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RADIOLOGICAL SURVEY OF THE FORMER GAMMA INDUSTRIES SITE PORT NORRIS, NEW JERSEY

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RADIOLOGICAL SURVEY
OF THE
FORMER GAMMA INDUSTRIES SITE
PORT NORRIS, NEW JERSEY

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Ganuna Industries - September 10, 1996

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ACKNOWLEDGMENTS

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ABBREVIATIONS AND ACRONYMS

~tR/h	n&roroentgens per hour
ASME	American Society of Mechanical Engineers
BKG	background
cm	centimeter
cm 2	square centimeter
cpm	counts per minute
CRC	Cumberland Research Corporation
DOE	U. S. Department of Energy
dpm/100 CM2	disintegrations per minute per 100 square centimeters
EML	Environmental Measurements Laboratory
EPA	Environmental Protection Agency
ESSAP	Environmental Survey and Site Assessment Program
GNI	Gulf Nuclear, Inc.
kg	kilogram
m	meter
m 2	square meter
MDC	minimum detectable concentration
MeV	megaelectron volts
mci	millicuries
NaI	sodium iodide
NIST	National Institute of Standards and Technology
NRC	U.S. Nuclear Regulatory Commission
ORISE	Oak Ridge Institute for Science and Education
pCi/g	picocuries per gram
PIC	pressurized ionization chamber

RADIOLOGICAL SURVEY
OF THE
FORMER GAMMA INDUSTRIES SITE

PORT NORRIS, NEW JERSEY

INTRODUCTION AND SITE HISTORY

Gamma Industries, located in Port Norris, New Jersey, was originally o wned and operated by the Cumberland Research Corporation (CRC) for the manufacturing of cobalt-60 and ifidium-192 radiography sources. CRC w as purchased by General Nuclear in the late 1960's or early 1970's and operated under this ownership until 1972. After the change in ownersh ip, iridium-192 operations were ceased at the facility.

In 1972, Gamma Industries purchased all nuclear-related assets and bus iness activities from General Nuclear. The facility's hot cell was the n used to encapsulate Co-60 teletherapy sources. These hot cell operat ions were ceased in 1977 and most of the radioactive material was remo ved at this time. Although the hot cell equipment was left intact, the facility was unoccupied for seven years.

In June of 1984, Gamma Industries' assets were purchased by Gulf Nucle ar, Inc. (GNI), which later became a subsidiary of The GNI Group. In S eptember 1984, radioactive contamination within the facility and burie d radioactive material on the adjacent property were identified. Site excavations identified 19 drums containing a total estimated Co-60 act ivity of 62 millicurie (mCi). These drums were shipped off-site for di sposal. Other decommissioning activities, including equipment removal, hot cell decontamination, and excavation and removal of a holding tan k were completed in February 1985. A final site survey was performed b y the licensee in December 1985, the results of which were provided in a February 1989 report (GNI 1989). The submittal was accompanied by a request for termination of U.S. Nuclear Regulatory (NRC) License No. 17-16109-01 which authorized facility operations.

In April 1989, the NRC performed a closeout inspection of the facility and identified locations of residual contamination in excess of the N RC's acceptable residual contamination guidelines inside the building, particularly within the hot cell. The survey of the exterior property showed three areas with

gamma radiation in excess of background levels. A soil sample collecte d from one location had a Co-60 concentration of approximately 1600 pi cocuries per gram (pCi/g). The depth of the contamination was not dete rmined.

As a result of GNI's request for license termination and the apparent residual contamination that remains at the site, the NRC's Region I Of fice requested that the Environmental Survey and Site Assessment Progr am (ESSAP) of the Oak Ridge Institute for Science and Education (ORISE) perform an independent radiological survey of the Gamma Industries f acility. This report describes the procedures and results of the surve y.

SITE DESCRIPTION

Gamma Industries is located on a 1.3 hectare (3.1 acre) plot on Peace Avenue in Port Norris, New Jersey (Figure 1). The site is surrounded by undeveloped land to the north, south, and east, and a private residence, which fronts Peace Avenue, on the west. Access to the property is via a dirt drive which borders the south end of the property. The building is located in the approximate center of the property and is surrounded by a chain link fence, a portion of which is no longer intact. A paved area adjoins the building on the south side. Figure 2 shows the plot plan of the site. The building has approximately 160 square meters (in') of floor area and is constructed of concrete block on a concrete slab floor with a flat, wood frame roof The western half of the structure contained a shop area, two offices, a laboratory, restrooms, the hot cell and associated observation pit and the eastern half of the building consisted of the storage and loading bay (Figure 3). The structure has been abandoned for the past seven years and has significantly deteriorated as well as having been extensively vandalized. Floors were previously strewn with significant debris making most of the floor surfaces inaccessible. The debris was removed from inside the building by the licensee in June 1996 and stockpiled outside of the main doors at the southeast corner of the building. At the time of the radiological survey, the observation pit to the hot cell was full of water and debris and the exterior grounds of the facility were heavily overgrown with brush.

OBJECTIVE

The objective of the radiological survey was to provide adequate data for use by the NRC in determining the facilities radiological condition relative to the NRC guidelines for release for unrestricted use.

DOCUMENT REVIEW

ESSAP reviewed the background documentation, radiological data, and other relevant information forwarded to ESSAP by the NRC. This information was used in the development of the site-specific survey plan (ORISE 1996).

PROCEDURES

A survey team from ESSAP visited the Gamma Industries site during the period July 9 through 11, 1996 and performed visual inspections and independent measurements and sampling. Survey activities were conducted in accordance with the ORISE ESSAP Survey Procedures and Quality Assurance Manuals and the site-specific survey plan (ORISE 1995a, b, and 1996).

SURVEY PROCEDURES: D4TERIOR

The following procedures were applicable to surveys of the interior po rtions of the facility.

Reference Grid

ESSAP established a I meter x I meter grid system on the floor and low er walls (up to 2 meters) for referencing measurement and sampling loc ations. Survey data from ungridded surfaces were referenced to the flo or or lower wall grid, or to prominent building features.

Surface Scans

Surface scans for gamma and beta activity were performed over 100 perc ent of the floor surfaces and approximately 50 percent of lower walls within all portions of the facility, excluding the hot cell, where 100 percent of the surfaces within the hot cell were scanned for both gam ma and beta activity. Scans were performed using NaI scintillation and gas proportional detectors coupled to ratemeters or ratemeter-scalers with audible indicators. Particular attention was given to cracks and joints in the floor and walls, ledges, ducts, drains, and other locat ions where material may have accumulated. Locations of elevated direct radiation levels identified by scans were marked for further investig ation.

Surface Activi1y Measurements

Direct measurements for total beta activity were performed at 52 locat ions on the floor and lower walls and at eight upper wall and ceiling locations. Measurement locations were selected either randomly or base d on surface scans results which included the areas of greatest residu al activity detected during surface scans. Direct measurements were ma de using gas proportional detectors coupled to ratemeter-scalers. A sm ear for determining removable contamination was collected at each dire ct measurement location. Figures 4 through 6 show measurement and samp ling locations.

Exposure Rate Measurements

Exposure rates were measured at eight locations within the building (F igures 4 through 6). Exposure rate measurements were performed using a pressurized ionization chamber (PIC) positioned at a distance of one meter above the floor.

Subfloor Soil Sampling

A total of eight subfloor soil samples were collected from five locati ons within the facility. Sample locations were based on surface scans

results and were also placed near breaks in the concrete slab floor (F igures 4 through 6).

SURVEY PROCEDURES: EXTERIOR

The following procedures were applicable to the exterior areas of the facility.

Clearing to Provide Access

Brush was cleared from the area inside the fenceline in order to provi de adequate access for survey.

Reference System

ESSAP established a 10 in x 10 in reference grid system within the pro perty fenceline and extending to the edge of the front drive of the si te. Measurement and sampling locations from ungridded areas were refer enced to the main site grid.

Surface Scans

Gamma surface scans, using Nal scintillation detectors coupled to rate meters with audible indicators, were performed over 100 percent of the grounds within the fence boundaries. Randomly selected trails and woo ded areas outside of the fenceline were also I scanned. Areas of eleva ted radiation, suggesting the presence of surface or near surface cont amination, were marked for further investigation. The asphalt pad on t he south side of the building was scanned for gamma as well as beta ac tivity. Surface scans of the asphalt pad were performed using NaI scin tillation and gas proportional detectors coupled to ratemeters or rate meter-scalers with audible indicators.

Surface Activi1y Measurements

Direct measurements for total beta activity were performed at ten loca tions on the asphalt pad (Figure 7). Direct measurements were made usi ng gas proportional detectors coupled to ratemeterscalers. Smear sampl es were not collected from these outdoor measurement locations.

Exposure Rate Measurements

Area background exposure rate measurements were made at five locations within a 0.5 to 10 kilometer radius of the site. Exposure rate measur ements were performed at ten exterior site locations (Figure 7). Locat ions were selected either randomly or where areas of elevated radiatio n were detected by the gamma surface scans. Exposure rate measurements

were performed using a PIC positioned at I m above the surface.

Soil Sam-pline

Surface soil samples (0- 15 cm) were collected from eight locations th at included areas of elevated direct gamma radiation identified by sur face scans and from each randomly selected exposure rate measurement l ocation (excluding the measurement locations associated with the aspha lt parking area). Additional subsurface samples were collected from tw o locations where elevated direct radiation was detected as well as fr om two boreholes within a suspected former drum burial area outside of the west fenceline. Figure 7 shows sampling locations. Indications of a former drum burial area to the northeast of the site could not be l ocated and therefore boreholes were not drilled in that portion of the site, as originally planned.

SAMPLE ANALYSIS AND DATA INTERPRETATION

Samples and data were returned to ESSAP's laboratory in Oak Ridge, Ten nessee for analysis and interpretation. Sample analysis was in accorda nce with the ORISE/ESSAP Laboratory Procedures Manual (ORISE 1995c). S oil samples were analyzed by gamma spectrometry. Spectra were reviewed for the primary radionuclide of interest, Co-60, as well as any other identifiable photopeaks. Soil sample results were reported in units o f pCi/g. Smear samples were analyzed for removable gross alpha and gro ss beta activity using a low-background proportional counter. Smear re sults and direct measurement data were converted to units of disintegr ations per minute per 100 square centimeters (dpm/100 cm'). Exposure r ates were reported in units of microroentgens per hour (ptR/h)-

The data generated were then compared with the NRC's generic and site-specific guidelines used to determine whether the facility may be rele ased without radiological restrictions.

FINDINGS AND RESULTS

IWERIOR

The findings of the survey of the interior of the building are provide d below.

Surface Scans

Surface scans for gamma activity identified elevated direct radiation within the floor of the emergency shower, at two other locations withi n the main part of the building where subfloor soils had been exposed, and beneath the hot cell floor where underlying soils had also been e xposed. Beta scans identified extensive elevated direct radiation leve ls on the walls and floor of the hot cell. Surface scans of the remain

ing areas of the building did not identify any additional areas of ele vated direct beta or gamma radiation.

Surface Activity Levels

Surface activity levels for the building are summarized in Table 1. To tal beta surface activity levels ranged from less than 320 to 15,000 d pm/100 CM2 . Removable activity levels were less than 9 dpm/100 CM2 fo r gross alpha and ranged from less than 15 to 22 dpm/ 100 CM2 for gros s beta.

Exposure Rates.

Exposure rate levels are summarized in Table 2. Because there were no unaffected buildings on site in which interior background exposure rat es could be made, the exterior background exposure rates were used for data comparison. Area exterior background exposure rate levels at 1 m ranged from 7 to 8 pR/h and averaged 8 gR/h. Exposure rates at I in w ithin the building ranged from 8 to 14 VRJh. Exposure rates in the bui lding were higher due to the naturally occurTing gamma emitting

radionuclides in the concrete construction materials. With the excepti on of the hot cell, most interior exposure rates were comparable to th e exterior background levels.

Radionuclide Concentrations in Soil

Concentrations of Co-60 in subfloor soil samples are provided in Table 3. The Co-60 concentrations ranged ftom less than 0. 1 to 73.1 pCi/g.

EXTERIOR

The findings of the survey of the exterior areas are provided below.

Surface Scans

Surface scans for gamma activity identified one location of elevated d irect radiation within the property fenceline. Scans of the remaining property, surrounding trails, and the asphalt pad identified one addit ional area of slightly elevated gamma activity outside the west fencel ine.

Surface Activity Levels

Surface activity levels for the asphalt pad are summarized in Table 1. Total beta surface activity levels were less than 320 dpm/100 cm2.

Exposure Rates

Exposure rate levels are summarized in Table 2. Area exterior backgrou nd exposure rate levels at I m ranged from 7 to 8 gR/h and averaged 8 gR/h. Exterior exposure rates at I In ranged from 7 to 8 gR/h.

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Radionuclide Concentrations in Soil

Concentrations of Co-60 in exterior soil samples are provided in Table 3. The Co-60 concentrations ranged from less than 0. 1 to 46.7 pCi/g.

COMPARISON OF RESULTS WITH GUIDELINES

The applicable guidelines for the survey of Gamma Industries are those for Co-60 (beta-gamma emitting radionuclides with decay modes other t han alpha emission or spontaneous fission) which are as follows (NRC 1 987):

Total Activity
5,000 P-y dpm/100 CM2 , averaged over I M2
15,000 P-y dpm/100 CM2, maximum in 100 CM2

Removable Activity
1,000 P-y dpm/100 CM2

Surface activity levels on most of the hot cell wall surfaces were bet ween the average and maximum guideline levels, with six of the seven m easurements performed exceeding 5,000 dpm/100 CM2. Surface scans of th e contiguous areas indicated that the contamination was not localized, but rather distributed over areas in excess of I M2 , and therefore d id not satisfy the average guideline. One of the seven measurements ma de on the floor of the hot cell had a surface activity level of 9,000 dpm/100 CM2, which exceeds the I In 2 average guideline, but is less t han the maximum allowable guideline. Surface activity levels at two me asurement locations within the surrounding floor area of the hot cell were less than 320 and 1,500 d PM/ 100 CM2 and indicated that the resi dual activity on the floor was not spread over a wide area.

The site-specific Co-60 guideline is 8 pCi/g (NRC 1994). Concentration s of Co-60 in soils exceeded the 8 pCi/g guideline at four of the five subfloor sampling locations and at one exterior soil sampling

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location to a depth of at least 0.45 meters below the surface, as gamma activity levels had not
7 decreased significantly after the final sample was collected f rom this location.

Exposure rates for both the interior and exterior areas of the buildin g were within the NRC's 10 gR/h guideline (NRC 1991 and 1994).

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SUMMARY

At the request of the U. S. Nuclear Regulatory Commission, the Environ mental Survey and Site Assessment Program of the Oak Ridge Institute f or Science and Education performed radiological survey activities for the former Gamma Industries sites in Port Norris, New Jersey. Survey a ctivities included review of the available site documentation, and dur ing the period July 9 through 11, 1996 ESSAP personnel visited the sit e and performed surface scans, direct surface activity measurements, s mear sampling, soil sampling, and exposure rate measurements.

The results of the radiological survey showed that surface activity le vels within the hot cell portion of the facility are in excess of the average surface activity guidelines, but less than the maximum allowab le guidelines. Surface activity levels in the east and west rooms of t he building satisfied the residual activity guidelines. Concentrations of Co-60 exceeded the site-specific guideline in the soils beneath th e floor of the building and at one.exterior location. Exposure rates f or the facility were less than the guideline value.

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SURVEYED AREA

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FIGURE 1: Location of the Former Comma Industries Facility, Port Norri s, New Jersey

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FIGURE 2: Former Gamma Industries Facility - Plot Plan

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JEMERGENCY STORAGE

AND

SHOWER LOADING BAY

SHOP

LABORATORY
MAIN I
OFFICE I OFFICE

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FIGURE 3: Former Gamma Industries Facility Floor Plan

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MEASUREMENT/SAMPLING
LOCATIONS
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LOWER WALLS AND FLOOR
A # SINGLE-POINT
UPPER WALLS AND CEILING

SURFACE SOIL

*# EXPOSURE RATE

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FIGURE 4: Former Gamma Industries Facility, Shop, Offices, and Laboratory - Measurement and Sampling Locations

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MEASUREMENT/SAMPLING LOCATIONS

SINGLE-POINT
LOWER WALLS AND FLOOR

SINGLE-POINT
UPPER WALLS AND CEILING

E SURFACE SOIL

* ' EXPOSURE RATE

N

0 FEET 6

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FIGURE 5: Former Gamma Industriep Facility, Storage and Loading Bay -
Measurement and Sampling Locations

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OBSERVATION ROOM

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HOT CELL

MEASUREMENT/SAMPLING
LOCATIONS

SINGLE-POINT
LOWER WALLS AND FLOOR

AL # SINGLE-POINT
UPPER WALLS AND CEILING

SURFACE SOIL

EXPOSURE RATE

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FIGURE 6: Former Gamma Industries Facility, Observation Room and Hot Cell -
Measurement and Sampling Locations

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MEASUREMENT/SAMPLING
LOCATIONS

SURFACE SOIL ASPHALT
A# BOREHOLE

SINGLE-POINT X X X FENCE

EXPOSURE RATE

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FIGURE 7: Former Gamma Industries Facility, Exterior - Measurement and Sampling Locations

Gamma Industries - September 10, 1996

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TABLE 1

SUMMARY OF SURFACE ACTIVITY LEVELS
FORMER GAMMA INDUSTRIES FACILITY
PORT NORRIS, NEW JERSEY

Location'		Range of Total Activity (dpm/100 cm') Surface Single Measurement		Range of Removable Activity (dpm/100 cm')	
					F Beta
					Alpha
					Beta
East Room					
1	Floor				
2	Floor	<320	<9	<15	
3	Floor	<320	<9	<15	
4	Floor	<320	<9	<15	
5	Floor	<320	<9	<15	
6	Floor	<320	<9	<15	
7	Lower Wall		<320	<9	<15
8	Lower Wall		<320	<9	<15
9	Lower Wall		<320	<9	<15
10	Lower Wall		<320	<9	<15
11	Lower Wall		<320	<9	<15
56	Upper Wall		<320	<9	<15
57	Upper Wall		<320	<9	<15
West Room					
12	Floor	<320	<9	<15	
13	Floor	<320	<9	<15	
14	Floor	<320	<9	<15	
1	Floor	<320	<9	<15	
16	Floor	<320	<9	<15	

TABLE 1 (Continued)

SUMMARY OF SURFACE ACTIVITY LEVELS
FORMER GAMMA INDUSTRIES FACILITY
PORT NORRIS, NEW JERSEY

Location'		Range of Total Activity (dpm/100 cm') Surface Single Measurement		Range of Removable Activity (dpm/100 cm')
Beta	Alpha	Beta		
West Room (Continued)				
17	Floor	<320	<9	<15
18	Floor	<320	<9	<15
19	Floor	<320	<9	<15
20	Floor	<320	<9	<15
21	Floor	<320	<9	<15
22	Floor	<320	<9	<15
23	Floor	<320	<9	<15
24	Floor	<320	<9	<15
25	Floor	660	<9	<15
26	Floor	<320	<9	<15
27	Lower Wall	<320	<9	<15
28	Lower Wall	<320	<9	<15
29	Lower Wall	<320	<9	<15
30	Lower Wall	<320	<9	<15
31	Lower Wall	<320	<9	<15
32	Lower Wall	<320	<9	<15
33	Lower Wall.	<320	<9	<15
34	Lower Wall	<320	<9	<15

TABLE 1 (Continued)

SUMMARY OF SURFACE ACTIVITY LEVELS
FORMER GAMMA INDUSTRIES FACILITY
PORT NORRIS, NEW JERSEY

Location'		Range of Total Activity (dpm/100 cm') Surface Single Measurement		Range of Removable Activity (dpm/100 cm')
Beta	Alpha	Beta		
West Room (Continued)				
-58	Upper Wall	<320	<9	<15
59	Upper Wall	<320	<9	<15
60	Upper Wall	<320	<9	<15
Observation Pit				
35	Floor	<320	<9	<15
36	Floor	<320	<9	<15
37	Lower Wall	<320	<9	<15
38	Lower Wall	<320	<9	<15
39	Lower Wall	<320	<9	<15

40	Ceiling	<320	<9	<15	
Hot Cell					
41	Floor	9,000	<9	<15	
42	Floor	<320	<9	<15	
43	Floor	1,500	<9	<15	
44	Floor	<320	<9	<15	
45	Floor	<320	<9	<15	
46	Floor	2,600	<9	<15	
47	Floor	390	<9	<15	
	Lower Wall		4,300	<9	<15
49	Lower Wall		7,100	<9	<15

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TABLE 1 (Continued)

SUMMARY OF SURFACE ACTIVITY LEVELS
FORMER GAMMA INDUSTRIES FACILITY
PORT NORRIS, NEW JERSEY

		Range of Total Activity (dpm/100 cm')	Range of Removable Activity (dpmI100 cm2)	
Location"		Surface Single Measurement		
Beta	Alpha	Beta		
Hot Cell (Continued)				
50	Lower Wall	9,200	<9	22
52	Lower Wall	15,000	<9	17
53	Lower Wall	12,000	<9	<15
54	Lower Wall	8,400	<9	<15
51	Upper Wall	5,000	<9	<15
55	Ceiling <320	<9	<15	
Exterior Loading Dock				
61	Floor <320	--- b	--	
62	Floor <320	---	--	
63	Concrete	<320	---	--
64	Floor <320	---	--	
65	Floor <320	---	--	
66	Floor <320	---	--	
67	Floor <.320	---	--	
68	Floor <320	---	--	
	Floor <320	---	--	
70	Floor <320			

'Refer to Figures 4 through 7. '---Sample not collected.

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TABLE2

EXPOSURE RATES

FORMER GAMMA MUSTRIES FACILITY
PORT NORRIS, NEW JERSEY

Location	-F-Exposure Rates at 1 m
Interior'	
East Room, Location 1	9
East Room, Location 2	8
West Room, Location 1	9
West Room, Location 2	9
West Room, Location 3	8
Observation Room	9
Hot Cell, Location 1	10
Hot Cell, Location 2	14
Exterior'	
Location 1	8
Location 2	8
Location 3	8
Location 4	8
Location 5	8
Location 6	8
Location 7	7
Location 8	8
Location 9	7
Location 10	8

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TABLE 2 (Continued)

EXPOSURE RATES
FORMER GAM114A INDUSTRIES FACILITY PORT NORRIS, NEW JERSEY .

Location	Exposure Rates at 1 m (ILR/h)
Background	
North Avenue	8
Warren Street at f1igh Street	8
South High at Miller Avenue	7
Hig ay 553 at Lincoln Street	7
Hghway 553 at Highway 555	8

'Refer to Figures 4 through 6. 'Refer to Figure 7.

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TABLE3

RADIONUCLIDE CONCENTRATIONS IN SOIL SAMPLES
FORMER GAMMA INDUSTRIES FACILITY
PORT NORRIS, NEW JERSEY

Sample ID	Location*	Depth (cm)	Radionuclide

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			Concentration (pCi/g) Co-60
Interior			
1	West Room, 14.7N, 4E	0-15	<0. 1
2	West Room, 14.6N, 0.5E	0-15	25.5 ± 0.6 b
3	West Room, 5N, 4E	0-15	9.2 ± 0.9
4	West Room, 5N, 4E	15-30	9.0 0.4
5	East Room, 9.2N, 3.5E	0-15	20.8 0.6
6	East Room, 9.2N, 3.5E	15-30	9.4 0.4
7	Hot Cell, North Wall/Floor Interface	0-15	73.1 1.3
8	Hot Cell, North Wall/Floor Interface	15-30	66.6 1.2
Exterior			
9	12.5N, 125E	0-15	2.1:1:0.2
10	12.5N, 125E	15-30	0.6 ± 0.1
11	ION, 23W	0-15	<0. 1
12	ION, 23W	15-30	<0. 1
13	14N, 24W	100-115	<0. 1
14	27N, 24E	0-15	0.3+0.1
15	27N, 24E	15-30	12.2 ± 0.5
16	27N, 24E	30-45	46.7+1.0
17	15N, 25E	0-15	<0. 1
18	21N, 6E 0-15	0.5 ± 0.1	
19	35N, 5E 0-15	<0. 1	
20	25N, 5W 0-15	<0. I	

aLocation referenced to southwest grid coordinate. Refer to Figures 4 through 7. bUncertainties represent the 95% confidence level, based on ly on counting statistics.

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APPENDIX A

MAJOR INSTRUMENTATION

APPENDIX A

MAJOR INSTRUMENTATION

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The display of a specific product is not to be construed as an endorse ment of the product or its manufacturer by the author or his employers .

DIRECT RADIATION MEASUREMENT

Instruments

Eberline Pulse Ratemeter
Model PRM-6
(Eberline, Santa Fe, NM)

Ludlum Floor Monitor
Model 239-1
(Ludlurn Measurements, Inc.,
Sweetwater, TX)

Ludlurn Ratemeter-Scaler

Model 2221
(Ludlurn Measurements, Inc.,
Sweetwater, TX)

Detectors

Ludlum Gas Proportional Detector
Model 43-37
Effective Area, 550 CM2
(Ludlurn Measurements, Inc.,
Sweetwater, TX)

Ludlum Gas Proportional Detector
Model 43-68
Effective Area, 126 CM2
(Ludlum Measurements, Inc.,
Sweetwater, TX)

Reuter-Stokes Pressurized Ionization Chamber
Model RSS- I I I
(Reuter-Stokes, Cleveland, OH)

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Victoreen Nal Scintillation Detector
Model 489-55
3.2 cm x 3.8 cm Crystal
(Victoreen, Cleveland, OH)

LABORATORY ANALYTICAL INSTRUMENTATION

High Purity Extended Range Intrinsic Detectors
Model No: ERVDS30-25195
(Tennelec, Oak Ridge, TN)
Used in conjunction with:
Lead Shield Model G- I I
(Nuclear Lead, Oak Ridge, TN) and

Multichannel Analyzer
3 100 Vax Workstation
(Canberra, Meriden, CT)

High-Purity Germanium Detector
Model GMX-23195-S, 23% Eff.
(EG&G ORTEC, Oak Ridge, TN)
Used in conjunction with:
Lead Shield Model G-16
(Gamma Products, Palos Hills, IL) and

Multichannel Analyzer
3 100 Vax Workstation

(Canberra, Meriden, CT)

Low Background Gas Proportional Counter
Model LB-5 I 00-W
(Oxford, Oak Ridge, TN)

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APPENDIX B

SURVEY AND ANALYTICAL PROCEDURES

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APPENDIX B

SURVEY AND ANALYTICAL PROCEDURES

SURVEY PROCEDURES

Surface Scans

Surface scans were performed by passing the detectors slowly over the surface; the distance between the detectors and the surface was mainta ined at a minimum - nominally about I cm. A large surface area, gas pr oportional floor monitor was used to scan the floors of the surveyed a reas. Other surfaces were scanned using small area (126 CM2) hand-hel d detectors. Identification of elevated levels was based on increases in the audible signal from the recording and/or indicating instrument. Combinations of detectors and instruments used for the scans were:

Beta gas proportional detector with ratemeter-scaler

Gamma Nal scintillation detector with ratemeter

Surface Activi1y Measurements

Measurements of total beta activity levels were performed using gas pr oportional detectors with portable ratemeter-scalers. Count rates (cpm), which were integrated over I minute in a static position, were conv erted to activity levels (dpm/100 cm) by dividing the net rate by the 4 7r efficiency and correcting for the active area of the detector. Di fferent building materials (poured concrete, brick, wood, steel, etc.) may have different background levels; average background count rates are typically determined for each material encountered in a surveyed a rea at a location of similar construction and having no known radiolog ical history. However, because there was only a single building locate d on the site-all of which was considered affected by radioactive mate rial use-a single representative background count rate was used for da ta conversions. The beta activity background count rates for the gas p

roportional detectors ranged from 363 to 383 cpm. Beta

efficiency factors ranged fi-om 0.23 to 0.25 for the gas proportional detectors and calibrated to Tc-99. The beta minimum detectable concent ration (MDC) was 320 dpm/100 CM2 The effective window area for the gas proportional detectors was 126 cm'.

Removable ActiviV Measurements

Removable gross activity levels were determined using numbered filter paper disks, 47 mm in diameter. Moderate pressure was applied to the s mear and approximately 100 CM2 of the surface was wiped. Smears were p laced in labeled envelopes with the location and other pertinent infor mation recorded.

Exposure Rate Measurements

Measurements of gamma exposure rates were performed using a pressurize d ionization chamber (PIC). The instrument was adjusted to one meter a bove the surface and allowed to stabilize. The measurement was read di rectly in (gR/h).

Soil Sampling

Approximately I kg of soil was collected at each sample location. Coll ected samples were placed in a plastic bag, sealed, and labeled in acc ordance with ESSAP survey procedures.

ANALYTICAL PROCEDURES

Removable Activi1y

Smears were counted on a low-background gas proportional system for gr oss alpha and gross beta activity.

Gamma SpectroscoM

Soil samples were dried, mixed, crushed, and/or homogenized as necessa ry, and a portion sealed in a 0.5-liter Marinelli beaker or other appr opriate container. The quantity placed in the beaker was chosen to rep roduce the calibrated counting geometry. Net material weights were det ermined and the samples counted using intrinsic germanium detectors co upled to a pulse height analyzer system. Background and Compton stripp ing, peak search, peak identification, and concentration calculations were performed using the computer capabilities inherent in the analyze r system. All photopeaks associated with the radionuclides of concern were reviewed for consistency of activity. The energy peak used for de

termining the activity of the radionuclide of concerns was: Co-60 at 1 . 173 MeV.

UNCERTAINTIES AND DETECTIoN LIMITS

The uncertainties associated with the analytical data presented in the tables of this report represent the 95% confidence level for that dat a. These uncertainties were calculated based on both the gross sample count levels and the associated background count levels. Additional un certainties, associated with sampling and measurement procedures, have not been propagated into the data presented in this report.

Detection limits, referred to as minimum detectable concentration (MDC), were based on 2.71 plus 4.65 times the standard deviation of the ba ckground count [2.71 + (4.651BKG)]. When the activity was determined t o be less than the MDC of the measurement procedure, the result was re ported as less than MDC. Because of variations in background levels, m easurement efficiencies, and contributions from other radionuclides in samples, the detection limits differ from sample to sample and instru ment to instrument.

CALIBRATION AND QUALITY ASSURANCE

Calibration of all field and laboratory instrumentation was based on s tandards/sources, traceable to NIST, when such standards/sources were available. In cases where they were not available, standards of an ind ustry recognized organization were used. Calibration of pressurized io nization chambers was performed by the manufacturer.

Analytical and field survey activities were conducted in accordance wi th procedures from the following documents of the Environmental Survey and Site Assessment Program:

0 Survey Procedures Manual, Revision 9 (April 1995)
0 Laboratory Procedures Manual, Revision 9 (January 1995)
0 Quality Assurance Manual, Revision 7 (January 1995)

The procedures contained in these manuals were developed to meet the r equirements of DOE Order 5700.6C and ASME NQA-1 for Quality Assurance and contain measures to assess processes during their performance.

Quality control procedures include:

Daily instrument background and check-source measurements to confirm t hat equipment operation is within acceptable statistical fluctuations. Participation in EPA and EML laboratory Quality Assurance Programs. T raining and certification of all individuals performing procedures. Pe riodic internal and external audits.

APPENDIX C

GUIDELINES FOR DECONTAMINATION OF FACILITIES AND EQUIPMENT PRIOR TO RELEASE FOR UNRESTRICTED USE OR TERMINATION OF LICENSES FOR BYPRODUCT, SOURCE OR SPECIAL NUCLEAR MATERIAL

GUIDELINES FOR DECONTAMINATION OF FACILITIES AND EQUIPMENT PRIOR TO RELEASE FOR UNRESTRICTED USE OR TERMINATION OF LICENSES FOR BYPRODUCT, SOURCE, OR SPECIAL NUCLEAR MATERIAL

U.S. Nuclear Regulatory Commission
Division of Fuel Cycle & Material Safety
Washington, D.C. 20555

August 1987

Gamma Industries - September 10, 1996

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The instructions in this guide, in conjunction with Table 1, specify t he radionuclides and radiation exposure rate limits which should be us ed in decontamination and survey of surfaces or premises and equipment prior to abandonment or release for unrestricted use. The limits in T able I do not apply to premises, equipment, or scrap containing induce d radioactivity for which the radiological considerations pertinent to their use may be different. The release of such facilities or items f rom regulatory control is considered on a case-by-case basis.

I . The licensee shall make a reasonable effort to eliminate residual contamination.

2. Radioactivity on equipment or surfaces shall not be covered by paint, plating, or other covering material unless contamination level s, as determined by a survey and documented, are below the limits spec ified in Table I prior to the application of the covering. A reasonabl e effort must be made to minimize the contamination prior to use of an y covering.

3. The radioactivity on the interior surfaces of pipes, drain lin es, or ductwork shall be determined by making measurements at all trap s, and other appropriate access points, provided that contamination at these locations is likely to be representative of contamination on th e interior of the pipes, drain lines, or ductwork. Surfaces or premise

s, equipment, or scrap which are likely to be contaminated, but are of such size, construction, or location as to make the surface inaccessible for purposes of measurement, shall be presumed to be contaminated in excess of the limits.

4. Upon request, the Commission may authorize a licensee to relinquish possession or control of premises, equipment, or scrap having surfaces contaminated with materials in excess of the limits specified. This may include, but would not be limited to special circumstances such as razing of buildings, transfer from premises to another organization continuing work with radioactive materials, or conversion of facilities to a long-term storage or standby status. Such requests must:

a. Provide detailed, specific information describing the premises, equipment or scrap, radioactive contaminants, and the nature, extent, and degree of residual surface contamination.

b. Provide a detailed health and safety analysis which reflects that the residual amounts of materials on surface areas, together with other considerations. such as prospective use of the premises, equipment, or scrap, are unlikely to result in an unreasonable risk to the health and safety of the public.

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5. Prior to release of premises for unrestricted use, the licensee shall make a comprehensive radiation survey which establishes that contamination is within the limits specified in Table 1. A copy of the survey report shall be filed with the Division of Fuel Cycle, Medical, Academic, and Commercial Use Safety, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and also the Administrator of the NRC Regional Office having jurisdiction. The report should be filed at least 30 days prior to the planned date of abandonment. The survey report shall:

a. Identify the premises.

b. Show that reasonable effort has been made to eliminate residual contamination.

Describe the scope of the survey and general procedures followed.

d. State the findings of the survey in units specified in the instruction.

Following review of the report, the NRC will consider visiting the facilities to confirm the survey.

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TABLE I
ACCEPTABLE SURFACE CONTAMINATION LEVELS

Nuclides b,f	Average b,f	Maximum b,',f	Removable
U-nat, U-235, U-238, and rA associated decay products	5,000 dpm a/100 cm'	15,000 dpm a/100 CM2	1,000 dpm a/100 CM2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227,1-125,1-129	100 dprn/ 100 ci-n'	300 dpin/100 CM2	20 dpm/ 100 cm'
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232,1-126,1-131,1-133	1,000 dpm/100 cm'	3,000 dpm/ 100 CM2	200 dpm/100 CM2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5,000 dpm Py/100 CM2	15,000 dpm py/100 CM2	1,000 dpm py/100 CM2

41 Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gammaemitting nuclides should apply independently. b As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation. Measurements of average contaminant should not be averaged over more than I square meter. For objects of less surface area, the average should be derived for each such object. d The maximum contamination level applies to an area of not more than 100 cm'. The amount of removable radioactive material per 100 cm' of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped. f The average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/h at I cm and 1.0 mrad/h at I cm, respectively, measured through not more than 7 milligrams per square centimeter of total absorber.