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PDR

UNION CARBIDE CORPORATION
MEDICAL PRODUCTS DIVISION
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October 21, 1982

U. S. Nuclear Regulatory Commission
Attn: Leland C. Rouse
Division of Fuel Cycle and Material Safety
Washington, DC 20555

Dear Sirs:

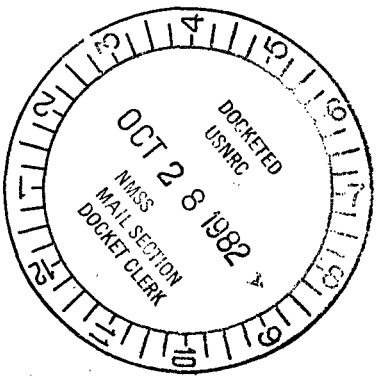
Attached please find six (6) copies of our response to your December 9, 1981 questions related to the enviromental review of our renewal application for License No. SNM-639.

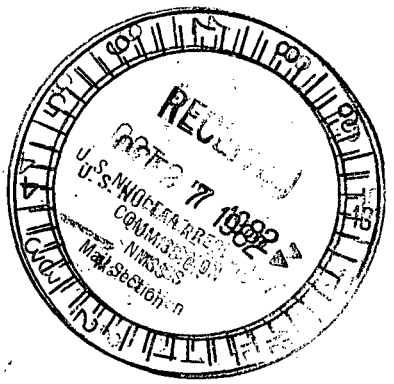
Yours very truly,

Marcus H. Voth

Marcus H. Voth
Manager
Nuclear Operations

MHV:js
Attachment





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ADDITIONAL ENVIRONMENTAL ASSESSMENT

INFORMATION REQUESTED 12/09/81 BY

U. S. NUCLEAR REGULATORY COMMISSION REGARDING

LICENSE NO. SNM-639 DOCKET NO. 70-687

SUBMITTED 10/21/82 BY

UNION CARBIDE CORPORATION

1.0 SITE EFFLUENTS

Effluents from the operations conducted at the Sterling Forest Facility originate in one of the five principle buildings on the site;

Building 1 - Reactor,
Building 2 - Hot Laboratory,
Building 3 - Maintenance,
Building 4 - Administration and
Building 5 - Heating Plant.

Figure 1.1 shows the relative location of these buildings. While activities licensed under the SNM-639 license are restricted to Buildings 1 and 2, the radiological and non-radiological effluents from each of the the five buildings will be discussed in this section.

1.1 EFFLUENT SOURCES, TREATMENT AND MONITORING

1.1.1 Gaseous Effluents

Building 1 is maintained at a negative pressure whenever the reactor is operating so as to direct any air leakage through the containment boundary into the building. The ventilation system is shown in Figure 1.2. The potential gaseous effluent of primary concern is radioactivity leaving the building via the main exhaust duct which leads to an elevated stack shared with Building 2. In the event of a release of radioactivity from the core, the ventilation system will automatically isolate by tripping the main supply and exhuast fans and dampers and initiating the emergency exhaust fans which exhaust through absolute and charcoal filters. Such an isolation will occur when initiated manually, when the facility experiences a loss of commercial power and upon high radiation from the bridge monitrons sensing a gaseous release from the core. Double doors at each access point to the building provide an airlock.

The ventilation system for the Hot Laboratory is also shown in Figure 1.2. Again, the potential gaseous effluent of primary concern is radioactivity. The general flow pattern is to supply air to the clean areas and exhaust air from areas having the potential to be contaminated. All flow from the Hot Lab passes through roughing and HEPA filters. Ventilation from the hot cells and the waste drum storage area passes through additional filtration before joining the larger flow of exhaust air. The hot

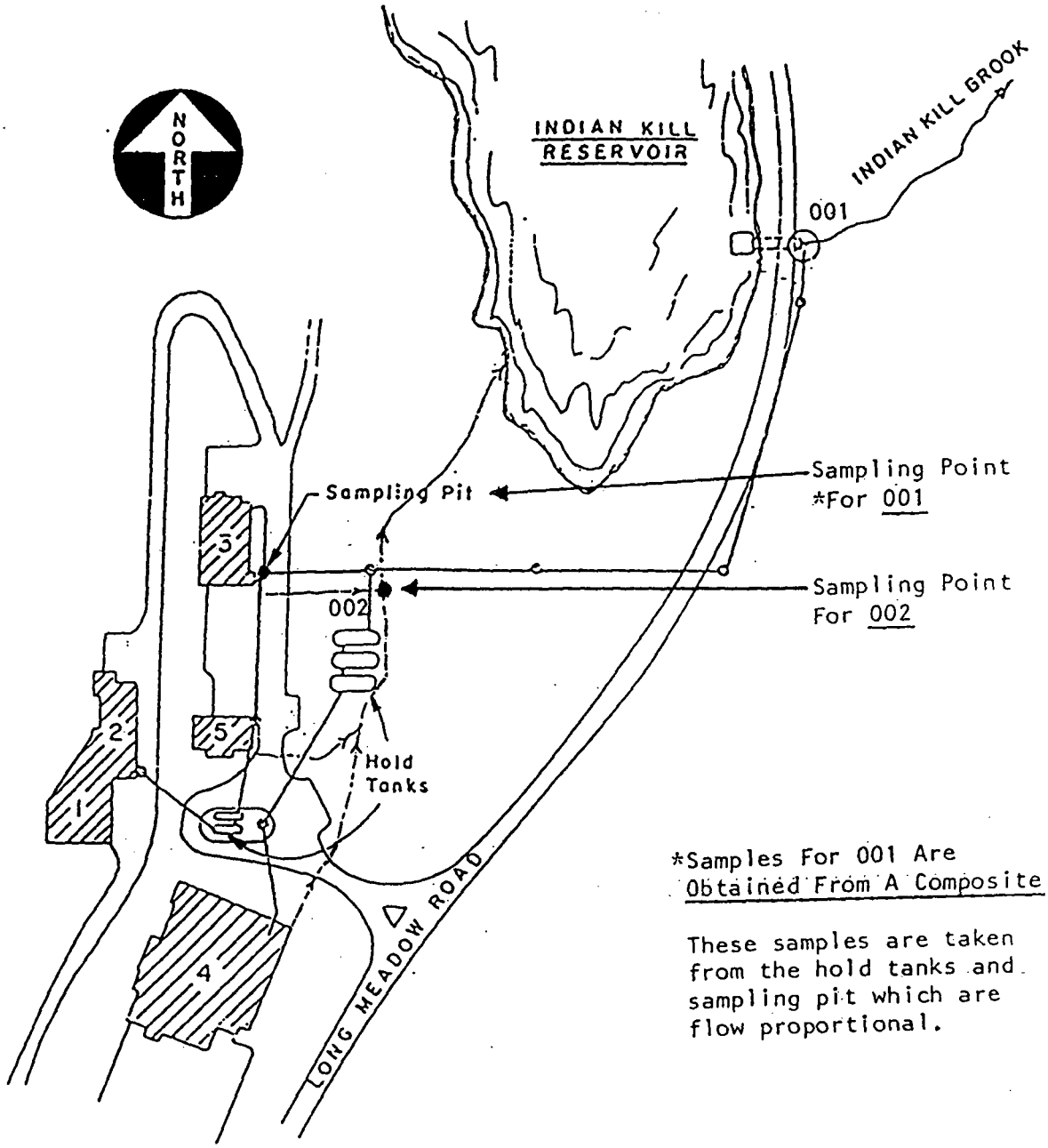
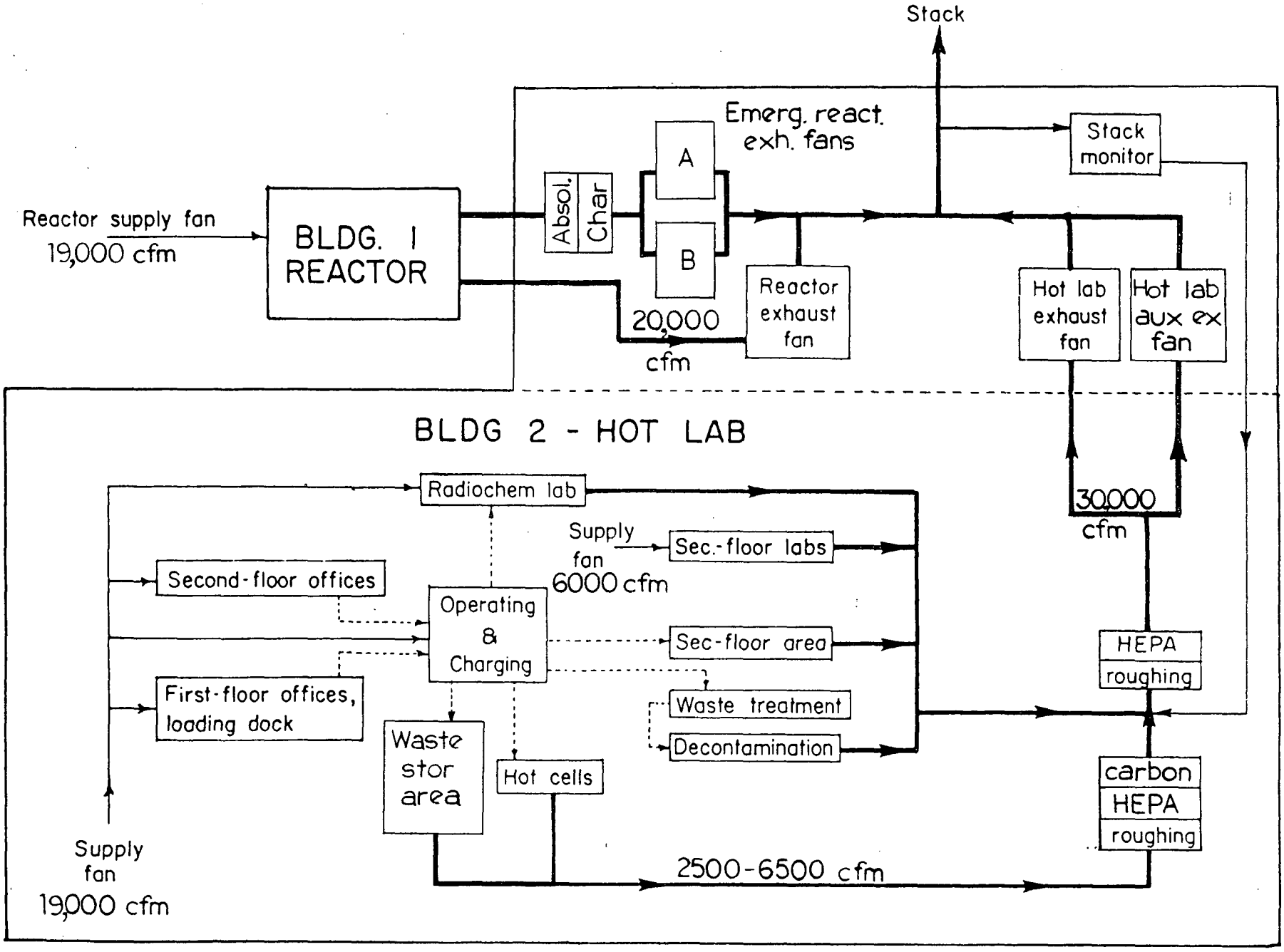


FIGURE 1.1

SITE LAYOUT



SCHEMATIC of REACTOR, HOT LAB AIR FLOW

FIGURE 1.2

cell and waste storage filter system provides roughing and HEPA filters followed by two charcoal filters in series, each charcoal bed being two inches; flow through the filters is adjusted to provide maximum residence time at an adequate negative pressure on the hot cells. An auxiliary exhaust fan is provided in parallel with the main exhaust fan to assure adequate ventilation at all times.

The discharge from the Hot Laboratory exhaust fan joins the reactor building exhaust and passes to the common elevated stack. The stack monitor for the two buildings is located in Building 2 with local readouts and alarms for the iodine, noble gas and particulate monitors; remote recorders and alarms for each of these three parameters are located in the reactor control room. No other gaseous effluents of significance originate in Buildings 1 and 2.

Buildings 1 and 2 share a commercial power feed transformer and emergency generators. Upon loss of commercial power the emergency generators will start and pick-up load automatically. The main exhaust fans will automatically transfer to half speed on loss of commercial power. If either fan fails to transfer the backup fan (emergency reactor exhaust fan or the auxiliary hot lab exhaust fan) will start automatically.

Building 3 presently houses the site maintenance operations and provides storage space. There are no gaseous effluents of significance from this building.

Building 4 has two areas in which radioactive by-product material is handled. These areas are identified as the generator production area shown in Figure 1.3 and the G-row area shown in Figure 1.4. Refined isotopes from Building 2, primarily Mo-99 and Xe-133, are delivered to this area where by they are converted into finished radiopharmaceutical products. The filtered exhaust from each of these areas is monitored for gross beta and gamma activity before passing to an exhaust stack on the roof of the building. An emergency generator will automatically assume critical electrical loads on loss of commercial power. No other gaseous effluents of significance originate in Building 4.

Building 5 is a conventional heating plant which is normally fueled from natural gas with provisions for using heating fuel oil in the event of a gas shortage. A small lead pouring operation is also housed in Building 5 to make shielded containers. Air is ducted from that area as part of the normal ventilation for the building. Periodic grab samples are taken as a means of monitoring effluents.

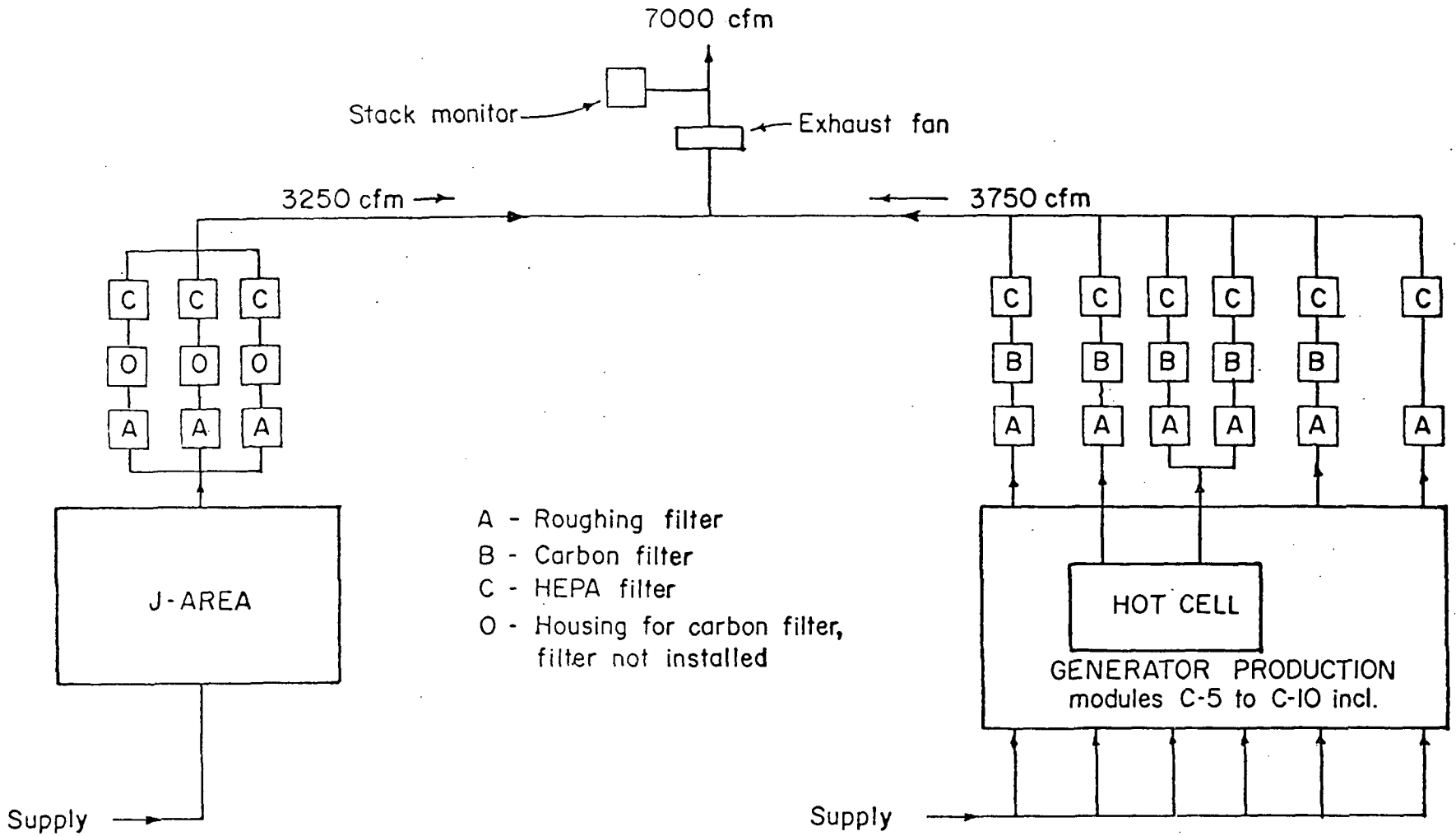


FIGURE 1.3

SCHEMATIC OF GENERATOR-PRODUCTION VENTILATION

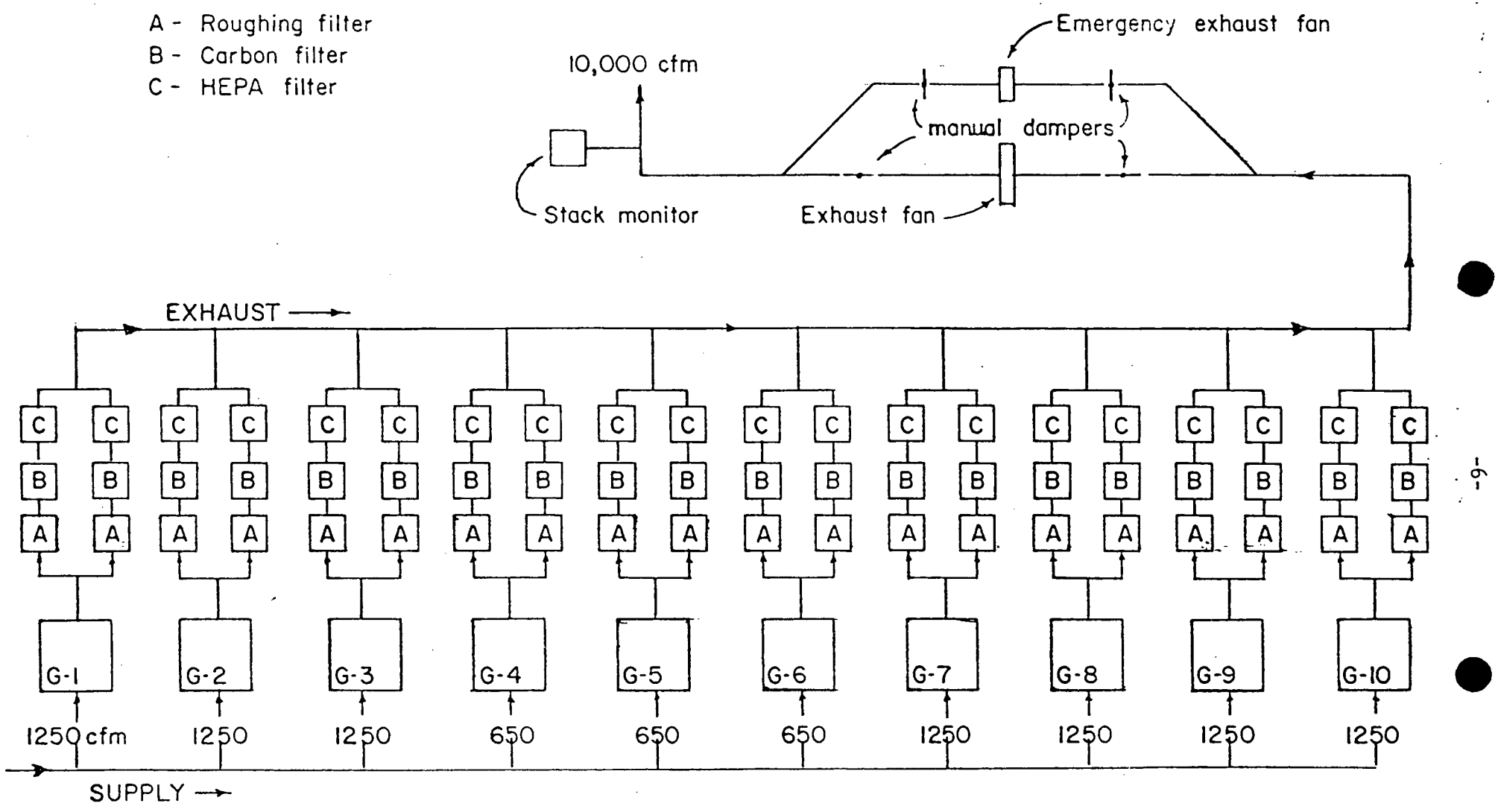


FIGURE 1.4

SCHEMATIC OF G-ROW VENTILATION

1.1.2 Liquid Effluents

Figure 1.5 gives an overview of the site liquid waste disposal system. There are three sub-systems identified as follows: radioactive waste, non-radioactive process waste, and sanitary waste.

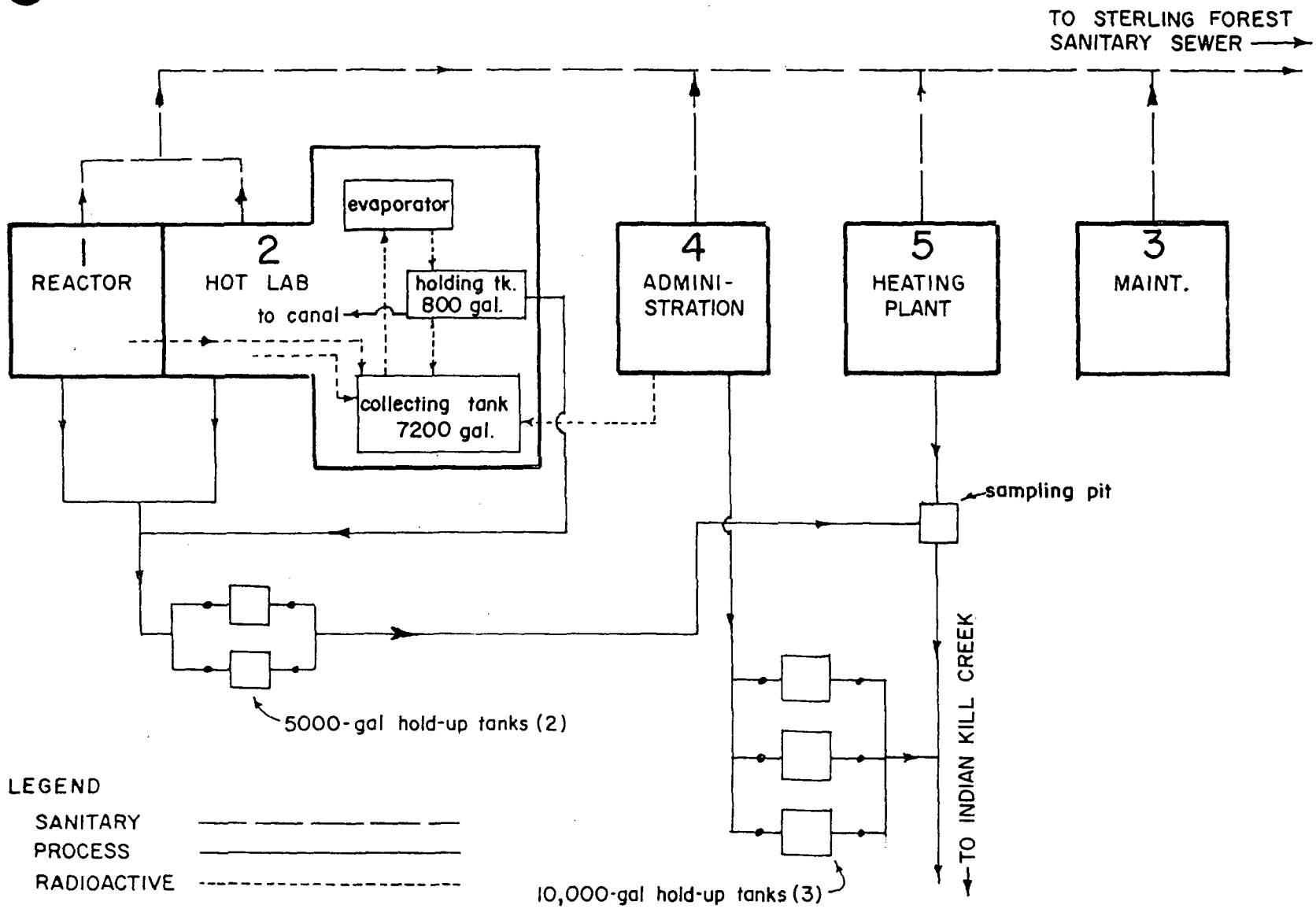
All radioactive liquid waste from the site is directed to a collector tank in Building 2 which feeds an evaporator for separating the liquid waste into radioactive sludge and decontaminated water. This system is shown in Figure 1.6. The potential sources for liquid radioactive waste are in Buildings 1, 2 and 4. Upon purifying the contaminated water the hold tank is sampled to assure that water has been adequately decontaminated. If the water is pure it is either returned to the canal which is part of the primary water system for the reactor or it is discharged to one of the 5,000 gallon mall tanks. If the water has not been decontaminated it will be returned to the collector tank and reprocessed. When a mall tank is full it is sampled and released as a batch. Non-radioactive process waste from Buildings 1 and 2 are also fed into the mall tanks for sampling and batch releasing.

The process liquid waste system is made up of drains directly from Buildings 4 and 5 and indirectly by way of the mall tanks from Buildings 1 and 2. Liquid process wastes from Building 5 and the mall tanks flow through a sample pit where periodic grab samples are taken to analyze the continuous waste stream. Process waste from Building 4 can be batch-released from hold tanks after sampling.

The sanitary waste system is the collection of sanitary plumbing and potable water drains from the five buildings on site. This waste stream flows directly to the Sterling Forest Sanitary Sewage Processing Plant.

Small volumes of liquid wastes identified as hazardous substances are collected and disposed by proper techniques. These are not handled routinely and are therefore treated on a case-by-case basis by qualified technicians on a small scale in a laboratory environment.

Storm water flows from roofs and paved areas by way of the natural terrain of the land, ulitmately flowing past a sampling point enroute to the Indian Kill reservoir as is shown in Figure 1.1.



SCHEMATIC OF LIQUID EFFLUENT LINES

FIGURE 1.5

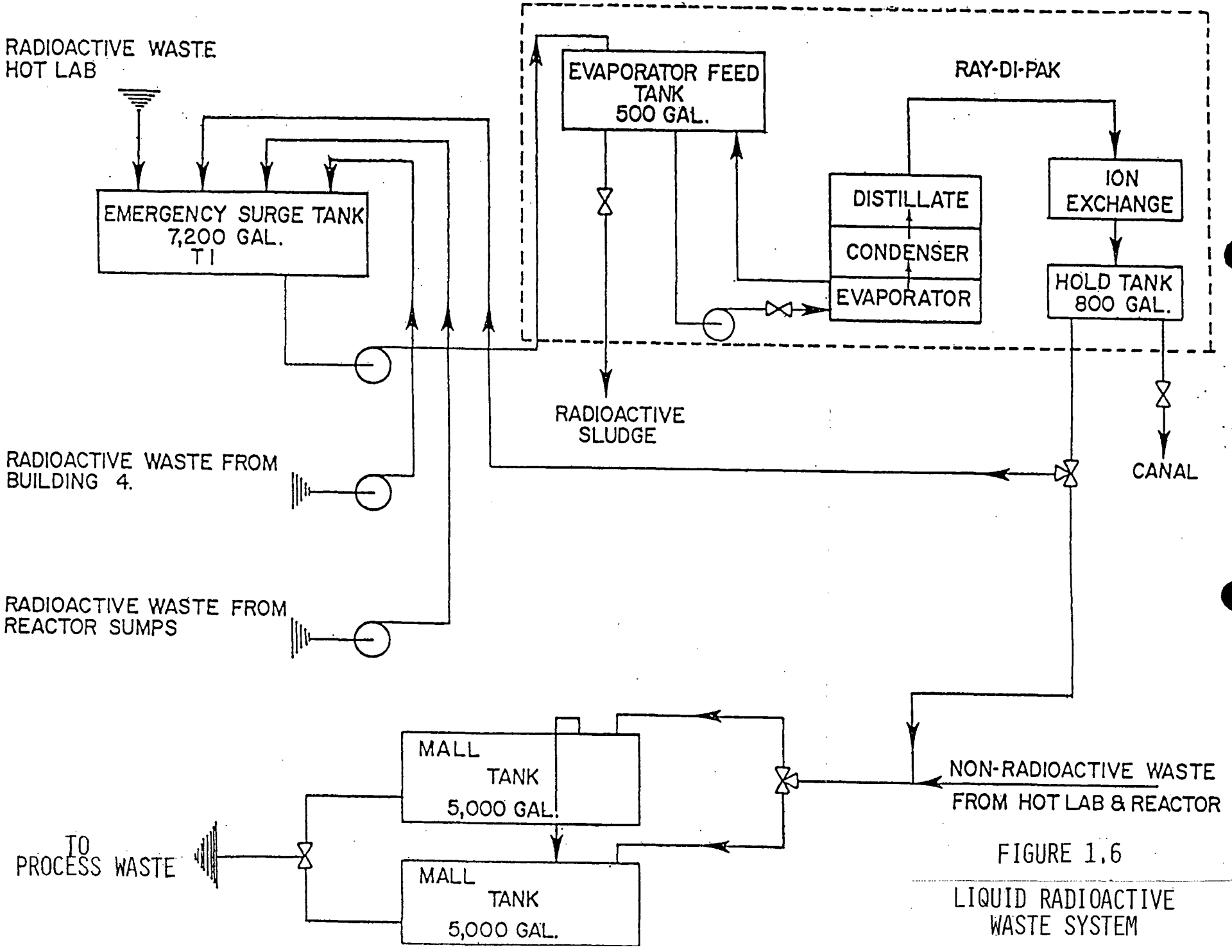


FIGURE 1.6

LIQUID RADIOACTIVE
WASTE SYSTEM

1.1.3 Solid Waste

The non-radioactive solid waste generated on the site is characteristic of the incidental waste generated by any laboratory facility. It is primarily composed of paper, plastics, wood or metal in the form of packaging material, worn out laboratory equipment, construction material and office supplies.

All solid radioactive waste originating at the site meets the definition of low level waste with the exception of spent reactor fuel. Approximately 24 reactor fuel elements are shipped to U. S. Department of Energy reprocessing facilities each year. The low level radioactive waste is further divided into low specific activity and high specific activity material. The low specific activity is made up of resins from the reactor water cleanup system which have been solidified, liquid wastes from the target preparation process which have been solidified, and compacted laboratory trash. While the volume of low specific activity waste is sizeable, the amount of radioactivity is very small. The high specific activity wastes, in contrast, are low in volume but high in radioactivity. These wastes are packaged in 55 gallon drums inside of the hot cells and require substantial shielding when removed from the cells. After removal from the hot cells where storage space is limited these waste drums are put into specially designed, ventilated storage pits where they are allowed to decay for an additional 6 to 8 months before disposal. All low level radioactive waste is shipped to licensed commercial burial facilities, the majority going to Barnwell, South Carolina.

1.2 COMPLIANCE WITH EFFLUENT LIMITS

1.2.1 Gaseous Effluents

Non-radiological effluents from the site consist of the normal ventilation air from buildings, the effluent flue gas from the site heating plant and vapors from laboratory hoods which are incidential to operations and tests. Because of the incidential nature of non-radiological releases there are no effluent limits placed on these sources.

In 1979 Technical Specifications for the reactor were issued under NRC License R-81 defining radiological release limits for the combined Reactor and Hot Laboratory stack. Limits are quoted in terms of the annual contribution to exposure in unrestricted areas as a result of gaseous effluents from the site. The limits for

whole body dose from noble gases is 5 mRem per year or 2.5 mRem per quarter. The limit for thyroid exposure due to iodine is 15 mRem per year or 7.5 mRem per quarter. Data has been carefully collected, analyzed and reported in that format for the two full calendar years since the Tech Specs were issued. The results, which typify a larger interval of time, are as follows:

Year	Annual Whole Body Exposure (mRem)	Percent of Limit	Annual Thyroid Exposure (mRem)	Percent of Limit
1980	0.46	9.2	1.19	7.9
1981	0.28	5.6	1.04	6.9

1.2.2 Liquid Effluents

Non-radiological liquid effluents from the site are regulated by the New York State Department of Environmental Conservation (DEC) State Pollution Discharge Elimination System (SPDES) discharge permit number NY-004464. The monitoring requirements and limiting concentrations for the process waste effluents (outfall 001) and the storm sewers (outfall 002) are given in Table 1.1 (The effective SPDES date was August 1, 1978. Since that time the discharge permit limits have never been exceeded, many of the parameters being below the mimimum level of detection.)

The limit on radiological liquid effluents from the site is found in the Technical Specifications issued under the R-81 license. That limit is 0.01 Curie Sr-90 equivalent per year when released from monitored batch tanks. The following liquid effluent data represents operating experience for the past years.

Year	Effluent (Curies)	Percent of Limit
1978	.0005	5
1979	.0008	8
1980	.0005	5
1981	.0007	7

1.2.3 Solid Waste

There is no limitation placed on the facility for the generation of solid waste. However, the facility is under the volume allocation system imposed by the State of South Carolina and administered by the Chem-Nuclear burial site at Barnwell which receives essentially all solid waste generated at the Sterling Forest facility. In addition, the numerous NRC, DOT and State requirements for the packaging and transporation of solid waste, as well as the operating permit issued by the South Carolina Department of Health and Environmental Conservation to the operators of the burial site, are adhered to closely.

TABLE 1.1

SPDES PERMIT NY-004464 CONDITIONS

Outfall Number	Effluent Parameter	Daily Max.	Measurement Frequency	Sample Type
001	Flow	N/A	MONTHLY	INSTANTANEOUS
	pH	6.5 - 8.5	MONTHLY	GRAB
	Aluminum	1.0 mg/L	MONTHLY	24 Hr. COMPOSITE
	Cadmium	0.1 mg/L	MONTHLY	24 Hr. COMPOSITE
	Chromium Total	1.0 mg/L	MONTHLY	24 Hr. COMPOSITE
	Chromium Hex.	0.05 mg/L	MONTHLY	24 Hr. COMPOSITE
	Copper	0.2 mg/L	MONTHLY	24 Hr. COMPOSITE
	Fluoride	1.5 mg/L	MONTHLY	24 Hr. COMPOSITE
	Iron	2.0 mg/L	MONTHLY	24 Hr. COMPOSITE
	Lead	0.05 mg/L	MONTHLY	24 Hr. COMPOSITE
	Nickel	1.0 mg/L	MONTHLY	24 Hr. COMPOSITE
	Silver	0.05 mg/L	MONTHLY	24 Hr. COMPOSITE
	Zinc	0.03 mg/L	MONTHLY	24 Hr. COMPOSITE
	Total Suspended Solids	45 mg/L	MONTHLY	24 Hr. COMPOSITE
002	pH	6.5 - 8.5	MONTHLY	GRAB
	Temperature	90^{o}F	MONTHLY	GRAB
	T.O.C.	N/A	QUARTERLY	GRAB
	Flow	N/A	MONTHLY	INSTANTANEOUS

By a series of actions discussed in Section 1.3, the volume of solid waste generated has substantially decreased from past experience as can be seen in the following data:

Year	Solid Waste (Ft^3)
1976	5800
1977	5900
1978	6400
1979	4900
1980	3500
1981	3400

1.3 ACTIONS TO REDUCE EFFLUENTS

1.3.1 Hot Cell Filter Bank

The need for radioisotopes gradually increased over the 1960's and 70's as the practice of nuclear medicine became more common. With the increase it was believed to be prudent to add filtration to the effluents of the hot cells. The hot cell filters shown in Figure 1.2 were constructed and made operational in 1978, making a significant reduction in the iodine efflents as seen in the following before and after data for annual I-131 effluents:

Year	I-131 Effluent (Curies)
1977	55.0
1978	40.1
1979	0.8

1.3.2 Replacing Control Rods

Approximately 550 cubic feet of solid radioactive waste is generated by the operation of the reactor each year. The activity in that waste has in recent years increased from approximately 50 millicuries per year to 300 millicuries per year, the primary constituent being Ag-110m. While the Ag-110m has essentially no effect on off-site exposures, because of the long half-life and the high energy gamma it is a significant contribution of occupational exposure. The source of silver contamination has been attributed to poor plating adhesion on the silver indium cadium control rods. A new set of control rods has recently been fabricated to eliminate this source of exposure.

1.3.3 Solid Waste Volume Reduction

In recent years the following steps were taken to reduce the volume of solid radioactive wastes:

- A commercial compactor was installed for low specific activity waste drums.
- Packaging procedures were reviewed to minimize the unnecessary volume of material placed in waste drums.
- Short half-life wastes were segregated so that after decaying to insignificant levels they could be disposed of by conventional methods.
- Special equipment was procured and built and procedures were developed such that the high specific activity waste packaged in hot cells could be better packed and compacted.

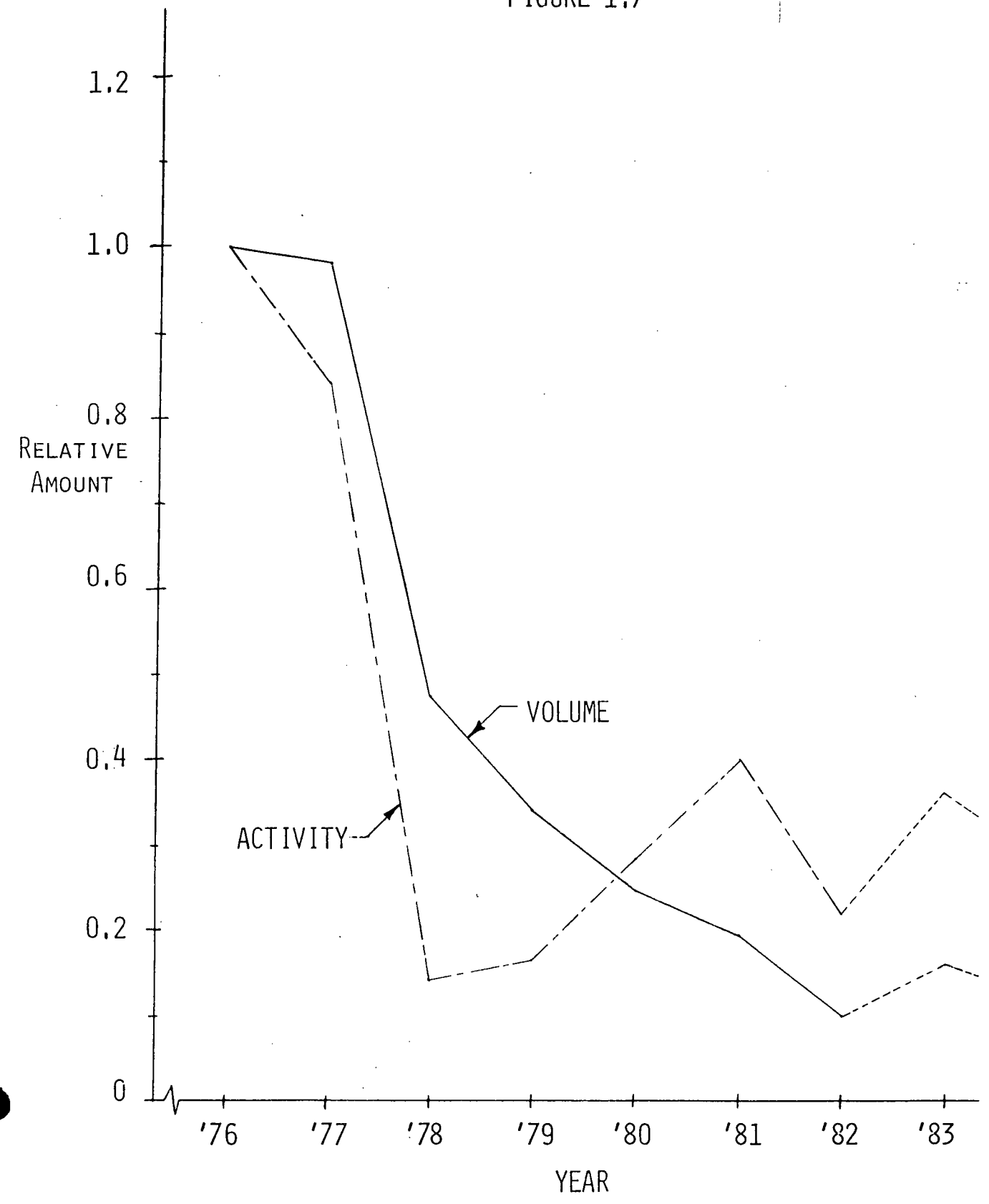
1.3.4 Decay of Solid Waste On Site

By reducing volume the on-site storage time increased, providing additional decay on-site and consequently less activity at the time of off-site shipment and disposal.

In addition to the reduction in activity resulting from better use of existing storage facilities, a new facilty was designed and constructed for decaying hot cell waste drums in storage. In early 1982, waste drums removed from the hot cells were first placed in the specially designed storage pits where they are allowed to decay for six to eight months prior to shipping. The combined effect of volume reduction and decay can be seen in Figures 1.7 and 1.8. Plotted are the wastes shipped per unit of usable radioisotope shipped, normalized to 1976 conditions. The 1982 data in Figure 1.8 is irrelevant in both the Curies and the volume shipped since the normal shipments were suspended six months as the storage facility became operational; however, projected data for 1983 shows a substantially reduced activity in essentially the same volume.

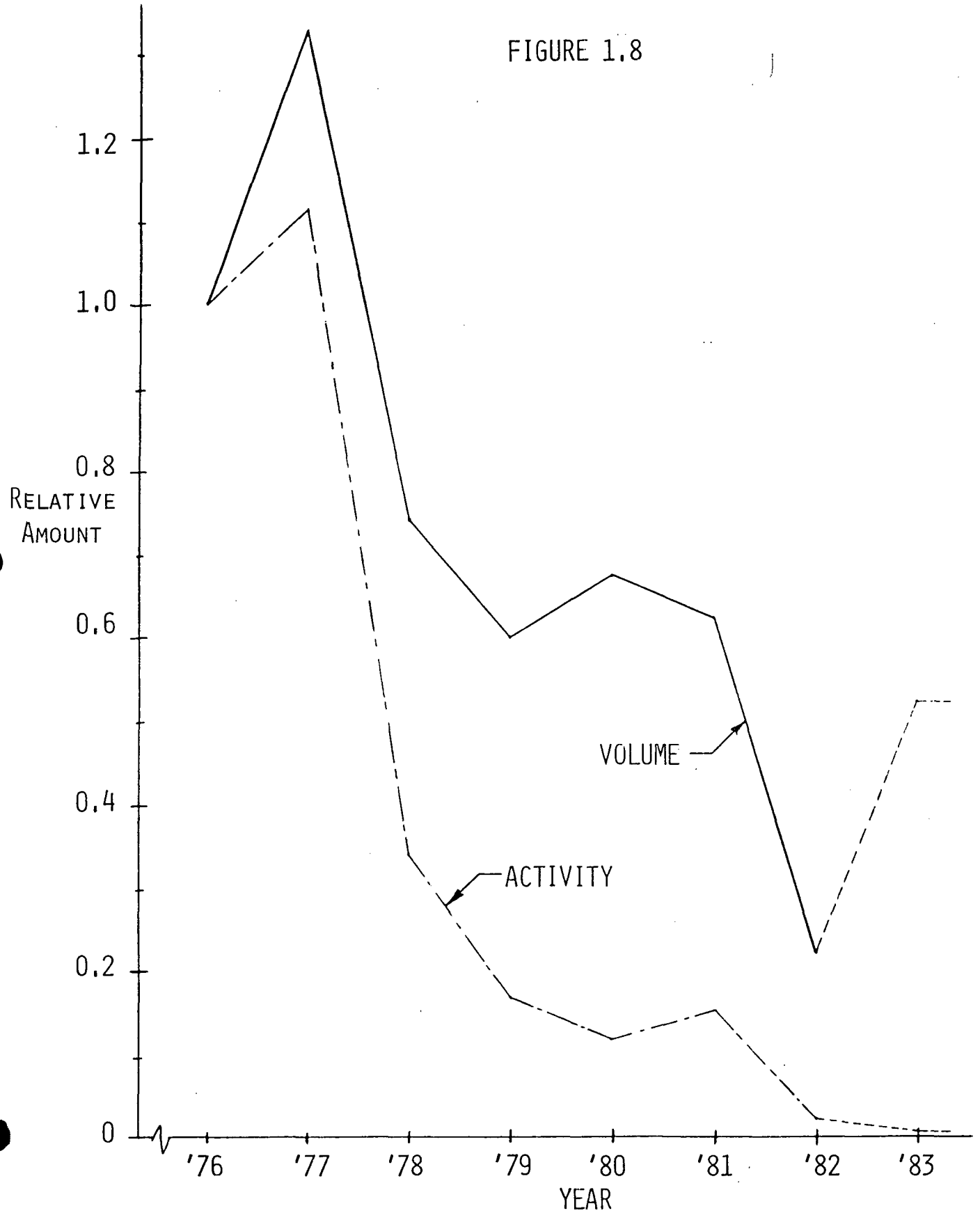
RELATIVE GENERATION OF
LOW SPECIFIC ACTIVITY WASTE

FIGURE 1.7



RELATIVE GENERATION OF
HIGH SPECIFIC ACTIVITY WASTE

FIGURE 1.8



2.0 ENVIRONMENTAL MONITORING

The environmental monitoring program was started in 1957 before the initial use of radioactive material at the facility. This program has been expanded in recent years and now includes five fixed sampling stations and twelve water sampling locations. The current program is directed toward measuring airborne activities and direct radiation from the by-product material used within the facility. Environmental monitoring for material licensed under SNM-639 is precluded because of exhaust stack monitoring results and our liquid accountability procedures.

2.1 LOCATION OF SAMPLING POINTS

Sampling point locations have been chosen to provide measurements of the maximum environmental effects from our operation. Consideration was given to residential areas and the prevailing wind direction. Station E differs from the other stations in that it samples air from an 80 foot tower which is above the tree level in the area. The location of all sampling points is shown in Figure 2.1.

2.2 SAMPLING BY OTHERS

The New York State Department of Environmental Conservation has conducted an environmental monitoring around our facility for many years. In addition special monitoring has been performed by several other agencies. All results have been compatible with our monitoring results and indicate very little environmental effect from our operation.

2.3 SAMPLING RESULTS

Table 2.1 - 2.5 provide a summary of our environmental monitoring results for the period from January 1978 through December 1981. Radioiodine measurements were obtained by a gamma analysis of charcoal canisters operated continuously at our fixed stations. Measurements of radioactivity in water is determined from a gross beta analysis of a boiled down 1000 milliliter sample. Results from the surface water sampling and the direct radiation measurements show no increase due to facility operation.

2.4 RADIOLOGICAL DOSE TO UNRESTRICTED AREAS

We have calculated the potential infant thyroid dose and the noble gas dose from the concentrations measured at Station D which is the nearest residential area. In 1980, we calculated 0.7 man-rem for Iodine-131, 0.5 man-rem for Iodine-125 and 0.5 man-rem for Noble Gas. In 1981, the values were 0.7 man-rem for Iodine-131, 0.3 man-rem for Iodine-125 and 0.3 man-rem for Noble Gas. These results are well within our ALARA commitment to ensure that the dose in the nearest residential area are less than 1 percent of the regulatory limit for unrestricted areas.

FIGURE 2.1

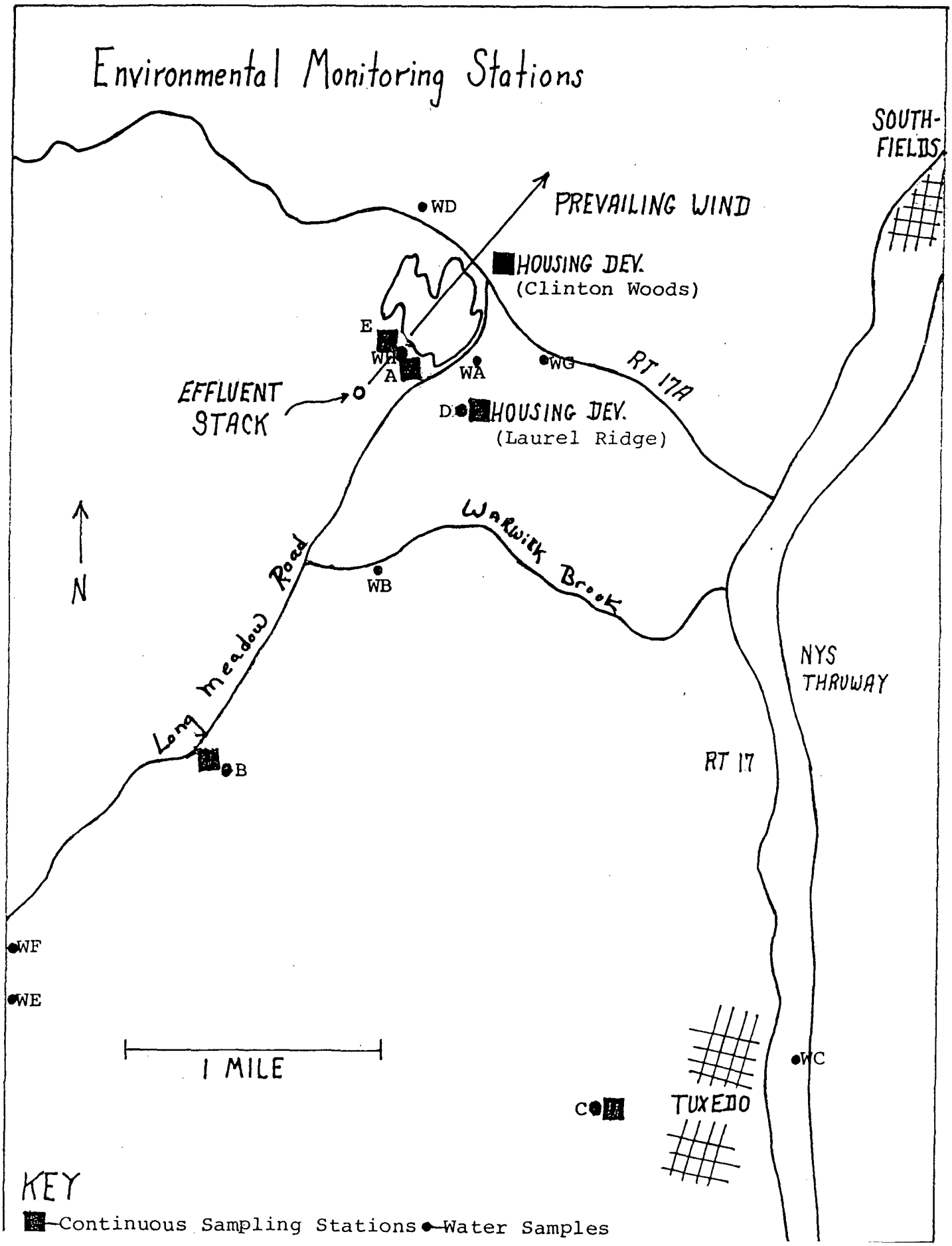

Environmental Monitoring Stations
SOUTH-FIELDS
WD
PREVAILING WIND
HOUSING DEV.
(Clinton Woods)
E
WB
A
WA
WG
RT 17A
EFFLUENT STACK
D
HOUSING DEV.
(Laurel Ridge)
N
Warwick Brook
WB
Long Meadow Road
NYS THRUWAY
B
RT 17
Sterling Lake
WF
WE
1 MILE
WC
C
TUXEDO
KEY
Continuous Sampling Stations
Water Samples

TABLE 2.1

ENVIRONMENTAL AIR AND RAIN WATER RESULTS $\mu Ci/cc$

FOR JANUARY - DECEMBER 1981

MDA I-125 < $1.2 \times 10^{-15} \mu Ci/cc$
MDA I-131 < $1.2 \times 10^{-15} \mu Ci/cc$

MDA Particulate (< $2 \times 10^{-15} \mu Ci/cc$)

MDA Water (< $4 \times 10^{-9} \mu Ci/cc$)

	I-125	I-131	Particulate	Rain Water
Station (A)				
No. Samples	(52)	(52)	(12)	(12)
AVG.	1.07×10^{-14}	4.81×10^{-14}	4.5×10^{-15}	1.25×10^{-8}
MAX.	6.0×10^{-14}	1.38×10^{-13}		
MIN.	0.58×10^{-14}	2.69×10^{-14}		
STATION (B)				
No. Samples	(12)	(12)	(12)	(12)
AVG.	0.15×10^{-14}	2.11×10^{-14}	4.5×10^{-15}	1.12×10^{-8}
MAX.	0.32×10^{-14}	4.13×10^{-14}		
MIN.	$<1.20 \times 10^{-15}$	1.00×10^{-14}		
STATION (C)				
No. Samples	(12)	(12)	(12)	(12)
AVG.	0.13×10^{-14}	1.40×10^{-14}	3.4×10^{-15}	1.42×10^{-8}
MAX.	0.32×10^{-14}	3.70×10^{-14}		
MIN.	$<1.20 \times 10^{-15}$	$<1.20 \times 10^{-15}$		
STATION (D)				
No. Samples	(52)	(52)	(12)	(12)
AVG.	1.18×10^{-14}	4.95×10^{-14}	4.5×10^{-15}	1.41×10^{-8}
MAX.	5.10×10^{-14}	1.42×10^{-13}		
MIN.	0.65×10^{-14}	2.77×10^{-14}		
STATION (E)				
No. Samples	(52)	(52)	(52) (Weekly)	
AVG.	0.98×10^{-14}	4.57×10^{-14}	3.4×10^{-14}	N/A
MAX.	3.40×10^{-14}	1.07×10^{-13}		
MIN.	$<1.20 \times 10^{-15}$	3.12×10^{-14}		

TABLE 2.2

ENVIRONMENTAL AIR AND RAIN WATER RESULTS $\mu Ci/cc$

FOR JANUARY - DECEMBER 1980

MDA I-125 < $1.2 \times 10^{-15} \mu Ci/cc$
MDA I-131 < $1.2 \times 10^{-15} \mu Ci/cc$

MDA Particulate (< $2 \times 10^{-15} \mu Ci/cc$)

MDA Water (< $4 \times 10^{-9} \mu Ci/cc$)

	I-125	I-131	Particulate	Rain Water
Station (A)				
No. Samples	(52)	(52)	(12)	(12)
AVG.	0.81×10^{-14}	3.10×10^{-14}	2.4×10^{-15}	1.12×10^{-8}
MAX.	3.75×10^{-14}	1.07×10^{-13}		
MIN.	$<1.20 \times 10^{-14}$	0.76×10^{-14}		
STATION (B)				
No. Samples	(12)	(12)	(12)	(12)
AVG.	0.13×10^{-14}	0.69×10^{-14}	2.49×10^{-15}	1.40×10^{-8}
MAX.	0.20×10^{-14}	2.15×10^{-14}		
MIN.	$<1.2 \times 10^{-15}$	$<1.2 \times 10^{-15}$		
STATION (C)				
No. Samples	(12)	(12)	(12)	(12)
AVG.	0.20×10^{-14}	0.45×10^{-14}	1.94×10^{-15}	1.44×10^{-8}
MAX.	0.80×10^{-14}	1.41×10^{-14}		
MIN.	$<1.20 \times 10^{-15}$	$<1.20 \times 10^{-15}$		
STATION (D)				
No. Samples	(52)	(52)	(12)	(12)
AVG.	0.46×10^{-14}	2.94×10^{-14}	2.83×10^{-15}	1.00×10^{-8}
MAX.	4.70×10^{-14}	1.09×10^{-13}		
MIN.	$<1.20 \times 10^{-15}$	0.51×10^{-14}		
STATION (E)				
No. Samples	(52)	(52)	(52) (Weekly)	
AVG.	0.70×10^{-14}	2.95×10^{-14}	2.6×10^{-14}	N/A
MAX.	4.38×10^{-14}	1.88×10^{-13}		
MIN.	$<1.20 \times 10^{-15}$	$<1.20 \times 10^{-15}$		

TABLE 2.3

ENVIRONMENTAL AIR AND RAIN WATER RESULTS μCi/cc

FOR JANUARY - AUGUST 1979

MDA I-125 $<4.32 \times 10^{-15}$ μCi/cc
MDA I-131 $<1.73 \times 10^{-15}$ μCi/cc
MDA Particulate ($<2 \times 10^{-15}$ μCi/cc)
MDA Water ($<4 \times 10^{-9}$ μCi/cc)

	I-125	I-131	*Particulate	*Rain Water
Station (A)				
No. Samples	(34)	(34)	(12)	(12)
AVG.	1.01×10^{-14}	2.30×10^{-14}	3.12×10^{-15}	1.05×10^{-8}
MAX.	4.14×10^{-14}	6.82×10^{-14}		
MIN.	$<4.32 \times 10^{-14}$	$<1.73 \times 10^{-14}$		
STATION (B)				
No. Samples	(8)	(8)	(12)	(12)
AVG.	$<4.32 \times 10^{-15}$	1.74×10^{-14}	5.60×10^{-15}	7.00×10^{-9}
MAX.	$<4.32 \times 10^{-15}$	2.00×10^{-14}		
MIN.	$<4.32 \times 10^{-15}$	$<4.73 \times 10^{-14}$		
STATION (C)				
No. Samples	(8)	(8)	(12)	(12)
AVG.	4.30×10^{-15}	1.73×10^{-14}	5.80×10^{-15}	1.00×10^{-8}
MAX.	$<4.30 \times 10^{-15}$	1.75×10^{-14}		
MIN.	$<4.30 \times 10^{-15}$	$<1.73 \times 10^{-14}$		
STATION (D)				
No. Samples	(34)	(34)	(12)	(12)
AVG.	1.05×10^{-14}	2.11×10^{-14}	3.40×10^{-15}	6.00×10^{-9}
MAX.	4.95×10^{-14}	5.36×10^{-14}		
MIN.	$<4.32 \times 10^{-15}$	4.73×10^{-14}		
STATION (E)	Started Sampling	Air Sept.-1979		

*Particulate and rain water results are average values for January-December 1989.

TABLE 2.4

ENVIRONMENTAL AIR AND RAIN WATER RESULTS μCi/cc

FOR SEPTEMBER - DECEMBER 1979

MDA I-125 < 1.2×10^{-15} μCi/cc
MDA I-131 < 1.2×10^{-15} μCi/cc
MDA Particulate (< 2×10^{-15} μCi/cc)
MDA Water (< 4×10^{-9} μCi/cc)

	I-125	I-131	Particulate	Rain Water
Station (A)				
No. Samples	(18)	(18)		
AVG.	3.62×10^{-15}	1.71×10^{-14}		
MAX.	3.58×10^{-14}	1.09×10^{-13}		
MIN.	$<1.20 \times 10^{-15}$	5.20×10^{-15}		
STATION (B)				
No. Samples	(4)	(4)		
AVG.	1.50×10^{-15}	7.93×10^{-15}		
MAX.	2.40×10^{-15}	2.07×10^{-14}		
MIN.	$<1.20 \times 10^{-15}$	$<1.20 \times 10^{-15}$		
STATION (C)				
No. Samples	(4)	(4)		
AVG.	$<1.20 \times 10^{-15}$	3.83×10^{-15}		
MAX.	$<1.20 \times 10^{-15}$	9.80×10^{-15}		
MIN.	$<1.20 \times 10^{-15}$	$<1.20 \times 10^{-15}$		
STATION (D)				
No. Samples	(18)	(18)		
AVG.	8.05×10^{-15}	1.88×10^{-14}		
MAX.	3.60×10^{-14}	1.11×10^{-13}		
MIN.	$<1.20 \times 10^{-15}$	5.40×10^{-15}		
STATION (E)				
No. Samples	(18)	(18)	(4)	
AVG.	1.73×10^{-14}	1.37×10^{-14}	$<1.7 \times 10^{-14}$	N/A
MAX.	1.22×10^{-13}	1.46×10^{-13}		
MIN.	$<1.20 \times 10^{-15}$	$<1.20 \times 10^{-15}$		

TABLE 2.5

ENVIRONMENTAL AIR AND RAIN WATER RESULTS μCi/cc

FOR JANUARY 1978 THRU DECEMBER 1978

MDA I-125 < 2.0×10^{-14} μCi/cc
MDA I-131 < 1.0×10^{-14} μCi/cc
MDA Particulate (<2×10^{-14} μCi/cc)
MDA Water (<3×10^{-8} μCi/cc)

	I-125	I-131	Particulate	Rain Water
Station (A)				
No. Samples	(52)	(52)	(12)	(12)
AVG.	2.15×10^{-14}	5.84×10^{-14}	$<2.00 \times 10^{-14}$	$<3.00 \times 10^{-8}$
MAX.	8.20×10^{-14}	1.94×10^{-12}		
MIN.	$<1.00 \times 10^{-14}$	$<1.70 \times 10^{-14}$		
STATION (B)				
No. Samples	(12)	(12)	(12)	(12)
AVG.	$<1.00 \times 10^{-14}$	$<2.00 \times 10^{-14}$	$<2.00 \times 10^{-14}$	$<3.00 \times 10^{-8}$
MAX.	$<1.00 \times 10^{-14}$	$<2.00 \times 10^{-14}$		
MIN.	$<1.00 \times 10^{-14}$	$<2.00 \times 10^{-14}$		
STATION (C)				
No. Samples	(12)	(12)	(12)	(12)
AVG.	$<1.00 \times 10^{-14}$	$<2.00 \times 10^{-14}$	$<2.00 \times 10^{-14}$	$<3.00 \times 10^{-8}$
MAX.	$<1.00 \times 10^{-14}$	$<2.00 \times 10^{-14}$		
MIN.	$<1.00 \times 10^{-15}$	$<2.00 \times 10^{-14}$		
***STATION (D)**				
No. Samples	(16)	(16)	(4)	(4)
AVG.	$<1.00 \times 10^{-14}$	$<2.00 \times 10^{-14}$	$<2.00 \times 10^{-14}$	$<3.00 \times 10^{-8}$
MAX.	$<1.00 \times 10^{-14}$	$<2.00 \times 10^{-14}$		
MIN.	$<1.00 \times 10^{-14}$	$<2.00 \times 10^{-14}$		

*STATION (D) was started August 1978

3.0 RADIATION PROTECTION PROGRAM

Union Carbide has a structured radiation safety program with a Health Physics staff equipped with radiation detection equipment to determine, control, and document radiation exposures within the facility and off site.

3.1 ALARA AND HEALTH PHYSICS PROGRAM

Union Carbide's Health Physics Program has been committed to the ALARA concept for over 20 years. The current program represents the culmination of our continuing efforts to minimize radiation exposure and provide a safe working environment. This comprehensive Health Physics Program addresses radiation protection through employee training, engineered safeguards, and an elaborate program of monitoring employees, work areas and the environment. The occupational radiation exposures directly associated with materials under our license SNM-639 is a very small part of the exposure associated with operation of the facility. We have estimated that this exposure is less than 1 percent of the exposure received during the processing, quality control, and packaging of our products. For 1981, this would be less than 1 man-rem of exposure. The other 90 man-rems is a direct result of the large quantities of by-product material we work with. In New York State by-product material is licensed by the Department of Labor. Our license with the State is 729-0322.

3.2 PLANT AND EQUIPMENT DESIGN

Plant and equipment design at UCC emphasizes handling ease and radiological control. Shielding of personnel, concentrating of worker effort in low radiation background areas, and minimizing work in higher areas of radiation are the techniques used to minimize dose and maintain the standards for protection against radiation in 10 CFR Part 20 and appropriate regulatory guides. Planning and process procedures are under the surveillance of the Nuclear Safeguards Committee (NSC) and ALARA considerations are formulated by NSC and the General Safety Committee. ALARA recommendations are directed to the General Safety Committee concerning operational radiation control, and ALARA for new projects is handled by NSC. In addition, a file is maintained which documents ALARA recommendations.

3.3 ORGANIZATION AND RESPONSIBILITIES

The Manager of Health, Safety, and Environmental Affairs is the site Radiation Safety Officer. He also administers the SNM-639 license as Site Accountability Officer. As such, he has responsibility for both the safety and accountability of the special nuclear material (SNM) on site.

The Health Physics Supervisor is responsible for the activities of the Health Physics Department, regarding control of SNM. His program consists of air sampling and analysis, contamination and exposure control, respiratory protection and bioassay. The Health Physics Supervisor presently has five technicians reporting to him who carry on the Health Physics Program under his direction.

3.4 FACILITY MODIFICATIONS

The reactor and hot cell complex is designed to handle the SNM and by-product material in a safe and efficient manner. The facility features a hot cell bank of 5 cells with four-foot thick high density concrete walls designed to effectively shield one million Curies of Cobalt-60. A twelve-foot deep canal connecting the hot cell bank and reactor ensures ALARA exposures during the transfer of the fission products resulting from irradiated targets.

Experience from past design is utilized in redesign of production process systems including U-235 work areas. These design changes are directed at reducing contamination to the lab in which the operation takes place. Second generation design changes are used in some cases to effect contamination control through a change in process method (U-235 target preparation). These modifications ensure that occupational radiation exposures are ALARA and that contamination incidents are negated.

General design guidance includes a criteria for dose rates less than 2 mR/hr in occupied work areas, effective contamination control and adequate ventilation for hoods and gloveboxes where airborne radioactivity could be significant.

Although the facility is over 20 years old it was originally designed to provide shielding and confinement for much larger amounts of radioactivity than we currently use. In spite of this overdesign we have continually made improvements to more closely match current operation to the ALARA concept. Some examples of this facility improvement are:

1. The conveyor station, which provides a means for moving material into and out of the hot cell bank, has been modified several times. The current system provides a shielded viewing window, a remote manipulator, and a radiation controlled interlock on the access door. In addition the drive unit for the conveyor in the hot cell has been relocated outside the shielded area so that maintenance can be performed with minimum radiation exposure.

2. The waste handling facilities have been improved and expanded to minimize radiation exposure. We now have a separate building to store low level waste prior to shipment off-site for disposal. This minimizes storage of waste in occupied areas which results in unnecessary exposure. We have also developed remote handling techniques which allow us to remove higher level waste from the hot cells and transfer it to highly shielded storage pits for decay prior to shipment. We also use this facility to safely store radioactive material during hot cell maintenance which also results in reduced exposure.

3. Another major example is the design and operation of the activated filters in the exhaust system for the hot cells. The original filters were located in the exhaust duct of each of the five hot cells. Experience gained from operating this system indicated that significant personnel exposure was received during routine maintenance of these filters because of their in cell location.

   A new system was designed which combined all carbon filters in a shielded pit in the hot cell exhaust outside of the hot cells. This system resulted in reduced personnel exposure during filter maintenance and was used for several years until increased production levels led to a second redesign of the system.

   Using the experience gained from operating the existing system a larger dual system was designed and installed which requires minimum maintenance and allows for isolating each part of the parallel configuration so that it can be decayed prior to maintenance. This resulted in a major reduction in employee exposure and also reduced off site releases by more than a factor of ten.

3.5 HEALTH PHYSICS PROCEDURES AND EXPOSURE HISTORY

The Health Physics Department ensures that occupational radiation exposures are ALARA through a daily monitoring program of air sampling, radiation monitoring and contamination measurements throughout the plant. Exposures to personnel above 30 millirem in any one day are followed by an investigation report to determine the cause so that the dose may be reduced. The detailed plans and procedures for this ALARA control program are based on many years of experience in operating the facility safely and on standard good Health Physics practices.

The occupational radiation exposure from material under our SNM-639 license is a very small part of the exposure associated with operation of the Reactor and Hot Laboratory. This material has also not caused any significant airborne radioactivity or environmental releases. During the years 1976 through 1978, the total man-rems for the Reactor and Hot Laboratory averaged 70 man-rems per year. For 1979 through 1981 the yearly exposures were: 73, 83, and 91 man-rems. During this same period of time our isotope production increased by almost 150 percent.

3.6 HEALTH PHYSICS INSTRUMENTATION

All portable and laboratory technical equipment and instrumentation is selected for reliability and efficiency for performing radiation and contamination surveys, airborne radioactivity sampling, area radiation monitoring, and personnel monitoring during normal operation and accident conditions. Technical equipment and instrumentation are maintained by an electronics department having responsibility for repair. Calibration of equipment is performed on a periodic basis by the Health Physics Department, using appropriate calibrated sources for each specific instrument type. Instruments not in use are stored in the Health Physics Laboratory. Health Physics radiation detection instrumentation is quite extensive and includes many instruments that detect gamma radiation at levels far in excess to that produced by special nuclear material under normal conditions. Under abnormal conditions, an extensive criticality monitoring system will sense the higher levels of radiation produced and activates the emergency evacuation system.

Health Physics facilities are equipped with instrumentation for alpha, beta, and gamma analysis of liquid and airborne radioactive materials. The health physics instrumentation is located in a laboratory suitably segregated from the operating areas of the facility, but close enough to these areas to permit rapid identification and quantification of radioactive samples. Showers and standard decontamination agents are available for use for personnel decontamination. Other contamination control equipment available includes shoe covers, lab coats, coveralls, and rubber gloves. Alpha and beta-gamma personnel friskers are used to detect contamination on personnel. A supplied air respiratory protection system is available for use in potential airborne radioactivity areas. The facility is designed and maintained so that respiratory protection is normally not required. However, in the event of an emergency, self-contained breathing apparatus (Scott Packs) are available for use. This emergency protective equipment is located at each entrance to the facility. Other protective gear for contamination control is located at the entrances to the areas where it is used.

The radiation detectors and monitors presently available include the following:

Type	Number Available	Calibration Method*	Range
Ion Chamber	5	Cesium-137	1 mR/h 1000 r/h
Ion Chamber	11	Cesium-137	1 mR/h 50 r/h
Portable Geiger Counter	7	Cesium-137	0.1 mR/h 50 mR/h
Alpha Scintillation Detector	6	Alpha Source	1 dpm 6×10^6 dpm
Gas Flow Proportional Counter	1	Alpha Source	1 dpm 1×10^6 dpm
Criticality Monitors	13	Cesium-137	0.1 mR/h 10000 mR/h
Gamma Multichannel Analyzer	2	Point Source Isotope	Up to 10^5 cpm

Calibration frequency for these instruments is at least quarterly.

*All calibration sources are traceable to NBS

3.7 HEALTH PHYSICS MONITORING

Contamination surveys are conducted daily in areas of potential contamination. Approximately 55 smear samples are collected and evaluated. Survey results are then transmitted to the supervisors of the areas surveyed. This is an integral part of the ALARA program. Radiation level surveys are conducted at least monthly in work areas, and more frequently as the situation warrants for areas in which significant quantities of radioactive materials are used.

ALARA is further implemented via investigation reports of exposure. All daily exposure levels above 30 man-rem are investigated. In addition, exposure histories are examined before planning work in radiation areas, and workers are chosen for the job according to good ALARA practices.

Access to high radiation areas is controlled by Health Physics via special authorization, using standardized radiation protection requirements or through lock and key control and mandatory Health Physics surveillance. Stay time and number of workers exposed is determined by good ALARA practices (i.e. criteria of ALARA for total man-rems).

Personnel are monitored for gamma, beta, and fast neutrons using film dosimetry. In addition, personnel wear gamma pocket chambers which are evaluated daily by Health Physics. A criticality accident, should one occur, would also be evaluated with this dosimetry. Reports of exposure are listed for each individual in an exposure history record, which clearly shows his total lifetime and permitted occupational exposure. Bioassay is performed at least once per year on radiation workers. Special bioassay samples are obtained whenever it appears that an individual may have been exposed to excessive airborne radioactivity.

Potential airborne radioactivity concentrations are evaluated and controlled by approximately twenty-five fixed air monitors whose collection filters are analyzed each day for radioactivity. Appropriate warnings are placed at boundaries to areas in which an exposure to more than 25 percent of MPC might occur. Special breathing zone air samplers are used by personnel in areas in which breathing zone air activity might be higher than that recorded by the fixed air sample near that location.

4.0 ACCIDENT ANALYSIS OF OPERATIONS WITH SPECIAL NUCLEAR MATERIAL

Accidents involving special nuclear material have been divided into three general categories, namely:

(a) Natural Phoenomena
(b) Abnormal Occurrences
(c) Credible Serious Accidents

4.1 NATURAL PHENOMENA

4.1.1 Windstorm

Since the Hot Laboratory is built into the side of Hogback Mountain and is constructed of steel and masonry, windstorm damage can be precluded. Over the past 20 years of operation, several severe storms (hurricanes) have occurred without causing damage or operational problems.

4.1.2 Lightning

Lightning could interrupt the power to the facility or cause a fire in the surrounding forest. In case of power failure, normal operations are suspended. An emergency power system will automatically supply power to critical equipment. (Refer to 4.2.1)

4.1.3 Forest Fire

The location of the building and its mansonry exterior construction should protect the critical equipment and areas from damage due to forest fire in the area.

4.1.4 Flood Hazard

The facility is located on the side of a hill that provides excellent drainage. The only credible flooding could be caused by flash flood. This is not likely to occur at the Hot Laboratory site because of the lack of collection area or fetch at local higher elevations. With one exception (See 4.2.3) a ground-water collection system around the building foundations is designed to carry away sub-surface water.

4.1.5 Earthquake

The New Jersey-New York Highlands have a long record of freedom from violent earthquakes. There is no historical record of earthquakes of intensity VIII or greater occurring in this area.

The Hot Laboratory is constructed in very firm hard rock. Only a very violent shock could affect it. Such a structure and particularly the hot cells would vibrate as a unit even under violent shock, which is not expected.

In the past 20 years, the number of seismic recording stations in this general area has increased steadily to the point where detection of events adjacent to New York City of magnitudes > 1.8 is practically complete. Coupled with interest in the seismic safety of the power reactors located at Buchanan, New York, this has resulted in intense study of the seismic activity in the vicinity of the Ramapo Fault and estimates of the probability of occurrence of strong earthquakes at those reactor sites. In Ref. 1 a detailed discussion of these matters appears. What follows is excerpted from this reference.

The UCNR reactor site is interior to a lithospheric plate. In such a case, potential earthquakes tend to occur along major pre-existing faults, with the larger shocks showing a greater tendency than the smaller ones to be located on the major throughgoing fault. A survey of the seismic events that have occurred in and around this region shows that most of the activity is in the Precambiran Hudson Highlands, that earthquakes in this area occur along pre-existing faults, with the large majority within 1-2 Km of the faults. About 50 percent of all events are almost collinear and lie along or close to the Ramapo Fault system. The Hot Laboratory site is located 12 Km from the closest point of approach to this fault.

Using data on small shocks obtained in the past 20 years or so from the seismic network, a relation for cumulative frequency of occurrence has been obtained. Extrapolation to larger magnitudes shows excellent agreement with the few larger historical events, of intensity VI and VII. This relation has been used by the authors (Ref. 1) to predict the probability of occurrence of intensity VII and VIII shocks at the upper end of the Ramapo Fault. The results realistically are as follows:

	Recurrence Period (Years)		Probabilty of Occurrence in 20-Year Period (Percent)	
	VII	VIII	VII	VIII
(a) Excluding events > 10 Km distant:	630	2870	3.2	0.7
(b) Including all events along Ramapo Fault:	340	1880	5.9	1.1

The UCNR site is about 12 Km northwest from the Ramapo Fault and therefore the predicted probabilities at this site should be less than those given above, viz. less than 3-6 percent for intensity VII, and less than 1 percent for intensity VIII, for a 20-year period in the future. (Ref. 2)

References:

1. Aggarwal, Yash P., and Sykes, Lynn R., "Earthquakes, Faults, and Nuclear Power Plants in Southern New York and Northern New Jersey", Science 200, 425 (Apr. 1978).

2. Aggarwal, Yash, and Miller, Henry, "Analysis of Seismic Activity Related to UCC Facility at Sterling Forest, New York", Ocean Seismic Survey, Inc. Report, (Dec. 1980).

4.2 ABNORMAL OCCURRENCES

4.2.1 Loss of Power

Normal power to the facilty is supplied from the local utility company by above-ground transmission lines that are affected occasionally by severe weather. Two emergency generators (50 Kw and 35 Kw), in different locations within the Hot Laboratory will start and pick up the emergency equipment load automatically when normal power to the building is interrupted. One generator is fueled with gasoline, the other with natural gas. A six day supply of gasoline is maintained in a dedicated underground tank outside of the building at the north loading dock.

The Hot Laboratory equipment that can be operated from this emergency power system are as follows:

(a) Normal ventilation at 1/2 capacity.
(b) Auxiliary exhaust fan at full capacity.
(c) Backup hot cell exhaust blower.
(d) Radiation monitoring and alarm systems.
(e) Selected electrical circuits located throughout the building for operating portable safety equipment and emergency ligthing.
(f) Gasoline feed pump to the emergency generator.

Normal operations in the Hot Laboratory are suspended while normal power is interrupted.

4.2.2 Fire

Combustible materials are stored in one area at the north end of the Hot Laboratory. This area is equipped with an automatically activated sprinkler system. This is the only area where enough combustible material exists to sustain a fire of any significant magnitude for significant periods of time.

All other areas are equipped with portable extinguishers or are capable of being serviced by fire hose from fire hydrants. Hydrants are connected to a dedicated fire main system which is connected to a storage and head tank on Hogback Mountain in addition to the site water supply main.

A significant fire requiring outside assistance has never occurred in the past 20 years of operation.

4.2.3 Flooding

It was indicated in Section 4.1 that flooding could occur inside the building due to infiltration of groundwater. This is possible to occur in the waste water storage tank (T-1) room located beneath the radiochem laboratory. This room is the only part of the facility that is below grade and that is not serviced by an exterior drainage system.

After prolonged heavy rains, water accumulates under the foundations of this room and enters a sump in the floor at the east end. Sump pumps (redundant pumps for reliability) automatically pump this water to the process waste water system where it is stored, sampled and analyzed for radioactivity before it is released. Remote alarms located in the reactor control room indicate when water is present in the sump and also when the pumps are operating.

Radioactivity has never been detected in this groundwater.

4.2.4 Loss of Hot Cell Ventilation

The probability of having airborne contamination in the hot cell exhaust ventilation is very high. To prevent leakage of airborne radioactivity from the cells, it is imperative to maintain exhaust ventilation through proper filters, and also to keep the air pressure inside the cells at a lower level than the outside pressure . The loss of normal ventilation for any reason is

sensed by a ΔP switch in the suction plenum of the main Hot Lab exhaust fan. If this pressure difference goes above -1" H_2O column (W.C.), the exhaust vent load is automatically switched to an auxiliary Hot Lab exhaust fan. The auxiliary fan will exhaust all cells and hoods through the main filter bank until normal ventilation can be restored.

In the event both fans fail, another backup blower that is used occasionally for maintenance work will be put on the line automatically when the cell pressure rises to -.1" W.C. This blower (1000 cfm capacity) is sufficient to maintain a slight negative pressure in the cells as long as all access ports are kept closed. The blower discharge is filtered through HEPA and charcoal filters. This blower can also be powered from a portable auxiliary generator.

4.2.5 Accidental Spill of Radioactive Material

Operations with radiochemical products in liquid and gaseous form are common in the hot laboratory. It is credible that a container loaded with 10's or 100's of curies of a radioisotope could accidentally become separated from its shielding. If this were to occur in either the reactor or hot laboratory facility, the area monitors, which are placed to monitor all controlled areas that are routinely inhabited, would sound at least one local and remote alarm. This would warn people in the locale of the spill and they would vacate the immediate vicinity. Alarms would continue to sound until the source was removed or shielded. The radiation level at the monitor remote readout located either in the reactor control room or the hot lab operating area, would show the actual radiation dose at the locale of the exposed source. Corrective action could then be planned and implemented by the emergency crew. There would be time to gather shielding, remote handling tools, remote TV equipment, or whatever was necessary to correct the problem without exposing the workers. In the event such a spill would cause airborne radioactivity, constant air monitors which are located on each level of the building, would alarm and the area would be evacuated. Protective and corrective measures could then be taken under the Emergency Plan.

4.3 CREDIBLE SERIOUS ACCIDENTS

4.3.1 Release of Fission Product Dissolver Batch in the Mo-99 Separation Process

The maximum credible accident in the Mo-99 separation process is assumed to be caused by the leakage of dissolved, mixed fission products during the dissolution of the uranium oxide target. At this point in the process, the dissolver solution containing the

fission products could be at 100 psig pressure or greater and at a temperature of 100°C. Such an accident could be caused by personnel error or equipment failure. The failure would occur at a rubber septum fitting at the end of the dissolver vessel and releases would occur as follows:

(a) All of the fission product iodine.
(b) All of the fission product noble gases.
(c) All of the fission product solution in the form of a fine spray.

Each target is allowed to decay a minimum of 6 hours before it is opened and processed. The fission product inventory, normalized to a high yield batch (350 Ci Mo-99 after 10 days irradiation) is calculated by the ORIGEN2 Code. The amount released and the consequential doses are calculated as follows:

A. Fission Product Iodine

Iodine inventory after 6-hour decay:

I-131	104 Curies
I-132	236 Curies
I-133	364 Curies
I-135	215 Curies

Iodine activity release from exhaust stack assuming a filter collection efficiency of 99.5 percent:

I-131	.52 Curies
I-132	1.18 Curies
I-133	1.82 Curies
I-135	1.075 Curies

Total stack exhaust flow per week:

$(5 \times 10^4\ Ft^3/min.)(2.83 \times 10^4 cc/Ft^3)\ (1.44 \times 10^3\ min/day)$
$(7\ days/wk.) = 1.4 \times 10^{13}\ cc/wk.$

Iodine concentration averaged over one week:

I-131 $.51 \times 10^6\ \mu Ci / 1.4 \times 10^{13}\ cc = 3.7 \times 10^{-8}\ \mu Ci/cc$
I-132 $1.18 \times 10^6\ \mu Ci / 1.4 \times 10^{13}\ cc = 8.4 \times 10^{-8}\ \mu Ci/cc$
I-133 $1.82 \times 10^6\ \mu Ci / 1.4 \times 10^{13}\ cc = 1.3 \times 10^{-7}\ \mu Ci/cc$
I-135 $1.075 \times 10^6\ \mu Ci / 1.4 \times 10^{13}\ cc = 7.68 \times 10^{-8}\ \mu Ci/cc$

Maximum permissible concentration using 400 dilution factor. (Assuming sigma theta 2.5° and wind speed of 1 m/sec toward nearest off-site resident.)

I-131	$.4 \times 10^{-7}$ µCi/cc
I-132	12×10^{-7} µCi/cc
I-133	1.6×10^{-7} µCi/cc
I-135	4×10^{-7} µCi/cc

Fraction of MPC averaged over one week:

I-131	.9
I-132	.07
I-133	.8
I-135	.2

Total iodine released averaged over one week = 1.97 MPC-wks.

B. Noble Gases

Noble gas inventory after 6-hours decay:

Isotope	Curies	µCi/cc /wk	Isotope	Curies	µCi/cc /wk
Xe-133m	11	7.85×10^{-7}	Kr-85m	33	2.36×10^{-6}
Xe-133	280	2×10^{-5}	Kr-85	.03	2.14×10^{-9}
Xe-135m	34	2.4×10^{-6}	Kr-88	54	3.86×10^{-6}
Xe-135	222	1.59×10^{-5}			

Total noble gas activity = 624 Curies.

Maximum permissible concentrations using a dilution factor of 400:

	Xenon		Krypton	
133m	1.2×10^{-4}	µCi/cc	85m	0.04×10^{-4} µCi/cc
133	1.2×10^{-4}	µCi/cc	85	1.2×10^{-4} µCi/cc
135m	0.4×10^{-4}	µCi/cc	88	$.08 \times 10^{-4}$ µCi/cc
135	0.4×10^{-4}	µCi/cc		

Fraction of MPC averaged over one week:

Xe 133m	.006	Kr 85m	.06
Xe 133	.16	Kr 85	0
Xe 135m	.06	Kr 88	.48
Xe 135	.4		

Total fraction of weekly MPC = 1.17

C. Fission Product Aerosol

Assumptions:

(a) Fission product inventory with 6-hour decay.

(b) Gross activity-[iodines + noble gases] = 4.66×10^3 Ci.

(c) Physical form: In solution with a minimum volume of 62.5 mL.

The solution released in the form of a spray will have a particle diameter range of from 10 microns up to greater than 1000 microns. (Reference 1). The filter efficiency for particles greater than 5 microns is 100 percent, therefore no stack release will result from this spray.

The solution released in the form of a fog will have a particle diameter range from less than 0.1 microns up to greater than 5 microns (Reference 2). With this particle size distribution, 0.0313 percent of the activity could be released (Reference 2).

Values for the liquid content of fogs range from about 0.01 g/m^3 to 1 g/m^3 (Reference 3). With this particle size distribution, 0.0313 percent of the activity could be released (Reference 2).

Values for the liquid content of fogs range from about 0.01 g/m^3 to 1 g/m^3 (Reference 3). Assuming the generation of five cubic meters of fog the maximum amount of radioactivity reaching the HEPA filters will be:

$$\frac{5 \text{ mL (in } 5m^3 \text{ of fog)}}{62.5 \text{ mL}} \quad 4.66 \times 10^3 \text{ Ci} = 3.7 \times 10^2 \text{ Ci}$$

Activity released:

$(3.7 \times 10^{2}\ \text{Ci})\ \ (3.13 \times 10^{-4}) = 1.17 \times 10^{-1}\ \text{Ci}$

Aerosol concentration averaged over one week:

$1.17 \times 10^{5}\ \mu\text{Ci} \ /\ 1.4 \times 10^{13}\ \text{cc/wk} = 8.35 \times 10^{-9}\ \mu\text{Ci/cc}$

The maximum permissible concentration for 6 hour old mixed fission product activity is $1 \times 10^{-8}\ \mu\text{Ci/cc}$ (Reference 4). Using this MPC will be conservative because it includes iodine activities which we have considered separately.

Fraction of MPC released averaged over one week:

$$\frac{8.35 \times 10^{-9}\ \mu\text{Ci/cc}}{1 \times 10^{-8}\ \mu\text{Ci/cc}} = .835\ \text{MPC-wks}$$

In summary, the maximum credible accident could result in a release of radioactivity which, when averaged over one week, would represent the following percentages of our off-site MPC.

	Percent of MPC-Wks
Fission Product Iodine	200
Noble Gases	120
Fission Product Aerosol	80
TOTAL Release	400

This represents 4 weeks release at the maximum permissible rate. It represents a maximum release since we have assumed iodine collection efficiencies which will only exist at the end of the useful life of a filter.

The collection efficiencies of all carbon filters will be determined routinely. Filters will be replaced when the effective iodine collection efficiency of the system falls below that claimed in the hazards analysis (99.5 percent). It also assumes a very stable meteoroligic condition. If the average measured dilution factor were used, these releases would be 25 times less or 0.16 MPC weeks.

References

Reference 1 — ORNL TM-2300, "Design and Construction of High-efficiency Air Filtration System for Nuclear Applications", (Sept. 1968).

Reference 2 — "Hazards Report for Radioactive Materials Laboratory", Revised 1-3-67.

Reference 3 — Particulate Clouds: Dusts, Smokes and Mists. H. L. Green and W. R. Lane, D. VanNostrand Co., Inc., 1957.

Reference 4 — Health Physics, Pergamon Press, Inc., Vol. 4, Numbers 3/4, April 1961.

4.3.2 Release of Fission Products From The Waste Form Process

This accident is defined as the release of the entire contents of an aluminum waste container into the cell as a cloud of dust during the calcining step of the waste form process. (Ref. 1) The accident would be the result of a thermal excursion, due to improper temperature control. A rapid temperature rise to a temperature in excess of 500°C could possibly create pressure inside the vented can in excess of the can burst pressure, since the decomposing uranyl acetate would generate gases faster than the venting rate, releasing some of the uranium and fission product particulates into the cell.

The following conservative assumptions were made for the design basis accident:

- Maximum allowed batch size of 200 grams U-235
- Minimum practical decay time of 10 days
- Maximum irradiation time of 320 hours (average time is approximately 240 hours)
- Average neutron flux seen by targets (since multiple targets are required for a batch, the average flux is representative)

The results of these conditions is a theoretical batch inventory of 11,000 Curies. In reality, the inventory based on the above conservative assumptions is approximately 3,000 Curies because the following materials are removed prior to the calcining step:

- The Mo-99 which is separated for radiopharmaceutical purposes
- The iodine and noble gases which are trapped during the Mo-99 separation process, many of these by-products likewise having radiopharmaceutical applications
- Approximately 60 percent of the remaining mixed fission product activity which precipitates or co-precipitates with the addition of barium acetate prior to the drying and calcining step

The main concern with the design basis accident is the potential offsite particulate release. Noble gases and iodines are removed earlier. The potential for each of the other remaining fission products to volatilize has been evaluated and determined not to be

consequential, a conclusion which has been verified during the startup program. In the event that particulate material were dispersed in the hot cell and entered the ventilation system, it would be filtered by two HEPA filters in series, each having a rated efficiency of 99.97 percent for particles larger than 0.3 microns, which is a much smaller particle size than expected for a dust dispersal. Assuming no plating out or settling of activity in the system and the theoretical batch inventory of activity, the maximum release would be:

$$11{,}000 \text{ Curies} \times (3 \times 10^{-4})^2 = .001 \text{ Curie} = 1{,}000 \ \mu\text{Ci}$$

Reference 2 presents MPC values for mixed fission products as a function of decay time. For 10 day old mixed fission products in air, the environmental MPC is 1.33×10^{-9} μCi/cc. Assuming no atmospheric dispersion, dilution by the normal 50,000 cfm exhaust ventilation flow from the facility results in a potential release of 0.4 MPC-day for the design basis accident.

While the potential release of the postulated design basis accident is very small, numerous conservativisms should be pointed out which, in a more realistic assessment, would make the potential consequences even smaller:

1. The calculation is based on a release of 11,000 Curies, in reality only 3,000 Curies would be expected in the most restrictive conditions.

2. The dust must first pass through roughing filters in the hot cell into a plenum below the table. Then it must pass through approximately 10 feet of 10 inch diameter duct and 50 feet of three foot diameter duct to reach the first HEPA filter. It must then pass through three charcoal beds containing a total of 4 3/4" of activated charcoal and additional ducting before reaching the final HEPA filter. Some fraction of the dust would settle or plate out before the final HEPA filters, as well as in the 400 foot long, four foot diameter duct which goes up the hill from the Hot Laboratory to the exhaust stack.

3. No credit is taken for atmospheric dispersion due to the elevated release of the fine particles escaping the HEPA filters; the 0.4 MPC-day result is based on the concentration in the stack itself. The average measured dilution factor for the site boundary in the direction of the nearest residential area is 11,000.

4. In the model for the design basis accident, it is postulated that the temperature rises to 500°C instantly so that all the uranyl acetate would decompose at this temperature, leading to a high pressure and can rupture. If, in fact, some of the uranyl acetate had already decomposed prior to the temperature rise, the pressure developed at 500°C might be low enough so that the can could vent instead of rupturing and much smaller amounts of powder would be dispersed. In addition, it has not actually been proven under actual operating conditions that the container can be ruptured at 500°C.

5. The assumption of instant temperature rise is an essentially impossible condition for two reasons. First, the thermal inertia of the furnace/can system is such that it would take approximately 45 minutes to reach 500°C from the normal operating temperature of 300°C at full power. Second, there would have to be multiple simultaneous failures in redundant electrical and mechanical safety devices associated with the temperature control circuitry for the temperature excursion to occur.

Reference 1 UCC letter of application for License Amendment SNM License 639 Docket 70-687 dated 6/2/80.

Reference 2 D.L. Summers and M. C. Gashe, Health Physics Journal, Vol. 4, No. 3/4, p. 289, 1961.

4.3.3 Criticality Incident In The Hot Laboratory.

Enriched U-235 solutions (350 g/l, 1 liter/batch or 5 g/l, ~ 16 liters/batch) are processed in the second level plating labs of the Hot Laboratory. Administrative and process controls are established to keep individual plating batches of material at less than 1/2 the single-parameter mass limit for solutions. Vessels that contain plating solution batches are of limited size. Accidental criticality is not likely to occur but, since the total quantity possessed in the entire Hot Laboratory exceeds the single-parameter limit, it is postulated that enough material could be brought together to cause a criticality incident. The plating labs are the only areas where solutions of uranium (other than assay samples) are handled and it is the most likely location for an accidental excursion to occur.

Regulatory Guide 3.34 states that the magnitude of accidental criticality incidents ranged from 10^{15} to 10^{20} fissions depending upon the circumstances involved in each. The median magnitude was 10^{17} fissions. The regulatory position states that a criticality event may be assumed as follows:

1. An initial excursion of 1×10^{18} fissions of 1/2 second duration.

2. 47 subsequent excursions of 1.9×10^{17} fissions at 10 minute intervals continuing for 8 hours.

3. 1×10^{19} total fissions over the course of the entire event.

A less conservative assumption may be made if it can be rationalized when considering the particular plant and process conditions.

The solutions of uranium that are processed under SNM-639 are of limited quantity and they are contained in open vessels of limited size. The initial excursion of 1×10^{18} fissions in 1/2 second will yield 64 MW of power. This is enough energy (32 MW sec) to evaporate most of the largest volume of solution (16 L) that is routinely in use. The initial burst will be the only excursion possible. The instantaneous evaporation or expulsion of the uranium solution out of its container will render the system subcritical.

The fission products are assumed to be dispersed throughout the upper level of the Hot Laboratory and they mix with the air in the building. Some will be exhausted through the hoods in the plating lab and will pass through two stages of HEPA filters. Some will be exhausted through the hot cells and will pass through two stages of HEPA and charcoal filters. All of the air will pass through the main filter bank and will be filtered at least once through HEPA filters before it goes to the stack. The Hot Laboratory ventilation system is set up to exhaust more air than what is supplied and the balance of the supply is provided by infiltration through seams in the structure. There will be no appreciable leakage from the building without it being filtered.

The following assumptions are made to assess exposures as a result of the above described accident:

a. The least favorable atmospheric condition exists. (Wind speed, 1 m/sec.; sigma theta, 2.5°; wind direction toward the closest offsite resident and road).

b. Dilution factor based on measurement data.

c. The main exhaust fan operates at normal capacity and the mean residence time of air in the building is 12.5 minutes.

d. The plume transit time to the site boundary (275 meters) is 4.6 minutes.

e. No credit is taken for iodine plate-out or holdup on building surfaces, filters, or ducts.

f. MPC's for isotopes not listed in Part 20 are estimated based on relative whole body exposure hazard from gamma (relative to the most conservative limit for other similar gaseous isotopes that are listed).

The airborne release from this criticality accident is calculated as follows:

Assume source term of A_0 Curies released in the Hot Lab of volume V (376,000 ft^3) with exhaust flow rate F (30,000 cfm). Release is instantaneous. Since exhaust is drawn from all parts of the building the concentration leaving the building will be $A_b(t)/V$ (Ci/ft^3) where
$A_b(t)$ is the activity in the building as a function of time. There are two loss terms, purging and decay.

$$\frac{d A_b}{d t} = - A_b F/V - \lambda A_b = - A_b (F/V + \lambda)$$

$$A_b (t) = A_o \, e^{- (F/V + \lambda) t}$$

The concentration of radioactivity leaving the building (a_b) at any time will be:

$$a_b (t) = \frac{A_b(t)}{V} = \frac{A_o}{V} \, e^{- (F/V + \lambda) t}$$

The concentration at the site fence (a_f) will be the activity leaving the building at time (t-T), where T = the transit time, decayed an additional time T, and dispersed by D, the dilution factor.

$$a_f (t) = \frac{a_b (t - T)}{D} \, e^{-\lambda T} = \frac{A_o \, e^{-\lambda T}}{VD} \, e^{-(F/V + \lambda) (t - T)}$$

$$a_f (t) = \frac{A_o \, e^{F/V \, T}}{VD} \, e^{-(F/V + \lambda) t}$$

The dose D_f in terms of equivalent MPC-time is the integrated concentration of activity from the time the plume arrives (T) until it is totally purged.

$$D_f = \int_T^\infty \frac{a_f (t) \, d t}{MPC} = \frac{A_o \, e^{F/V \, T}}{VD \; MPC} \int_T^\infty e^{- (F/V + \lambda) t} \, d t$$

$$D_f = \frac{A_o \, e^{-\lambda T}}{VD \; MPC \; (F/V + \lambda)}$$

where:

$$T = \frac{\text{distance}}{\text{wind speed}} = \frac{300\ \text{m}}{1\ \text{m/sec}} = 300\ \text{sec.} = 5\ \text{min.}$$

$F/V = 30{,}000$ cfm

$V = 376{,}000\ ft^3$

$F = 0.0798\ \text{min.}^{-1} = 1.33 \times 10^{-3}\ \text{sec.}^{-1}$

$$\lambda = \frac{.693}{T\ 1/2}$$

The dose in terms of MPC-wks at the site boundary in the direction of the nearest residential area is calculated as follows:

For the plant boundary, and for the nearest residential area, D = 400 for Pasquill Type F conditions. Subsituting the above in the expression for D_f gives

$$D_f = 3.97 \times 10^{-13}\ A_0/MPC \quad x \quad e^{-300\lambda}/(1.33 \times 10^{-13}\lambda)\ \text{MPC-wks},$$

where: A_0 = activity released (Ci)

MPC is in μCi/cc

λ is disintegration constant (Sec.^{-1})

A single pulse of 10^{18} fissions is considered because all chain-reacting liquid will be expelled from the small container used. It is assumed that all noble gases and 25 percent of the iodines are released through the building stack. Aerosols will be trapped by the filters.

The airborne releases and the resulting integrated doses in MPC-weeks are given in Table 4.1 for each significant radionuclide.

The integrated whole body dose from all the noble gases is 13.6 MPC-weeks, which is equivalent to about 130 mR. The integrated thyroid dose from all the iodines is 14.1 MPC-weeks, equivalent to about 400 mR.

TABLE 4.1

Airborne Releases and Integrated Doses From Inhalation

($1x10^{18}$ Fission; "F" Stability, Wind 1 M/s)

Nuclide	T 1/2	A_o (Ci)	MPC (μCi/cc)	D_f (MPC-wks)
Kr-83m	1.86 h	14.8	6×10^{-2}	0
Kr-85m	4.5 h	15.2	1×10^{-7}	.06
Kr-85	10.7 y	.00016	3×10^{-7}	0
Kr-87	76.0 m	104	2×10^{-8}	1.33
Kr-88	2.8 h	66.6	2×10^{-8}	.93
Kr-89	3.2 m	4620	$1.4x 10^{-8}$	8.77
Xe-131m	11.9 d	.0072	$4.0x 10^{-7}$	0
Xe-133m	2.0 d	.13	$3.0x 10^{-7}$	0
Xe-133	5.2 d	2.8	$3.0x 10^{-7}$	0
Xe-135	9.1 h	29	$1.0x 10^{-7}$	.085
Xe-135m	15.6 m	215	$2.0x 10^{-7}$	.16
Xe-137	3.8 m	5000	$1.0x 10^{-7}$	1.85
Xe-138	14.2 m	1380	$5.0x 10^{-7}$	0.4
I-131	8 d	.19	$1.0x 10^{-10}$	.57
I-132	2.3 h	24	$3.0x 10^{-9}$	2.2
I-133	20.8 h	4.2	$4.0x 10^{-10}$	3.1
I-134	52.6 m	113	$6.0x 10^{-9}$	4.5
I-135	6.6 h	12.7	$1.0x 10^{-9}$	3.68
			Total Nobel Gas =	13.6
			Total Iodines =	14.1 (MPC-wks)

The beta skin dose from the plume is calculated as follows. From the preceding section, the time-integrated concentration of activity at the receptor site is given by C_f, where:

$$C_f = D_f \times MPC = \frac{A_o e^{-\lambda T}}{VD\,(F/V + \lambda)}$$

Substituting the previous values of V, D, F gives:

$$C_f = \frac{2.35 \times 10^{-7} A_o e^{-300\lambda}}{(1.33 \times 10^{-3} + \lambda)} \quad \frac{Ci - sec}{m^3}$$

with A_o in Curies
λ in sec^{-1}

The dose to the skin from beta rays is given, using the semi-infinite assumption, by a summation over all significant nuclides of the following expression evaluated for each nuclide:

$$D_B = 3.16 \times 10^4 \; C_f \cdot DFS \qquad mrem$$

where: DFS = beta skin dose factor (Table B-1, Reg. Guide 1.109)
3.16×10^4 = conversion factor, mR-m^3/pCi-y to mR-m^3/Ci-s

Calculated results are shown in Table 4.2.

TABLE 4.2

Beta Skin Doses

Nuclide	A_o(Ci)	λ (Sec^{-1})	DFS ($mR\text{-}m^3/pCi\text{-}y$)	D_B (mRem)
Kr-83m	14.8	1.04 E-4	3.47 E-4	.03
Kr-85m	15.2	4.30 E-5	1.46 E-3	.12
Kr-85	.00016	2.05 E-9	1.34 E-3	0
Kr-87	104	1.52 E-4	9.73 E-3	4.8
Kr-88	66.6	6.88 E-5	2.37 E-3	.82
Kr-89	4620	3.66 E-3	1.01 E-2	23.2
Xe-131m	.0072	6.69 E-7	4.76 E-4	0
Xe-133m	.13	3.60 E-6	9.94 E-4	0
Xe-133	2.8	1.52 E-6	3.06 E-4	0
Xe-135m	215	7.55 E-4	7.11 E-4	.43
Xe-135	29	2.92 E-5	1.86 E-3	.29
Xe-137	5000	3.01 E-3	1.22 E-2	42.4
Xe-138	1380	8.14 E-4	4.13 E-3	15.5
			TOTAL	87.6

The prompt direct radiation doses at the nearest, normally occupied site boundary are calculated as follows. A single excursion of 1×10^{18} fissions in a laboratory within the Hot Laboratory is assumed. Successive excursions are not credible due to the nature of the containers used for the SNM. Areas necessary to maintain the plant in the safe condition after the excursion can thus be occupied without hazard from later excursions.

a. Prompt Gamma Dose:

$$D_{\gamma} = 2.1 \times 10^{-20} \times 1 \times 10^{18}/(d^2 e^{3.4d})$$

$$= .021/(d^2 e^{3.4d}) \text{ rem, d=distance (km)}$$

The last expression gives the unshielded dose. Any intervening concrete will reduce the dose.

(1) At Site Boundary:

Distance, d = 300 m
Total of 12-in concrete shielding is in the line of sight, for a reduction factor of 5.

$$D_{\gamma} = .021/ (5x .3^2 \times \exp (3.4x.3)) = .017 \text{ rem}$$

(2) At Nearest Residential Area:

Here d = 450 m; shielding is also 12" concrete.

$$D_{\gamma} = .021/(5 \times .45^2 \times \exp (3.4 \times .45)) = .0045 \text{ rem}$$

b. Prompt Neutron Dose:

$$D_n = 7 \times 10^{-20} \times 1 \times 10^{18}/(d^2 \, e \, 5.2d)$$

$$= 7 \times 10^{-2}/ (d^2 \, e \, 5.2d) \text{ rem}$$

In this case the shielding factor for 12" concrete is 4.6.

(1) At Site Boundary

$$D_n = 7 \times 10^{-2}/ (4.6 \times .3^2 \times \exp (5.2 \times .3) =$$
.036 rem

(2) At Nearest Residental Area

$$D_n = 7 \times 10^{-2} /(4.6 \times .45^2 \times \exp (5.2 \times .45)) =$$
.007 rem

To summarize, the resultant exposures from this postulated criticality is estimated to be as follows:

1. Prompt gamma dose at the closest, normally occupied point along the site boundary.

   = .017 rem

2. Prompt gamma dose at the closest residential area.

   = .0045 rem

3. Prompt neutron dose at the closest, normally occupied point along the site boundary.

   = .036 rem

4. Prompt neutron dose at the closest residential area.

   = .007 rem

5. Projected airborne release to the closest, normally occupied point along the site boundary from:

   (a) Noble Gas = 13.6 (MPC-wks) (.130 rem)

   (b) Iodines = 14.1 (MPC-wks) (.4 rem)

6. Projected beta skin dose at the closest, normally occupied point along the site boundary

   = .088 rem

It should be noted that if the average measured dilution factor were applied in this postulated accident, the projected airborne release and consequent doses would be reduced by more than a factor of 25.

A criticality accident occurring inside the hot cell bank would have insignificant consequences due to the extra shielding and filtration.

5.0 Benefits Derived From Operations at the Sterling Forest Facility

5.1 Nuclear Medicine

The major endeavor at this facility is to provide a variety of radioactive chemicals that serve as the active ingredients of diagnostic drugs used in the practice of Nuclear Medicine. Over 75 percent of the in-vivo diagnostic procedures depend upon Technetium 99m (Tc-99m) a six-hour half-life radionuclide which is the daughter of Molybdenum-99 (Mo-99), a 66-hour half-life radionuclide. The Sterling Forest Laboratory currently provides approximately 50 percent of all the Mo-99 requirements in the United States and approximately 25 percent of that required in W. Europe and Japan. The isotopes, Iodine-131, Xenon-133 and Iodine-125 are also important diagnostic agents that are provided from this laboratory. The Sterling Forest Laboratory is the only source of these radioisotopes that is located within the U.S..

The number of Nuclear Medicine diagnostic procedures that are performed in the U.S. each year and which use isotopes supplied from this facilty are as follows:

	No./Yr.	Percent UCC Contributes
Mo-99/Tc-99m Diagnostic Scans		
Brain	960,000	50 Percent
Liver/Spleen	1,400,000	"
Bone	1,660,000	"
Thyroid	240,000	"
Lung	720,000	"
Heart	420,000	"
Kidney	160,000	"
TOTAL	5,620,000	50 Percent
I-131 Diagnostic Scans		
Thyroid	200,000	50 Percent
Kidney	40,000	"
TOTAL	240,000	50 Percent
Xe-133 Lung Scans	340,000	30 Percent
I-125 Radiommunoassay	120,000,000	30 Percent

5.2 Other Contributions

In addition to radioisotopes, other benefits that are derived include:

(a) Neutron Transmutation Doping of Silicon is performed which provides a superior grade of Si-based semi-conductor material which is primarily used in high-power control devices. Through the use of such devices there is a significant conservation in electrical energy.

(b) Radioactive Sources are made for use as implants for tumor therapy. The active ingredient for approximately 300,000 implants are provided annually.

(c) Numerous custom services using nuclear technology are also provided. Such projects include making reference standards for industry and activating materials for R and D work by others in private industry. Approximately 50 to 100 such individual projects benefit from the technology provided by the Sterling Forest Laboratory annually.

The aforementioned benefits should be included in the environmental appraisal for proper perspective.