Idaho Nuclear cafeterias was initiated in 1968. The Industrial Hygiene staff reviewed each cafeteria operation on a monthly basis. In addition biannual joint reviews of each facility are made with the area State Health Department sanitarian.

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6. TRAINING PROGRAMS

Considerable time was spent by all members of the Industrial Hygiene Section preparing for and delivering lectures to the first and second year ISU Radiation Protection Technology classes. Subject matter presented by this section included ventilation, noise, toxicology, human anatomy, human physiology, basic biochemistry, and nonionizing radiation (including microwave sources, etc).

In addition numerous lectures on a variety of Industrial Hygiene subjects were presented to various Idaho Nuclear groups.

7. CRITICAL MATERIALS INFORMATION

To obtain up-to-date information dealing with the procurement of materials inherently or otherwise hazardous, a critical materials information program was initiated. The purchasing and stores groups have been supplied with lists of materials considered by Industrial Hygiene to be in the sensitive category and, in turn, have periodically supplied Industrial Hygiene with purchase information.

8. CHARCOAL ADSORBER TESTING

An investigation of methods of leak testing charcoal adsorbers was continued in 1968.

Tetrachlorodifluoroethane (TCDFE) was utilized as a tracer in testing the ARA Hot Cell adsorber installation in March, 1968. On the basis of this testing experience, various methods of sampling and quantitating TCDFE were evaluated. Future sampling on the CF-690 environmental chamber adsorbers will involve taking the samples to a gas chromatograph.

A TCDFE vaporizer unit was designed, and fabrication was completed late in 1968.

9. SEWAGE TREATMENT OPERATIONS

Based on plant records, Industrial Hygiene made a review of all Idaho Nuclear sewage treatment operations for the past eight years. The need for new and/or revised sampling procedures was thus determined. A report will be issued in 1969. In 1968, Industrial Hygiene also assumed technical supervisory responsibilities for all such Idaho Nuclear sewage operations.

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10. MERCURY SPILLS

These incidents numbered about 12 for the year and ranged from minor spills to major upsets requiring confiscation of the clothing of personnel involved. In all incidents, Industrial Hygiene did the initial air monitoring and supplied monitoring and special equipment to those concerned with cleanup operations.

11. POTABLE WATER

Industrial Hygiene has technical supervision over the INC potable water supply systems. Bacteriological analysis is performed at the CF Dispensary. With the exception of two semistagnant storage tanks at TAN and a short period at CPP, satisfactory water quality was maintained.

12. NOISE

The major areas of investigation during 1968 were the ATR basement levels. This investigation continues. Several other noise surveys were made in other areas; however, the noise was found only to be of nuisance levels.

13. CARBON MONOXIDE SURVEYS

These surveys were made in various Idaho Nuclear areas and included cab concentration levels in selected pieces of motorized equipment.

14. VENTILATION SURVEYS

Several surveys of ventilation equipment performance were made, principally at TRA and CPP. Included in these tests were new equipment performance verification tests at CPP.

15. DESIGN REVIEWS

At the request of D. A. Davis, the Industrial Hygiene Section assisted in the design reviews of approximately 25 projects. Several of these reviews involved field measurements, predominantly of ventilation rates.

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VI. HEALTH AND SAFETY SERVICES SECTION

1. REORGANIZATION OF SECTION

The Health and Safety Services Section was initiated in the summer of 1959. It was to pursue any special problems that might be proposed by Operational Health Physics and conduct any necessary health and safety training.

Because of expanded commitments in training and research, the section was divided in September of 1968. The newly formed Health and Safety Research Section was given a research assignment while the educational activities and operations associated problems remained with the Health and Safety Services Section.

2. SUBMICRON PARTICLE SIZING INVESTIGATION

(E. A. King)

Preliminary experimental work was conducted on three different particle sizing methods. A diffusion chamber system was set up using glass microscope slides coated with Formvar. Thirty of the slides were placed side by side on each side of a chamber with a 0.23-inch spacing between each row of slides. Particle-laden air was introduced into the chamber at a low velocity and the particles collected as fallout and plateout. Radioactive particle distribution was determined using autoradiography, and particle sizing was determined using optical and electron microscopy. Experiments to date have encountered some physical difficulties. The experiments conducted indicate that the method is valid, but the results are not as yet sufficiently definitive.

The second system investigated used particle-laden air bubbled through an epoxy resin. The resin was then ultracentrifuged and hardened slowly during centrifuging. The resin-particle matrix was microtomed into thin sheets, autoradiography was made of each thin slice, and optical and electron microscopy used to establish particle size distribution and density. Resins and methods of hardening were given preliminary tests.

Further experimentation was carried out using a migration time ultracentrifuge method with a silicone oil and organic solvent solution as a gas-washing condition.

3. PROPER USE AND LABELING OF RADIOACTIVE WASTE CONTAINERS

(J. L. Clark)

At the request of the Safeguard Review Committee, consideration was given to methods and control of solid radioactive waste disposal in Idaho Nuclear operated areas. Their concern was the areas of potential or existing hazards

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associated with the handling of solid radioactive waste within the plant areas. Subsequently, it was considered worthwhile to research radioactive waste procedures more thoroughly and make recommendations where they seemed needed.

A summary of the areas considered follows: (a) liquids, (b) noncompatible chemicals and pyrophoric materials, (c) weight, (d) excessive radiation, (e) use of "hot" containers for "cold" waste, (f) standardization of containers now in use other than the 2- x 2- x 3-foot boxes, (g) preparation of instructions for use, to be located on or near the containers, and (h) need for explicit written standard practices in the form of an INPP or branch standard practice.

3.1 Investigations

3.11 Liquids. The presence of liquids or at least "wet" materials in the hot waste boxes is without question one of the greatest problems associated with the collection and handling of solid hot waste. On many occasions when attempting to remove cardboard boxes to the dumpsters, the bottoms have fallen out resulting in the contamination of the immediate area. This is usually a result of wet rags, blotting paper, mop heads, etc. The present procedure recommended by Health Physics is the securing of damp items in polyethylene bags prior to their deposition in the boxes, but this procedure obviously lacks proper administrative control and/or enforcement. The possibility of lining all boxes with polyethylene bags was investigated, but the expense (approximately $0.42/bag in quantities of 1,000 for the type and size necessary) would be about $2,000 per year based on approximately 4,000 boxes from IN areas that were buried in 1967.

3.12 Noncompatible Chemicals and Pyrophoric Materials. One should never be unconcerned with potential hazards, which it seems is the very basis for good "defensive" safety programs. The fact is, however, that at the TRA and CPP there have been only three reported minor fires associated with hot waste containers in approximately 15 years of operation. This would indicate commendable practices on the part of the personnel handling waste materials of this category and good established control regulations.

The ICPP has a current standard practice which reflects a satisfactory proven procedure for the safe handling of radioactive organic wastes.

3.13 Excessive Weight. This is the second of the two greatest problems in handling solid waste. Here again, when attempting to remove boxes to dumpsters, excessive weight has caused boxes to collapse. Besides the contamination spill, several "near" personnel injuries have occurred by heavy (usually metal) items falling from the boxes.

Idaho Nuclear has two (3 x 4 x 7 feet) contaminated dumpsters that are shuttled throughout the Site when major jobs are scheduled where gross quantities of contaminated metal will need to be removed. It would appear ideal to have one such dumpster at each major area, but here again, cost versus alternate solutions must be considered.

All hot metal waste or other items of excessive weight should be surveyed by Health Physics, wrapped, and placed directly in the dumpsters. But once

again this policy lacks the necessary administrative control.

3.14 Excessive Radiation. This should not present a problem, as daily radiation surveys are required in most areas where hot waste containers are kept. There also have been no significant exposures to custodians, health physicists, or others who routinely handle hot waste receptacles, except on specific occasions when known excessively hot items had to be removed from an area.

3.15 Use of "Hot" Containers for "Cold" Waste. It is a well known fact that a very large percentage of items in the hot waste containers could be more appropriately placed in cold waste.

Consideration was given to obtaining large "cold" containers to be placed throughout the areas, with construction similar to the standard "hot" containers for fire protection. A half-dozen boxes in each area would run approximately $4,000 to $5,000. At present there is no accurate method of determining disposal costs per unit of volume for cold waste to make the comparisons necessary for proper justification for this consideration. Also, Operations is generally reluctant to have an excess of receptacles and other items on the reactor floor; this appears to be where additional large cold waste containers would be needed most.

3.16 Standardization of Containers. Other than standard 2- x 2- x 3-foot boxes, there are several different kinds of hot waste receptacles now in use. These are found in the labs and Health Physics areas, and range from various sized and shaped waste baskets, "step-on" lid-type containers, to various sizes of garbage-type containers with and without lids. Some are marked with radiation tape and others painted with standard Health Physics colors and symbols.

Due to the large numbers of containers in use, the economic advisability of requiring the disposal of all the containers and requiring a subsequent acquisition of acceptable containers is questionable. As new containers are acquired, they should meet standard specifications.

3.2 Recommendations

(1) A metal sign approximately 2 x 3 feet should be affixed to the inside lid of all standard solid hot waste containers. This sign should bear the legend:

(a) NO METAL WASTE WITHOUT HEALTH PHYSICS APPROVAL

(b) NO LIQUIDS ALLOWED

(c) ALL DAMP ITEMS MUST BE SECURED IN POLYETHYLENE BAGS

(d) CONTACT HEALTH PHYSICS FOR PROPER DISPOSAL OF ALL CHEMICALS OR OTHER ITEMS LISTED ABOVE.

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This sign should be of nominal cost, and attaching it to the container in the manner prescribed will alleviate the moving of a "standard" whenever the container is moved and still be in a conspicuous place.

(2) All metal waste should be checked by Health Physics and, depending on weight, size, and radiation, placed in a box or wrapped and placed directly in the dumpster.

(3) All containers (other than the box type) purchased in the future should be one of two types: (a) "step-on" lid type or (b) approximately 30-gallon garbage cans with "swinging door" type lids, both of which are available through the stores warehouse. Further, these containers should be painted yellow with magenta radiation symbols on at least two sides. All nonstandard type receptacles should be lined with available polyethylene bags.

(4) The CPP method of disposing of liquid organic waste should be adopted as standard branch policy (ICPP-SP 5.62.4).

(5) Through individual plant Health Physics supervision, explicit instructions should be disseminated via proper channels to yardmen, custodians, Health Physics technicians, Operations, and others associated with hot waste handling as to proper practices. This possibly could be done through safety meetings.

(6) A branch standard practice and/or INPP covering plant disposal of all types of radioactive waste should be prepared by responsible Health and Safety personnel.

4. EMERGENCY HANDLING OF BIOLOGICAL SAMPLES AND DOSIMETRY DEVICES

(E. A. King)

4.1 Introduction

Following a query from ID, a study was conducted of the emergency handling of biological samples and dosimetry devices. This resulted in certain procedural recommendations.

It was recommended that three categories be established for the handling of biological samples: (a) urgent conditions requiring immediate results, (b) special conditions requiring results as soon as Health Services Laboratory personnel can provide them during normal working hours, and (c) routine analysis of routine samples.

4.2 Category Recommendations

It was recommended that the following procedures be established for each category.

4.21 Urgent Situations

(a) Whole Body Counting, Urine Analysis, and Fecal Analysis. The responsible health physicist or an informed alternate should call the Health Services Laboratory (HSL) during daytime operations or a designated representative on off-shifts and on weekends. He should provide the following information: (a) who is calling and the location and company; (b) the name of the individual(s) involved and the type of analysis required, ie, whole body counting, urinalysis, fecal analysis, and for what radionuclides; (c) the name and phone number of the person to receive results.

The responsible health physicist should also correlate time schedules with HSL personnel and provide transportation to CF-690 for the individual(s) and sample(s) involved. He should fill out the required forms, ID-104 and/or ID-166. All urgent samples and forms should be flagged with florescent orange tape.

(2) Badge Pulls. The designated health physicist also should call the plant security post and give the following information: (a) who is calling, (b) the plant, (c) whose badge is to be pulled, and (d) the degree of urgency. The guards have the responsibility of changing the dosimetry device in the badge and delivering the exposed device to the Health Services Dosimetry Branch. The guard should flag all notes, paper work, and shipping envelopes with florescent orange tape if an urgent condition exists. The guard also has the responsibility of requesting dosimetry personnel action on off-shifts.

4.22 Special Situations. It is assumed that for some special situations analysis can be accomplished during normal working hours and not on off-shifts.

(1) Biological Samples and Whole Body Counting. When employee terminations are involved these procedures should be followed:

(a) The employee's supervisor will give the Health Physics supervisor involved the individual's name, S-number, and the date of termination.

(b) The Health Physics supervisor will determine whether or not whole body counts, biological samples, and physicals are required. He then informs the employee's supervisor and the plant nurse of his decision.

(c) If whole body counting and/or biosamples are required, the nurse will set up the appointments with the ID Health Services Medical and Analysis Branches. She then notifies the employee and his supervisor of the appointments in writing. These appointments should occur the day of termination.

At present it takes about one week to complete an analysis for strontium. If a strontium analysis is required, a urine sample should be collected and analyzed for strontium one week before termination.

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When potential exposure dictates special sampling and/or whole body counting, these procedures should be followed:

(a) The designated health physicist fills out forms ID-104 and/or ID-166, indicating for what the sample is to be analyzed and who is to be notified of the results if the information is required faster than normal channels permit. The responsible individual flags the forms with green tape and delivers them to the plant nurse. If biological samples are required, the exposed individual reports to the plant nurse and leaves an adequate sample. On off-shifts the health physicist assures that the correct sample is left for the nurse.

(b) Upon receipt of ID-104 and/or ID-166, the plant nurse will notify the ID Health Services Analytical Branch and set up appointments for whole body counts and/or biosample analysis according to the dictates of the forms received. She will arrange for the delivery of the samples and forms ID-104 and ID-166 to the Analytical Branch. She also will arrange for the whole body count and send the individual to be counted.

(2) Badge Pulls. The procedure is the same as for urgent situations with the change that the paper work and shipping envelopes are flagged with green tape.

4.23 Routine Situations. All routine sampling, whole body counting, and dosimetry device analysis should continue according to established standard practice.

4.24 General Considerations. Forms ID-104 and ID-166 are not self-explanatory. It is recommended that instructions for the use of these forms be printed on the back of each form.

The use of flagging tape can be effective if everyone involved recognizes that florescent orange tape on a sample, package, or form requires immediate attention, dispatch in handling, and analytical results as soon as possible. Green flagging tape on a sample, package, or form requires immediate attention, dispatch in handling, and analytical results as soon as possible during normal working hours, ie, 0800 to 1630 hours, Monday through Friday.

5. CPP ANTI-C SHOE EVALUATION

(C. E. Nichols)

A study was made to determine which of two safety shoes would be best for use as plant shoes at the Chemical Processing plants. One shoe had crepe soles, the other cellular grit soles.

5.1 Research

A search of the literature was undertaken in an attempt to find if any tests had been conducted on crepe soles versus cellular grit soles. This literature search produced little useful information. The manufacturer of

the safety shoes was contacted and stated that no tests comparing the two had ever been conducted, but his opinion was that the crepe soled shoe was more comfortable.

An Idaho Falls podiatrist was consulted concerning the shoes, but he would not recommend one over the other. There were advantages and disadvantages, medically speaking, for each, but he also believed that the crepe sole shoe was probably more comfortable.

A pair of shoes of each kind was obtained for testing. The individual who wore the different shoes in the test stated that the crepe sole was more comfortable than cellular grit soles.

Tests also were made to determine the ease of decontamination of the two shoe types. The shoes were contaminated in the CPP Decontamination Facility by an individual who wore a crepe sole on the right foot and grit sole shoe on the left foot. Then the mates to the shoes were worn and contaminated by walking in water. The first decontamination procedure was to clean the shoes with soap and water. The results are shown in Table II.

TABLE II

SOAP AND WATER DECONTAMINATION OF NEW SHOES

	Crepe	Cellular Grit
Original Contamination	R. 1.8 mrem/hr L. 8.8 mrem/hr	L. 6.0 mrem/hr R. 1.0 mrem/hr
After Cleaning with Soap and Water	R. background L. 0.2 mrem/hr	L. 1.8 mrem/hr R. 0.7 mrem/hr

The second contamination test was conducted, first grinding off the very rough surface of the crepe sole to simulate wear. The shoes were again contaminated as in the first test; Table III gives the results of decontamination by one cleaning with soap and water.

The contamination levels on the crepe shoes were reduced to background on a second washing but the grit shoes remained contaminated at approximately the same contamination levels. The grit shoes were then subjected to grinding and this reduced the contamination to 1.0 mrem/hr on each shoe.

Several CPP health physicists stated that they also have experienced comparable results with the decontamination of the crepe shoes. According to the technicians in the CPP Decontamination Facility, either shoe would be extremely difficult, if not impossible, to decontaminate if the upper parts of the shoe became contaminated.

TABLE III

SOAP AND WATER DECONTAMINATION OF SMOOTH SOLED SHOES

	Crepe	Cellular Grit
Original Contamination	R. 19.5 mrem/hr L. 3.0 mrem/hr	L. 2.2 mrem/hr R. 3.1 mrem/hr
After Cleaning with Soap and Water	R. 0.1 mrem/hr L. 0.5 mrem/hr	L. 1.2 mrem/hr R. 2.1 mrem/hr

5.2 Recommendations

It was recommended that the crepe shoe be adopted as the plant shoe, and that these shoes be decontaminated with soap and water when possible. The crepe shoe sole was not as porous as the grit sole, therefore, soap and water accomplished the necessary decontamination quite easily. Although the crepe shoe costs $1.55 more, it is believed that it will outlast the grit sole because of less wear to the shoe during decontamination. The present practice of decontaminating shoes is to buff off the contamination with a high speed wire brush which takes off a layer of shoe sole, in some cases almost the whole sole.

It was also recommended that the crepe soled boot be used instead of the grit soled boot when a boot is needed and approved by supervision.

6. ANTI-C SURGEON'S CAP EVALUATION

(E. A. King)

About 250 stocking knit surgeon's caps were procured from ITT Federal Support Services, Inc., Richland, Washington, for evaluation of adequacy and durability. The caps looked promising in use and cost. Each cap cost less than $0.20 as compared to $2.00 for the cap presently being used. However, upon being washed the cap shrunk too much and was rejected for that reason.

7. X-RAY TRAINING

(H. W. Stroschein, T. C. Roy)

Beginning in 1965, the Idaho State University Health Physics technicians were instructed in X-ray procedure, primarily along theoretical lines. In 1968, training in the use of X-ray was expanded to include experiments with X-ray machines provided by Oregon State University. Classes conducted by Dr. Dale E. Trout of OSU created a great interest in the field with second

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year students. The first year ISU students completed three experiments in their X-ray classes with T. C. Roy.

8. CONSTRUCTION PERSONNEL HEALTH PHYSICS INDOCTRINATION

(H. W. Stroschein, P. H. Maeser)

A standardized outline and visual aids were developed by plant construction health physicists for indoctrination lectures presented to construction personnel working in Idaho Nuclear Corporation controlled areas. Copies of the outline and slides were distributed to the CPP, TAN, and TRA construction health physicists.

9. REINDOCTRINATION LECTURES

(H. W. Stroschein, E. A. King, P. H. Maeser)

The annual Health Physics reindoctrination lecture for 1968 discussed problems associated with internal body contamination. The main emphasis was on prevention of internal contamination so that the elaborate and time consuming biological sampling and dose calculation procedures would not be necessary. One thousand four hundred and eighty-two employees and 63 visitors were reindoctrinated. Eleven TRA employees were allowed credit for attendance because of their satisfactory participation in the Health Physics section of the FONTR classes during 1968.

10. GRADUATE HEALTH PHYSICS PROGRAM

(E. A. King)

The University of Idaho, through the NRTS Education Program, is now offering a graduate Health Physics Program leading to an MS degree in radiological physics.

Prerequisite to being accepted for candidacy toward an MS in radiological science is the following: (a) BS or BA degree with an overall GPA of at least 2.4/4.0 (acceptance with a lower GPA requires special permission); (b) sufficient background undergraduate course work in calculus, modern physics, general chemistry, and biology; and (c) familiarity with the subject matter found in the new courses titled Radiological Health I and II.

The graduate curriculum will consist of courses in radiation biophysics, shielding and design engineering, radiation dosimetry instrumentation, and radiation physiology. In addition to these core courses, three other supporting courses are required. These may be chosen from a wide range of electives

based on the needs of individual students. Recommended, however, are such courses as nuclear physics, reactor physics, radiochemistry, statistics, nuclear engineering, reactor engineering, and electronics.

The degree also calls for a seminar and a research and thesis project for which a total of nine semester hours credit may be granted.

E. A. King has been appointed to the NRTS Academic Committee to assist with this new degree.

11. NRTS CERTIFICATION PROGRAM

(E. A. King, H. W. Stroschein, P. H. Maeser)

A Health Physics Certification Program in conjunction with the University of Idaho was announced in 1968. Upon successful completion of a series of courses, the individual will be awarded a Certificate of Proficiency.

The courses leading to a certificate in health physics are: biological science, engineering graphics, chemistry, English composition, college algebra, speech, technical writing, introduction to management theory, physics, radiological health. The individual must also complete certain components of the on-the-job checklist administered by the Health and Safety Services Section.

Course work from any accredited institution may be used to satisfy certification requirements if approved by university officials.

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