SIEMENS

ECLIPSE CYCLOTRON

Radiation Safety Aspects

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List of Acronyms

GUI	graphic user interface
ICRP	International Commission on Radiological Protection
NCRP	National Council on Radiation Protection and Measurement
PPE	Personal Protective Equipment
QF	quality factor
SF	Screening factor
TLVs	tenth value layers
TSUs	Target support units

1.0 Introduction to the Eclipse Cyclotron

This document describes radiation safety aspects of the Eclipse family of cyclotrons. It includes information regarding installation, operation, and maintenance of the cyclotron, targetry, and the associated equipment. Consequences of various accident scenarios are analyzed to aid customers in the preparation of their application for a radioactive materials license and/or an accelerator registration. Such scenarios include the failures of gaseous and liquid targets that have been identified as presenting credible hazards. In most cases, these scenarios are models, but some have also been corroborated by actual experience.

2.0 System Description

The Eclipse cyclotron is a compact, self-shielded, cyclotron-based system used to produce positron-emitting isotopes. The system includes an 11 MeV negative-ion proton accelerator that is capable of extracting up to 60 μA through one beamline or 120 μA through two beamlines simultaneously.

There are two types of cyclotrons in the Eclipse family: the HP, and RD. The HP is rated at 60 μA and has 4 targets/beamlines. The RD is rated at 40 μA and has 8 targets/beamlines.

The systems include the necessary targets and processing apparatus for producing labeled products and chemical precursors. Routine operation is automatic. The graphic user interface (GUI) is oriented toward the end product rather than intermediate physics and engineering parameters.

The Eclipse family of cyclotrons is designed to meet the following criteria.

- The system shall be installed in a room of solid concrete block wall construction.
- Radiation levels inside the cyclotron room and in the immediate vicinity of the shield shall be consistent with exposure guidelines established for monitored personnel (i.e., operators and technicians working in a controlled-access environment where radioactive materials are prepared and dispensed).
- The radiation levels outside the cyclotron room are dependent on the design and construction of the room walls. Calculated radiation levels are provided to the customer for guidance; however, these are based on design information provided by the customer. The assumptions used in the calculations are conservative, and wall thicknesses are chosen so that the radiation fields present during operation are consistent with guidelines established for general occupancy.

Radiation protection guidelines are published by national and international advisory committees, such as the National Council on Radiation Protection and Measurement (NCRP) and the International Commission on Radiological Protection (ICRP). In particular, it is recommended that NCRP Report No. 147 be reviewed.

Maintenance, modification, and repair of the cyclotron and its accessories shall be done only by Siemens personnel or by Siemens-trained and -authorized Customer Service Personnel. Such maintenance, modification, and repair include handling of parts of the cyclotron, beamlines, and related components that become radioactive as a result of direct proton irradiation or of secondary neutron activation. Handling of other radioactive materials, setting up of chemical synthesis procedures, and monitoring of radiation levels and emissions is the responsibility of the customer, and must be carried out by qualified personnel trained and certified in accordance with state and local regulations.

3.0 Hazard Description

3.1 Prompt Radiation Fields

Radiation exposure during operation comes from a combination of gamma and neutron radiation. On the surface of the shield, around regions nearest to a target undergoing bombardment and near seams between shield blocks, the neutron dose equivalent is approximately 50% of the total measured dose. Further away from the shield, around the inside perimeter of the cyclotron room, the neutron dose equivalent averages only 25% of total measured dose.

The room that houses the cyclotron must be posted as a controlled area. Entering the cyclotron room during target bombardment does not create a radiation hazard during normal operation, so the entrance need not be interlocked. But it is recommended that the door be kept locked to prevent casual entry by unauthorized persons. While the level of prompt radiation from the accelerator in the cyclotron room is consistent with allowable levels for radiation workers, anyone entering must be wearing a film badge or personal dosimeter.

3.1.1 Shielding

The shield provides protection against radiation from the cyclotron and its targets in keeping with the design guidelines outlined above. Walls of the cyclotron room provide additional shielding. In general, rooms in which radioactive materials are handled and processed, such as b adjoining laboratories, sample prep rooms, scanning rooms, etc., should be maintained as controlled areas, reducing the need for additional shielding in the walls other than that which is provided by typical building materials. For areas immediately adjacent to the cyclotron room that cannot be controlled access, design of the room walls as additional shielding should be considered. The following section is provided as an aid to determine room dimensions, additional wall shielding, and access controls specific to the facility being considered. This document does not provide information necessary to determine the required shielding of a non-self-shielded system. Please contact Siemens for non-self-shielded designs.

The first part of this section addresses specifications and measurements of dose rate, and validation of simplified models for calculating dose rate outside the cyclotron self shield system. The second part addresses further measurements of neutron spectra outside the cyclotron self-shield for purposes of completing radiation shielding and facility planning, and further adjustments to simplified modeling to allow for attenuation calculations.

4.0 Measurements, Validation, and Specifications of Dose Rates

4.1 Initial Measurements and Validation of a 1/r^2 Model

Dose rates for neutrons and gammas have been measured at 40 points around an operating Eclipse cyclotron, running a dual 60 μA bombardment and equipped with the copper door option. A graphic of the points used in the measurements is shown Figure 1 and the readings are laid out in Table 1.

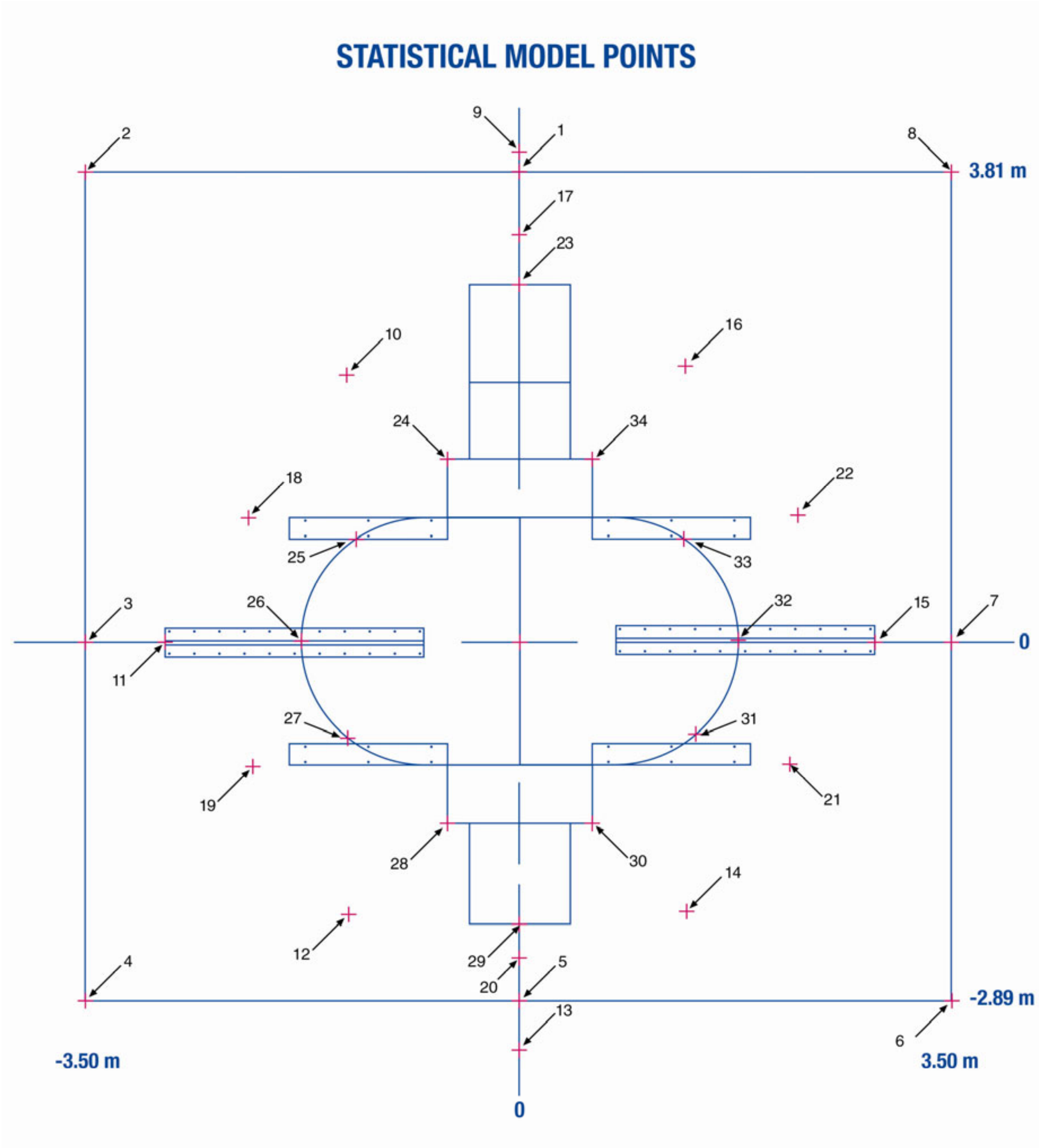

Figure 1 Statistical Model Points for the Cyclotron Eclipse

Table 1 Statistical Model Points Reading

	Position (cm)			Actual (mrem/hr)			Model (mrem/hr)		
Point	x	y	z	Gamma	Neutron	Total	Gamma	Neutron	Total
1	0	381	100	1.2	0.9	2.1	1.3	0.6	1.9
2	-351	381	100	1.3	0.6	1.9	0.7	0.3	1.1
3	-351	0	100	2.2	0.6	2.8	1.5	0.7	2.2
4	-351	-290	100	1.5	0.3	1.8	0.9	0.4	1.4
5	0	-290	100	1.3	0.6	1.9	2.2	0.9	3.1
6	351	-290	100	1.1	0.3	1.4	0.9	0.4	1.4
7	351	0	100	1.5	0.6	2.1	1.5	0.7	2.2
8	351	381	100	1.2	0.5	1.7	0.7	0.3	1.1
9	0	394	100	0.8	1.1	1.9	1.2	0.5	1.8
10	-140	218	100	3.0	1.4	4.4	2.7	1.2	3.8
11	-282	0	100	2.8	0.3	3.1	2.3	1.0	3.3
12	-140	-218	100	3.4	1.6	5.0	2.7	1.2	3.8
13	0	-333	100	1.0	0.9	1.9	1.7	0.7	2.4
14	140	-218	100	2.6	0.7	3.3	2.7	1.2	3.8
15	282	0	100	2.3	0.3	2.6	2.3	1.0	3.3
16	140	218	100	3.0	1.3	4.3	2.7	1.2	3.8
17	0	325	100	0.9	1.4	2.3	1.8	0.8	2.5
18	-208	102	100	3.7	0.4	4.1	3.2	1.4	4.6
19	-290	0	100	2.1	0.5	2.6	2.2	0.9	3.1
19	-208	-102	100	4.5	1.3	5.8	3.2	1.4	4.6
20	0	-264	100	1.3	0.8	2.1	2.6	1.1	3.7
21	208	-102	100	3.2	0.5	3.7	3.2	1.4	4.6
22	290	0	100	1.8	0.8	2.6	2.2	0.9	3.1
22	208	102	100	3.7	2.2	5.9	3.2	1.4	4.6
23	0	295	100	1.0	0.8	1.8	2.1	0.9	3.0
24	-41	119	100	22.0	50.0	72.0	7.9	3.4	11.3
25	-145	81	100	7.0	1.4	8.4	5.5	2.4	7.8
26	-183	0	100	8.0	1.2	9.2	4.7	2.0	6.8
27	-145	-81	100	9.5	2.1	11.6	5.5	2.4	7.8
28	-41	-119	100	13.5	10.5	24.0	7.9	3.4	11.3
29	0	-234	100	1.6	0.8	2.4	3.2	1.4	4.6
30	41	-119	100	12.0	12.4	24.4	7.9	3.4	11.3
31	145	-81	100	6.5	0.7	7.2	5.5	2.4	7.8
32	183	0	100	7.0	0.4	7.4	4.7	2.0	6.8
33	145	81	100	8.0	2.3	10.3	5.5	2.4	7.8
34	41	119	100	24.5	58.6	83.1	7.9	3.4	11.3
35	-127.0	0.0	231	1.7	0.4	2.1	2.9	1.3	4.2
36	0.0	0.0	231	65.0	54.6	119.6	3.8	1.7	5.5
37	127.0	0.0	231	1.4	1.0	2.4	2.9	1.3	4.2
38	-127.0	0.0	330	3.3	0.3	3.6	1.6	0.7	2.3
39	0.0	0.0	330	13.0	7.2	20.2	1.9	0.8	2.7
40	127.0	0.0	330	3.0	0.4	3.4	1.6	0.7	2.3

The transport of neutrons and gammas through air is expected to follow $1/r^2$ form, owing to low attenuation of both forms of radiation by air. The data best fits a $1/r^2$ model based on one point source located at the center of the accelerator. The modeled gamma source strength is 20.5 mrem/hr at 1 meter from the center point, and the modeled neutron source strength is 8.9 mrem/hr at 1 meter from the center point. Using these strengths and locations of point sources, the goodness of fit (R value) for the gamma measurement is .786 and for neutrons is .581 (ideal fit is R=1.00).

The poorness of the fit to the neutron data can be ascribed to the nature of the radiation. Neutrons are inherently non-interactive and require long counting times. Moreover, thermal neutrons diffuse as a gas, and therefore each gap in the shields acts as a radiator. As the distances from the center point increase, the $1/r^2$ model becomes more valid. However, the low radius measurements have high variation.

4.1.1 One-Meter Radiation Profile

Points 9 through 16 in the statistical model table (shown previously in Table 1) represent a sampling of points that are 1 meter from the surface of the shielded accelerator. The maximum gamma dose rate is 3.4 mrem/hr at point 12. The maximum neutron dose rate is 1.60 mrem/hr at point 12. The maximum total dose rate is 5.0 mrem/hr at point 12. This supports the claim that the 5mrem/hr isocontour lies entirely within an envelope 1 meter from the surface.

4.2 Siemens Box Model Description

Given the $1/r^2$ nature of the radiation field, a simplified model for radiation estimation purposes has been developed. The Siemens box model is a simplified calculation tool that allows quick, simple, and conservative estimates of the radiation field around an Eclipse cyclotron. It also is intended to harmonize with specifications for Siemens Eclipse accelerator shielding, which is defined on a box-shaped surface analogous to a real room boundary.

The box model defines the boundary where the total radiation levels above the ambient background will be at or below the specified field (total dose from neutrons and gammas) provided the accelerator is operated with the targetry and operational guidelines established in below. The radiation boundary is shown in Figure 2 below. This model is intended to define a simple geometry and single criterion for acceptable shield performance. The actual field on the surface defined by the box is significantly lower than the specified level at almost every point.

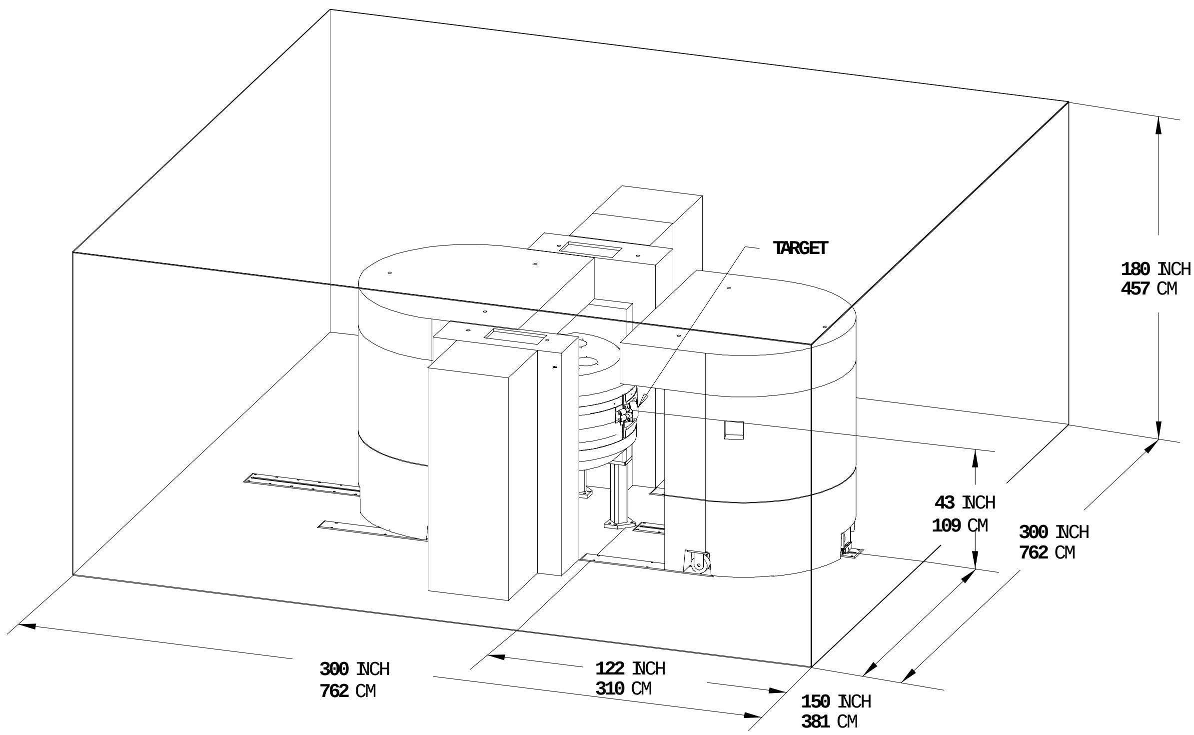

Figure 2 Siemens Box Model

The highlighted (762 cm x 762 cm x 457cm) box defines the radiation boundary. For purposes of conservative estimation of required additional shielding and as shown in the figure, the source of radiation can be considered as originating at the center of the cyclotron; a point 381 cm to the left of the right hand corner, 381 cm in front of the right hand corner and 100 cm above the level of the floor. This is the basis for the Eclipse box model calculation. Note that the radiation field at most points on the box model is significantly lower than 2.0 mrem/hr.

4.3 Dose Rate Summary from Factory Acceptance Measurements

The dose rate around each Eclipse cyclotron is surveyed in the factory as part of final acceptance test. The strongest neutron flux can be generated by dual bombardments at specified current (40 µA or 60 µA) depending on configuration of shields. A summary of statistics of the survey data from the first 99 cyclotrons produced is shown below in Table 2. This also represents further validation of the box model above.

Table 2 Survey Data from Cyclotron 1 Through 99

System	Radiation type (mrem/hr)		Position							
			1	**2**	**3**	**4**	**5**	**6**	**7**	**8**
Shield model: HP	neutrons	average	0.4	0.4	0.3	0.4	0.5	0.4	0.4	0.4
Number of systems: 27		std. dev.	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2
beam current: 60	gammas	average	0.5	0.7	1.0	0.8	0.9	0.8	1.1	0.9
		std. dev.	0.2	0.3	0.3	0.3	0.3	0.4	0.4	0.3
	total	average	**1.0**	**1.0**	**1.3**	**1.2**	**1.4**	**1.2**	**1.5**	**1.2**
		std. dev.	0.4	0.3	0.4	0.3	0.4	0.4	0.4	0.4
Shield model: RD	neutrons	average	0.3	0.3	0.3	0.4	0.5	0.3	0.3	0.3
Number of systems: 64		std. dev.	0.3	0.2	0.2	0.2	0.3	0.2	0.2	0.2
beam current: 40	gammas	average	1.0	0.6	0.7	0.8	1.5	0.7	0.5	0.6
		std. dev.	0.6	0.3	0.4	0.5	0.8	0.4	0.4	0.4
	total	average	**1.3**	**1.0**	**1.0**	**1.1**	**2.0**	**1.0**	**0.8**	**0.9**
		std. dev.	0.8	0.4	0.5	0.6	1.0	0.5	0.5	0.5

System	Radiation type (µSv/hr)		Position							
			1	**2**	**3**	**4**	**5**	**6**	**7**	**8**
Shield model: HP	neutrons	average	4.1	3.6	3.5	3.8	5.1	3.9	3.7	3.8
Number of systems: 27		std. dev.	2.3	2.3	2.0	1.9	2.1	2.1	1.8	2.0
beam current: 60	gammas	average	5.4	6.8	9.7	8.5	9.3	8.2	11.0	8.6
		std. dev.	2.4	2.8	3.2	3.0	2.7	3.5	3.5	3.5
	total	average	**9.5**	**10.4**	**13.2**	**12.3**	**14.4**	**12.1**	**14.7**	**12.4**
		std. dev.	3.5	3.4	3.5	3.4	4.0	3.9	4.0	3.9
Shield model: RD	neutrons	average	3.3	3.1	3.0	3.6	4.5	3.2	2.5	2.8
Number of systems: 64		std. dev.	2.9	2.4	2.1	2.4	3.2	2.0	1.9	2.2
beam current: 40	gammas	average	9.8	6.5	6.8	7.7	15.1	6.8	5.3	6.4
		std. dev.	5.9	3.1	4.0	4.7	8.0	3.9	3.6	4.0
	total	average	**13.1**	**9.6**	**9.9**	**11.4**	**19.6**	**10.0**	**7.8**	**9.3**
		std. dev.	7.9	4.2	4.8	6.0	9.9	4.9	4.8	5.4

4.4 Neutron Energy Measurements

Measurements for determination of the neutron spectrum at three locations outside the shield of a Siemens Eclipse cyclotron were performed on May 25, 2005. Locations for these measurements and their dose rates are reported in Table 3. Unfolded results for spectra are obtained by using an on-line unfolding code and are shown in Figure 3. The scaled dose rates assume that the measurement at each location is accurate and that the relative calculated dose rates are also correct.

Table 3 Dose Rates at Each Location

Location	Position X	Position Y	Position Z	Total Fluence	Average Energy	Unfolded Dose	Uncalibrated Dose Rate	Scaled Dose Rate*
#1	-97	333	56	246.4 n/cm^2	0.12 MeV	1.014e-06 REM	3.6 mrem/hr	0.9 mrem/hr
#2	-124	429	145	138.5 n/cm^2	0.18 MeV	5.815e-07 REM	2.1 mrem/hr	0.6 mrem/hr
#3	-325	0	145	43.7 n/cm^2	0.20 MeV	3.159e-07 REM	1.1 mrem/hr	0.3 mrem/hr

* Based on a measurement at location 3.

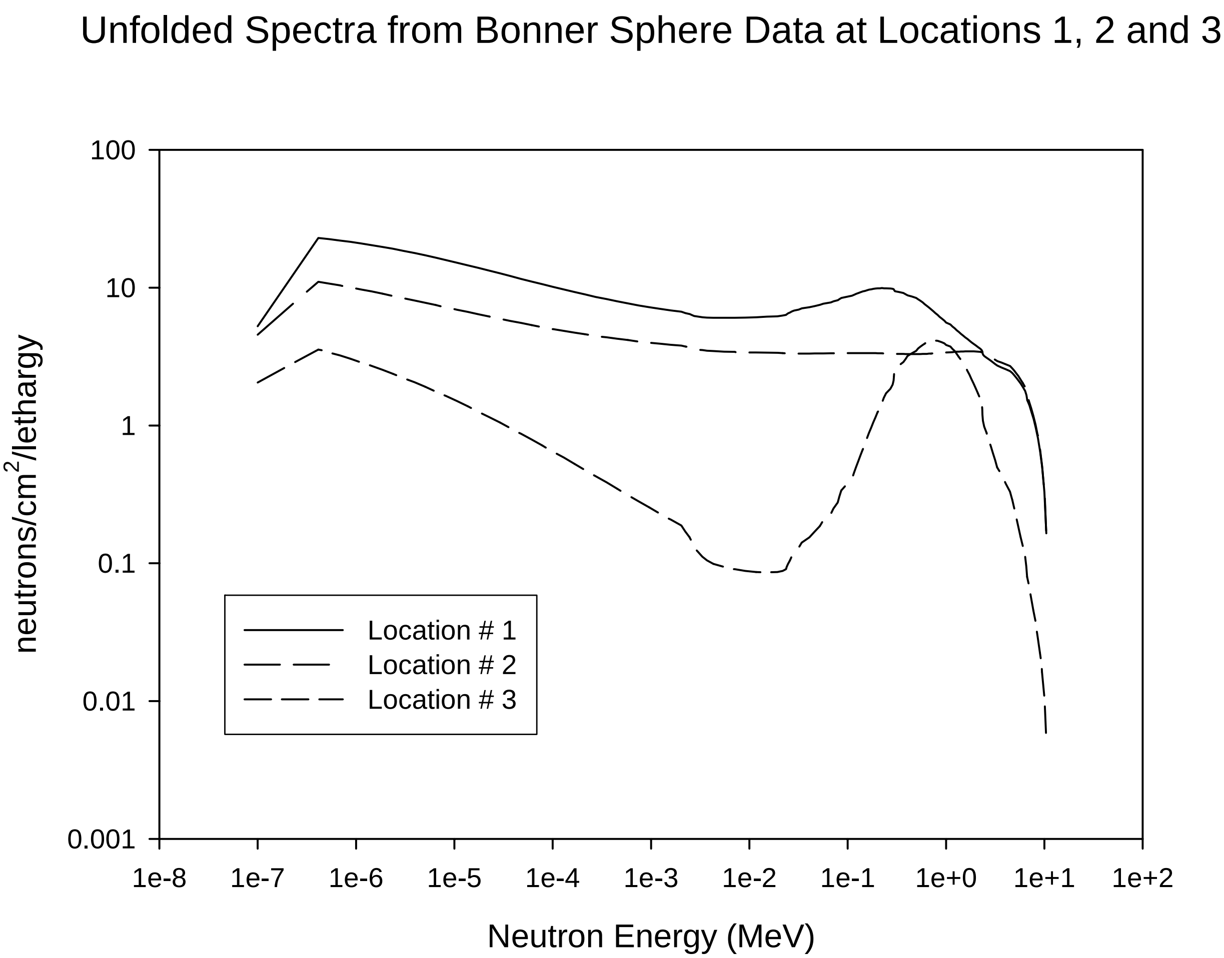

Figure 3 Unfolded Spectra for Three Measurement Locations

4.5 Calculation of Additional Shielding

As stated above, the radiation field around the self-shielded cyclotron is, at all points on the box, less than or equal to the box model specification. A conservative estimation of radiation field outside the shields at any distance can be made by treating the cyclotron as a combined gamma and neutron point source centered at the center of the cyclotron whose strength is as described above, and applying $1/r^2$ factors. The effect of shielding materials other than air can be estimated by making conservative estimates of tenth value layers (TVLs) knowing the spectra of gamma and neutron radiation and the effect of the material in question, and applying these attenuation factors on top of $1/r^2$ reduction in field. A model for simple 1-D radiation shielding calculations is shown below in Figure 4. It uses five of the most conventional shielding materials available for normal construction. This calculation model is provided only as a guide. It is the model used by Siemens Site Planning in preparation of site specific drawings. After installation minor modifications to the facility may be necessary to reach desired dose rates in surrounding areas, under the direction and at the discretion of facility radiation protection officials.

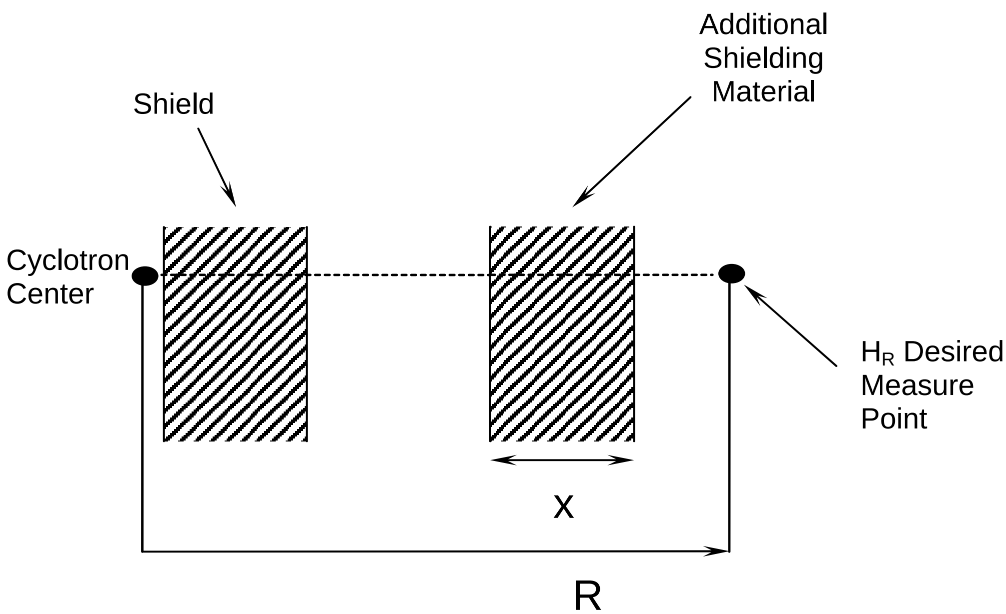

Figure 4 Model for 1-D Radiation Shielding Calculations

Formula

$$H_R = \left(\frac{100}{R}\right)^2 \left[H_n \times 10^{-x/TVL_n} + H_g \times 10^{-x/TVL_g} \right] \cdot \frac{I_{tot}}{120}$$

where:

R	Distance from center of cyclotron to point to be measured (cm)
x	Thickness of shielding material (cm)
H_n	Neutron dose rate at 1 meter from cyclotron center (89 μSv /hr)
H_g	Gamma dose rate at 1 meter from cyclotron center (205 μSv /hr)
H_R	The equivalent dose rate at the desired measurement point (μSv /hr)
$TVL_{(*)}$	The respective gamma or neutron tenth value layer for the associated source (cm)
I_{tot}	*Total possible beam current* *For an RD/RD, I_{tot} = 80 μA*

For an RD/HP, I_{tot} = 100 µA
For an HP/HP, I_{tot} = 120 µA

The TVL's for both gamma and neutrons are given in the table below.

Table 4 TLVs for Gamma and Neutrons

Material	Density (density $^g/_{cm^3}$)	TVL (cm) Gamma	TVL (cm) Neutron (#)
Concrete	2.35	38	43 (1)
Baryte	3.2	24	43 (1)
Lead	11.3	5	* % (7)
Polyethylene	0.97	80*	24 (3)
Iron	7.85	10	* % (7)

* Ineffective shielding medium for this radiation species.
% Useful as an energy moderator for this species.
(#) The number in parenthesis in the last column refers to the corresponding references in this document.
Gamma Mean Energy = 8 MeV
Neutron Mean Energy ≈ .15 MeV QF ≈ 8

A quality factor (QF) of 8 is used in computing neutron dose equivalence.

4.5.1 Additional Shielding – Downward Vector

In most cases, it is strongly recommended that Eclipse cyclotrons be located on the ground floor. It allows the use of soil as a shielding material for all downward directions. Radioisotope transfer lines are also shielded in this way by simply burying the conduit under a sufficient (45 cm) layer of earth and concrete.

However, Eclipse cyclotrons have been placed over accessible spaces, such as unoccupied equipment rooms. The tenth-value calculations above do not account for the lack of self-shielding provided in the downward direction, so alternate calculations are necessary. Monte Carlo calculations have been performed to asses shielding effectiveness from solid layers of normal concrete below the cyclotron.

The neutron and gamma ray source used in this study were derived using the ALICE-91 computer code from the Lawrence Livermore National Laboratory to determine the neutron and gamma ray spectra and intensity following the bombardment of ^{18}O with 10.5 MeV protons. To validate this source, a series of experiments were performed at Siemens (formerly as CTI Molecular Imaging) by removing the main shielding, operating the cyclotron at µA, and measuring the neutron and gamma dose rates at various distances and orientations. The ALICE source spectrum is consistent with the experimentally measured particle spectra and a neutron source intensity of 1×10^{10} neutron/sec/µA can be used for the intensity. Gamma source strength is much weaker and is unimportant in overall shield performance. The source strength was modeled as 60 µA.

The concrete modeled was “ordinary” concrete with a density of 2.35 grams/cm^3. Thickness of the concrete varied from 10 to 150 cm. The calculated neutron and photon dose rates are given in Table 5.

Table 5 Tenth-Value Calculated Neutron and Photon Dose Rates

Thickness (cm)	neutron (mrem/h)	photon (mR/h)	TVL - neutron (cm)	TVL - photon (cm)
10	3039	155	N/A	N/A
20	859	238	18.2	-53.7
30	285	200	20.9	132.4
40	109	127	24.0	50.7
50	39.4	79.4	22.6	49.0
60	15.6	48.8	24.9	47.3
70	6.01	27.7	24.1	40.7
80	2.36	15.7	24.6	40.6
90	0.861	8.8	22.8	39.8
100	0.371	4.9	27.4	39.3
110	0.159	2.7	27.2	38.6
120	0.0642	1.5	25.4	39.2
130	0.0319	0.9	32.9	45.1
140	0.013	0.5	25.7	39.2
150	0.00575	0.3	28.2	45.1

The TVLs for each additional 10 cm of concrete were calculated using the following formula:

$$TVL = \frac{23.02}{1n\left({D_1}/{D_2} \right)}$$

where:

TVL is in cm,
D_1 is the dose rate at the previous thickness, and
D_2 is the dose rate at the current thickness.

The data was plotted, as shown in Figure 5 below, then examined for exponential behavior by visual inspection and by fitting a line to the data. The neutron data shows a better fit, although the photon fit is also acceptable. TVLs were calculated for each adjacent pair of results beginning with the 20-cm thickness for neutrons, and 30-cm thickness for photons. An average was calculated using the 60- to 150-cm thicknesses for neutrons, and" 40- to 150-cm thicknesses for photons. The remaining values were conservatively rejected as outliers. The net effect was to increase the resulting average.

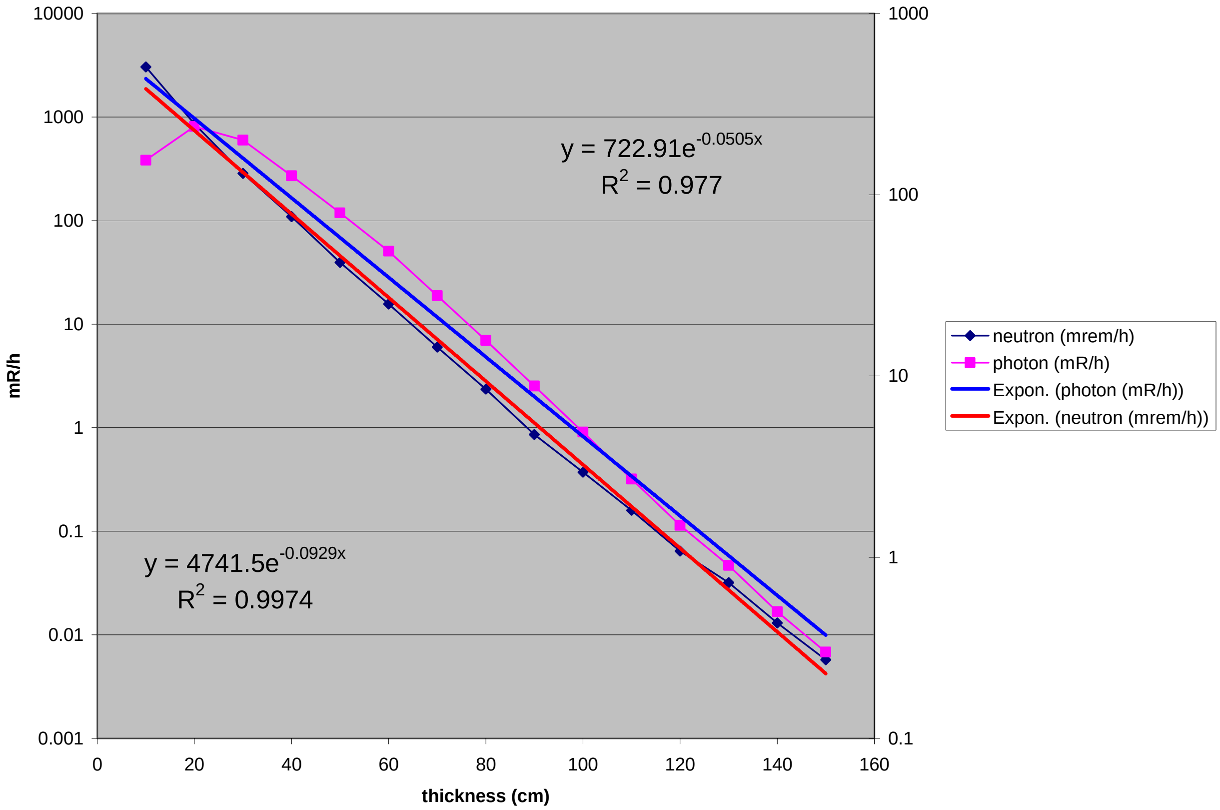

Figure 5 TVL Data Plotted

The resulting TVLs are, 26.32 ± 2.83 cm for neutrons, and 42.87 ± 4.33 cm for photons. These are close to the previously referenced values of 43 cm and 38 cm for neutron and photon, respectively. It should be noted that the tenth-value model is not appropriate until the floor thickness is at least 40 cm. Also note that a floor thickness of approximately 95 cm results in a combined dose rate from neutrons and gammas of approximately 2 mrem/hr.

4.6 Interlocks

The cyclotron and shield are interlocked so that one cannot accelerate and run beam on targets unless the two movable shielding blocks are closed and have actuated their respective limit switches. Testing, maintenance, etc., can be carried out simply and efficiently without having to “defeat” any radiation interlocks, thus the potential hazard of forgetting to restore the interlocks is avoided.

4.7 Monitoring Equipment

Personnel exposure monitoring equipment is not supplied as part of the Eclipse cyclotron. A dedicated area radiation monitoring system with appropriate set-points and audio and visual alarms may be installed in the cyclotron room at the discretion of the user or as mandated by in-house institution, local, or state regulations.

4.8 Residual Radiation

The residual radiation exposure level of the Eclipse accelerator itself varies from several times normal background, to several roentgens/hr in the vicinity of the targets. Servicing work on the targets, or in their

vicinity, should only be undertaken after a careful survey of the radiation fields, and an estimate of the radiation dose to be received by service personnel. While this topic is covered during the factory service training, the customer should have in place procedures governing maintenance work on the cyclotron.

4.8.1 Proton-Induced Activation

Radiation resulting from proton-induced activation has been minimized by design (i.e., careful attention to the vacuum system and its pumping configuration, plus careful selection of materials lying in the path of energetic protons produced by stripped, neutral beam, or due to beam loss on collimators).

The material used for intercepting stray beam is tantalum. Tantalum, or impurities present do activate under proton bombardment. Tungsten-181, which has a relatively low-energy X-ray emission of 58 keV, is the primary activation product. Tantalum 182 is the next most abundant, with Zinc-65, Cobalt-56, and Sodium-22 also seen in gamma ray spectroscopy of a target.

When the Eclipse is properly adjusted and operated, residual radiation levels that result from proton-induced activation inside the cyclotron are generally quite low; typically a few mrem/hr. Just after an irradiation, however, the graphite foil holder(s) on extractors may be activated with ^{13}N (10-minute half-life) because of the (p,n) reaction on 1% abundant ^{13}C in the graphite. There also may be ^{11}C (20.2 minute half-life) from the (p,pn) reaction on ^{12}C. The level of localized radiation will depend on the beam current during the run. After a high-current irradiation (e.g., 60 μA extracted), it is advisable to wait at least one hour before opening the accelerator for repairs.

Routine maintenance procedures, such as changing ion source cathodes and extractor foils, are accomplished by removing the ion source or extractor carousel from the cyclotron and replacing worn components at the workbench. In addition, any target can be removed or installed in less than one minute. This operation involves no break or reconnection of water lines.

4.8.2 Proton Induced Activation – Target Area - General

The Eclipse cyclotron can be provided with two beam lines positioned 180° apart. Each beam line utilizes a target changer that allows the installation of four to eight separate targets. The targets are modular in nature, and any combination of targets can be installed. Only one pair of water lines is needed to support the target changer, thereby reducing the complexity of the system and the likelihood of leaks. Moving any target into the beam line can be done automatically from outside the shield, and involves no breaking or making of connections to vacuum, water, or product lines. Target support units (TSUs) are also modular in nature.

In contrast with low activation levels inside the cyclotron, the residual activity in the targets resulting from irradiated target material, as well as proton activation of foils, can be quite high (refer to Appendices B and C of this document for details). Although unlikely to occur, under improper operating conditions, the target body might become activated. Servicing targets requires care and attention to instructions in order to minimize radiation exposure to maintenance personnel. The Eclipse target shield blocks should not be opened without a radiation survey meter in hand, particularly just after a target bombardment. It is also recommended that an alarming electronic dosimeter be worn by personnel during maintenance activities, with appropriate alarm levels set to reduce the potential for inadvertent exposure.

Specific procedures for servicing the targets are outlined in the pertinent instruction manuals. General procedures are outlined in Appendix A of this report.

4.8.3 Proton Induced Activation–Target Area–Immediately Post-Irradiation/Before Unloading

An assessment of the self-shield's effectiveness at reducing the external radiation field resulting from the target activity at end of bombardment has been accomplished. The maximum activity in the target at the end of a 150-minute bombardment at 60 µA of current onto 96 % enriched ^{18}O water would not exceed 4 Ci (148 GBq).

There are two sections to the self-shields (each of different composition) in the direct line outwards from the target to the surface of the shield. The composition of these shields is proprietary and will not be discussed in detail. The elemental composition of each section was modeled as a custom material in MicroShield® 6.01. The inner section is commonly called the "high Z insert," while the outer is referred to as "Shield Concrete." The "high Z" insert is 11.4 inches (29 cm), and the "Shield Concrete" is 19.5 inches (49.5 cm) thick respectively. This represents the shortest distance from the target to the surface of the shield.

Since the source is far from the dose point, it was modeled as a simple point source in MicroShield® 6.01. The buildup factor was calculated for the "high Z" section and represents the worst case scenario. The calculated dose rate on the surface of the shield is 7.0E-14 mrem/hr (7.0E-16 mSv/hr).

4.8.4 Proton Induced Activation–Target Area–Immediately Post-Irradiation/After Unloading

Contact dose rate on the targets for a target with approximately 2 Ci of ^{18}F has been measured at over 20 rem/hr Thirty minutes after bombardment, the contact dose rate on a target that has been loaded and unloaded properly remains 1.5 rem/hr. Contact with the back of the target by the survey meter allows the sensitive volume to within about 3 inches (7.6 cm) of the source, which in this case is the target window and beam strike area. Following the $1/r^2$ rule, the dose rate for the properly unloaded target drops to 300 mrem/hr at about 4 inches off of the rear surface.

Siemens offers optional copper shielding doors over the target area to provide local shielding to the most highly activated parts of the cyclotron (the targets) immediately after shutdown. With the copper doors in place the same dose rate (properly unloaded target 30 minutes after bombardment) drops from 300 mrem/hr to 7.5 mrem/hr, or a 0.025 attenuation factor from the copper doors. Similarly, after a 1-hr bombardment at 60 µA on silver ^{18}F targets with proper unloading, the activity exposure rate outside the copper doors is 22 mrem/hr on the front, 41 mrem/hr on the bottom, and 25 mrem/hr on the top. The first of these numbers can be compared to an estimated dose rate without copper doors in the same location of approximately 1 rem/hr, which gives an attenuation factor of 0.022.

While the installation of the copper doors over the target requires removal of some material from the inside surface of the shield, there is not a significant change to the radiation levels on the outside surfaces of the shield.

4.8.5 Neutron-Induced Activation

Neutrons are produced in targets as an unavoidable by-product of the nuclear reactions that produce the desired isotopes. Neutrons are also produced, albeit to a much lesser degree, from the reaction of neutral stripped beam on the tantalum neutral beam baffles.

Neutrons interact with the nuclei of atoms through elastic and inelastic scatter and are eventually absorbed through one of several possible nuclear reactions. Unlike the direct proton beam, which produces a nuclear reaction for only one particle in a thousand, every neutron eventually produces a nuclear transformation somewhere within the shield or within the components of the accelerator itself with the potential of producing a gradual build-up of activation products.

A summary of pertinent activation reactions is provided for reference in Appendix B. Neutron activation can result in a diffuse, low-level background radiation reading (a few mrem/hr) around the target area even after the targets are removed. The levels are generally small in comparison with the primary, proton-induced activation component within the targets themselves.

Neutrons give up their energy most readily through elastic collisions with the nuclei of hydrogen atoms losing, on the average, half their energy in each collision. After about 20 such "bounces" in the Eclipse shield a neutron reaches thermal equilibrium and is then absorbed or "captured." When a neutron is captured by hydrogen, an atom of stable deuterium is formed and energy is liberated in the form of a 2.2 MeV gamma ray. When a neutron is captured by heavier elements, such as iron or copper, gamma rays of much higher energy (up to 10 MeV) are produced, often in conjunction with a transformation into a radioactive isotope.

To appreciate the possible effects of these secondary neutron-induced reactions, consider a simple example of an ^{15}O target bombarded so as to reach a saturation activity of 1 Ci in the target. By definition, there are then 3.7×10^{10} neutrons/sec being produced, all of which must be absorbed within the shield or within the mass of the accelerator itself.

An important innovation in the design of the Eclipse shield is the use of polyethylene and boron-carbide-loaded concrete. The hydrogen content of this material is quite high, approximately 90% that of water by volume, resulting in the highest possible dose attenuation for neutrons. Boron has an extremely high thermal neutron capture cross-section while producing relatively low energy (478 keV) capture gamma rays and no residual activation product. The boron content of the polyethylene (content by weight) is such that it substantially reduces thermal neutron capture by the hydrogen in the shield blocks, by trace elements in the concrete, and by components in the accelerator itself, such as the iron in the magnet.

4.8.6 Residual Radiation – Waste Products

Refer to Appendix C for details on direct proton activation products. It is reasonable to assume from that data that no target windows (or other waste products) will need to be disposed of during commissioning, nor at the end of each production run. It is also reasonable to assume that all activation products except ^{56}Co can be eliminated through decay in storage over a two-month period.

Windows will need to be disposed of monthly during maintenance. Assuming a full 8-hr operation daily, the possibility of producing 0.8 mCi/day of ^{56}Co does exist. Over a month, one can assume 20 operating days. This would produce 16 µCi of ^{56}Co. The volume of the disposed material is negligible. The activation products remain in the Havar alloy matrix.

The targets are provided with hexagonal window support grids made of copper. The primary activation products of these assemblies are tabulated in Appendix C. The activity estimate presumes a 12% beam interception, which is conservative. Support grids will need to be disposed of annually during maintenance. Assuming a full 8-hr operation daily, it is possible to produce 0.14 µCi/day of ^{65}Zn. Over a year one, can assume 250 operating days. This would produce 35 mCi of ^{65}Zn. The volume of the disposed material is < 10 cc. The activation products remain in the copper metal.

Activation products exist in the target bodies, as well. However, the targets are designed such that the target material is intended to stop the beam. Therefore, any activation of the target bodies is minimized, and the total accumulated activity is on the order of a few mCi. The only activation product in the most commonly used target (Tantalum target body for production of ^{18}F) is W-181, with a 121 day half life.

4.8.7 Residual Radiation - Decommissioning

Quantitative estimation of the likely amounts of contamination is highly dependent on the modes of operation and the duration of service. Estimation of the costs associated with decommissioning is dependent on the above factors, plus local regulations and market pricing for disposal services. For this reason, an estimate of decommissioning costs cannot be offered. Because of the self-shielded design and minimal activation, many Siemens cyclotron customers have not had to secure any decommissioning bond whatsoever.

4.9 Target Material Hazards

Two liquid and three gas targets are available with the Eclipse. The two liquid targets are ^{18}O water (for the production of ^{18}F fluoride) and ^{16}O water for the production of ^{13}N ammonia). The three gas targets are ^{15}N (for the production of ^{15}O), ^{14}N (for the production of ^{11}C), and ^{18}O gas (for the production of $^{18}F_2$).

Depending on the particular product and bombardment conditions, as much as 4.0 Ci may be present in a target during and immediately after bombardment. All of the target bodies and collimator assemblies are water-cooled.

The targets are designed for computer-controlled remote loading and unloading using a push-gas or fluid. After irradiation, the target material is transferred through small-bored tubing to the appropriate shielded compartment (such as a mini hot cell) in the hot lab. After transfer to the hot lab, one of three options can be used to carry out further processing under automatic control, collect the material in a vial, or pipe the material through smallbore tubing to the scanner room.

Pressure in the gas and liquid targets is monitored by the Eclipse computer control system. Target pressure during irradiation is typically maintained in the range of 270-900 psi. The Havar target windows have been burst-tested at 1240 psi.

4.10 Radioactive Effluents

Most of the data on radioactive effluents from PET isotope production comes from the production of ^{18}F. During the production of ^{18}F, there is ^{13}N activity produced due to the ^{16}O impurity in the ^{18}O target material. The amount of ^{13}N produced depends strongly on the ^{18}O enrichment level. Normally this is not less than 90%, but lower enrichment recycled water is also used for rinsing targets and load lines. The presence of ^{13}N has been verified by collecting a sample of the vent gas and observation of a ten-minute half-life. The chemical form of the ^{13}N that is releasable is generally thought to be as N_2 gas because it has been observed to pass through the filtration system rapidly.

Very low level releases have also been observed during bombardment. These are thought to be ^{11}C from beam striking various graphite components inside of the tank.

The production of ^{11}C in-target will also create activation products but their half-lives are much shorter. Releases from ^{11}C production are generally assumed to be all ^{11}C. Very little neutron activation products (e.g. see Appendix D below) will be created as compared to the activity created during ^{18}F production as the neutron source term is orders of magnitude lower.

4.10.1 ^{18}F Overview

Overview of release points:

1) Sources of Effluents from FDG production
    a) Cyclotron -
        i) during target venting prior to unload (^{18}F & ^{13}N)
        ii) during target delivery (^{18}F & ^{13}N)
        iii) target failures (^{18}F & ^{13}N)
        iv) during single beamline operation (C-11)
    b) FDG Production -
        i) during chemistry - mostly during evaporation but also occurs elsewhere (^{18}F)
        ii) from nitrogen compounds bleeding out during chemistry (^{13}N)
    c) FDG QC
        i) Theoretical concern from a few regulatory jurisdictions (Washington)
2) Sources of Effluents from other chemical forms of ^{18}F (under development)

4.10.2 Common Control Methods

There are two basic approaches to controlling effluents. They are collection and filtration. Each has benefits and drawbacks but both will normally require radiation shielding to protect personnel working at the site, as well as careful placement to reduce potential for increased public radiation exposure and to ensure ease of access for periodic maintenance.

4.10.3 Cyclotron and Related Sources of Effluents

Bags are commonly used on the Target Support Unit (TSU) vent lines to capture effluents originating from the venting of the target (*1-a-i above*) prior to unloading. Several configurations have been tested including 3 liter bags made out of rubber and 10 liter Tedlar bags. In either case bags were hooked up to the vent line in the TSU. The bag is a source of radiation and must be shielded. Moreover, they are connected to a vent port inside the shields, so the bags are usually placed inside the cyclotron shields and are subject to intense bombardment from neutrons. This degrades rubber bags rapidly leading to leaks. For this reason, Tedlar is a preferred material. Total volume is very small so a once per week timeline for emptying the bag is adequate. The primary radionuclide captured is ^{13}N in the form of N_2 gas.

If the target is not vented prior to delivery of the isotope, effluents during target delivery (*1-a-ii above*) are collected at the point of delivery. This may be on the outlet of an anion exchange column or on the vent outlet of a delivery vial. These effluents can also be captured with 10 liter Tedlar gas bags which are typically located in the hot cell or mini cells. These bags are a source of radiation and must be shielded. Placement in the hot cell can cause higher hand exposures for personnel removing doses after dispensing.

Collecting all of the effluents from a target failure (*1-a-iii above*) is difficult as some of the material that escapes from the vacuum system will be in a chemical form that is difficult to filter using HEPA and carbon, but the ^{18}F that is released should be removable by filtration. Previously this was considered an insignificant route of release since target failures after the first ten to twenty minutes of bombardment are relatively rare. While infrequent, the consequences of target loss near the end of bombardment are high due to the much higher activity present in the target.

The use of a KEP3S filtration unit with HEPA and carbon cells, properly sized for air flow, will greatly reduce any activity that could escape from this route. If the KEP3S is only on the cyclotron exhaust, then it may be possible to eliminate the radiation shield from around the filter depending on placement as the occurrence of a full target failure is low. See Section 5.0, Accident Scenarios and Analysis for more information.

It has also been noted that low level releases occur during cyclotron operation, i.e. before target venting or unloading (*1-a-iv*). It is thought that these emissions are ^{11}C in the form of CO_2 or CO, and originate from ^{12}C (p,pn)^{11}C reactions on graphite used inside the cyclotron at various locations. The proton beam will graze these pieces and, when heated due to this, give off the gas. This effluent will not easily be collected, has a very low impact on the dose due to effluents, and will not be considered further.

4.10.4 Effluents from FDG Production

The primary source of effluents is during the synthesis of FDG in the chemical synthesis units (*2-b-i above*). We have found that both HEPA and carbon filters are useful for controlling this effluent. It appears that some material that is released, most often when there is a major chemistry failure, is in an aerosol form that can be collected using HEPA filters. Normally most of the effluent is in a gaseous phase that can be collected with carbon. A typical embodiment is the B&S KEP3S filter unit in the smallest cell size. If the cyclotron effluent is to be included along with the hot cells the next larger cell size is typically used. Based on an experiment, we have determined that the unfiltered release fraction averages 1.7% of the activity delivered to the chemistry processing unit. Using the small KEP3S filter, we have seen a collection efficiency of 99.83%.

The standard filter arrangement for both sizes of the KEP3S filter is as follows:

- One pre-filter (reduces dirt and dust loading of HEPA and extends its life)
- One HEPA cell (high collection efficiency for aerosols)
- Two carbon cells (loaded with either 209C KINA or TEDA impregnated carbon)

4.11 Air Activation Hazard

The 11 MeV protons from the Eclipse are absorbed entirely inside the targets and cannot escape to activate the air inside the shield or inside the room.

The third most abundant isotope in the atmosphere is argon at a little less than 1 percent. ^{41}Ar is created in the air inside of the cyclotron shields due to neutron activation. There is also some argon gas used to purge various components of the cyclotron. This radionuclide has a half-life of 109 minutes and emits a 1200 keV beta and a 1293 keV gamma with a yield of 99% for both. It is estimated that approximately one microcurie of ^{41}Ar is released per run. Releases from this isotope will not be detectable with coincidence counting techniques (Thermo monitor), but could be detectable with systems that are based on positron or gamma detection (Canberra & Laboratory Impex). ^{41}Ar has very little impact on public radiation exposure calculations due to it being a noble gas.

Secondary gamma or neutron-induced air activation reactions on some other gases are theoretically possible, albeit at extremely low levels. A summary of such reactions is given in Appendix D.

In most cases, the reaction threshold energy is higher than the maximum neutron or gamma ray energy from the Eclipse targets. In cases where a reaction is theoretically possible, the cross-section is too low or the product too long-lived to allow any appreciable buildup of activated air inside the Eclipse shield during the course of a day's operation, even assuming the absence of airflow.

In actual operation, the shield is closed around the targets and the interior is maintained at a negative pressure so that air is drawn in through cracks between shield blocks and then out the exhaust at approximately 500 ft^3/min minute. Opening the shield will, of course, dilute whatever minute concentration may be present into the larger room volume.

Another potential "activation" product around the Eclipse targets is ozone. However, during many years of operating experience with the Eclipse, no measurable amount of ozone has been detected around the targets as a result of secondary gamma or neutron radiation.

5.0 Accident Scenarios and Analysis

This section estimates the magnitude of an uncontrolled release to the environment resulting from target failure, based on physical and environmental factors (target volumes, etc.) and on actual operational experience with the Eclipse. The Risk Estimates of this document contains a detailed example calculating the consequences of releasing the contents of fully loaded target into the cyclotron room or into the outside environment.

5.1 Case 1: A Gas Target Window Rupture with One Curie of Activity in the Target

The contents of the target (150 cc at s.t.p.) are pulled into the cyclotron vacuum tank and foreline. The diffusion pump power is cut upon an abrupt rise in pressure. Some small amount of the activity is exhausted by the roughing pump and will go into the customer supplied ventilation system. The majority of the activity remains trapped inside the cyclotron vacuum tank, roughing pump and diffusion pumps. This is true regardless of the isotope and chemical form. This activity should be allowed to decay for several half-lives before the accelerator is accessed by personnel.

Relevant data has been obtained through actual experience in our laboratory. In one instance, a window failed in a ^{11}C target in which the helium cooling had been incorrectly plumbed. After approximately 30 minutes of running at 40 μA, the window ruptured and vented the contents of the target (approximately one Ci) into the cyclotron vacuum tank. The vacuum gauge interlocks immediately tripped, shutting down the accelerator and cooling off the diffusion pumps, all in accord with the above scenario.

After 30 minutes, the tank was opened and a survey conducted. There was a residual field of approximately 60 mrem/hr at the center of the cyclotron, presumably resulting from absorbed [^{11}C] CO_2 on metal surfaces. The survey reading decayed with a 20-minute half-life associated with ^{11}C, indicating that the activity was sequestered and not leaving, even though the tank was open.

The oil reservoirs on the diffusion pumps did not register on the survey meter, but the mechanical foreline pump measured approximately 1 rem/hr at the surface 50 minutes post rupture. There was no indication of any activity escaping from the outlet of the mechanical pump.

The survey readings at the indicated locations and times post-rupture suggest that approximately 90% of the activity was trapped in the mechanical pump, while 10% of the activity remained in the vacuum tank.

During the ensuing 30 to 40 minutes, minor repairs were completed (e.g., the cyclotron-extractor foils replaced, etc.), and the tank vacuum restored. Approximately five half-lives had then elapsed since target failure. The residual survey meter reading at the mechanical pump immediately dropped, indicating that substantial flow was required to purge activity from the vacuum system.

This experience demonstrated that if a window ruptures with a fully-charged ^{11}C gas target, activity remains mostly sequestered inside the tank and inside the mechanical foreline pump. This is true because of the containment capability inherent in the system, and assumes that correct procedures are followed (i.e., the operator waits for the activity to decay). The end result is a negligible release of activity into the environment.

5.2 Case 2: ^{18}O Water Target Failure Possibilities

The target loading is approximately 2.5 ccs of enriched ^{18}O water. The water target operates under high pressure (600 psi above atmosphere) and is extremely robust. Windows rupture infrequently in normal operation, but are more likely under conditions of impaired target body cooling or absence of helium window cooling. In some cases, the target system can have a rupture in overpressure lines and spill or spray the 2.5 cc load inside the target shield. This will lead to high levels of contamination inside of the shields. Proper Personal Protective Equipment (PPE), such as double gloves, sleeve covers, head covers, and booties should be worn if opening the shields to investigate situations involving a sudden loss of

target pressure during operation. Radiation field surveys should also be performed as the shields are opened in order to ascertain the hazard level.

During bombardment of ^{18}O water, some ^{13}N in a gaseous form will also be produced as a result of the ^{16}O impurity present in the target. As a result, a release of gaseous ^{13}N into the ventilation system will occur when the target is ruptured. The amount released will depend on the target ^{18}O enrichment level, as well as the length of bombardment that has occurred.

Any spilled material would remain inside the shield and orderly cleanup could commence after the appropriate wait for isotope decay. Activity trapped inside of the vacuum tank will be relatively well shielded. It is recommended that the material be allowed to decay prior to opening the tank for maintenance.

5.3 Case 3: N-13 Target Failure Possibilities

This target operates at high pressure (200-400 psi) and has been failure tested on an operating Eclipse. The ammonia target is similar to the ^{18}O water target above in that most of spilled material will remain inside the shield, and orderly cleanup could commence after the appropriate wait for isotope decay. There will be some ^{13}N in a gaseous form that will escape into the cyclotron ventilation system as described above for the ^{18}O water target.

5.4 Case 4: $^{18}F_2$ Target Failure Possibilities

This target operates at 900 psi pressure. In case of a rupture, some of the ^{18}F- activity will escape into the ventilation system. Because of the chemically reactive nature of $^{18}F_2$, it will tend to adhere to the inner surfaces of the ventilation system. If this occurs near an effluent monitor, a false release accumulation would result. A properly designed carbon filter system is effective in reducing the amount of activity released.

5.5 Case 5: Leak of Activity into the Cyclotron Room

In this section, we shall assume the worst case: a full target load of gaseous activity is vented into the cyclotron room or into the environment, and then the maximum exposure for personnel coming in contact with such a release is calculated.

A release of activity into the room or into the environment is possible in the event that a target delivery line ruptures or a plumbing fitting fails under pressure. These hazards are generally of importance only in the case of gaseous [^{11}C]CO_2, [^{15}O]O_2, or [^{18}F]F_2 targets. The activity from liquid targets [^{13}N]NH_4 and [^{18}F]F- is not dispersed in the event of target failure and is readily cleaned up following appropriate decontamination procedures.

The following calculation estimates the maximum external exposure dose rate for a person entering the cyclotron room immediately after the postulated rupture. The maximum rate will decline with isotope decay and will also drop rapidly as the room air is mixed and exhausted by the ventilation system. Assume the smallest practical room dimensions for installing an Eclipse:

$$V = 7m \; * \; 6.9m \; * \; 3m = 144 \; m^3$$

Assume a uniformly dispersed gaseous release of one Curie into the room:

$$\text{activity concentration} = {}^{1}\!/_{V} \; = \; .00694 \; Ci/m^3$$

Approximate the room volume by a sphere of radius R = 3.25 m. For a point *r* meters from a point source of activity, the dose rate in roentgens per hour is:

$$D = 0.53 \ CE/r^2$$

where:

C is the source activity in Ci,
E is the energy/decay in MeV = 1.022 for positron emitters.

The dose rate at the center of a sphere of radius R is:

$$W_o = 0.53\frac{C}{V}\int_o^R 4\prod dr$$

$$= 150 \text{ mrem/hr}$$

Assuming the room is ventilated sufficiently to produce 10% air turnover/min (500 ft^3/min in a 5000 ft^3 room) then the activity concentration and dose rate will diminish exponentially:

$$W = W_o \ \exp^{-(6+\Lambda)t}$$

where:

t is expressed in hrs.

The effective decay constant is the sum of the air turnover constant (1/.1667=6) plus the isotope decay constant, λ.

The integrated external gamma exposure, assuming one stays in the room indefinitely and including isotope decay for the various positron emitters, is:

$$W \begin{array}{l} = \mathrm{W}_0/26.8 = 5.6 \text{ mrem for Oxygen-15} \\ = \mathrm{W}_0/10.16 = 14.76 \text{ mrem for Nitrogen-13} \\ = \mathrm{W}_0/8.1 = 18.5 \text{ mrem for Carbon-11} \\ = \mathrm{W}_0/6.38 = 23.5 \text{ mrem for Fluorine-18} \end{array}$$

For a person immersed in a gaseous cloud of positron emitting radioisotope, the beta dose to the exposed surface of the skin can be inferred from ratios of skin dose to whole-body dose calculated in ref. 18, assuming one stays in place indefinitely.

Interpolating in Table 3 through Tables 6 in Reference 18 of this document to find the respective ratios, and assuming a spherical cloud of 3.25 meters radius, the integrated beta exposure of the skin are:

330 mR for Oxygen-15
546 mR for Fluorine-18.

The exposure resulting from inhaled activity for someone entering the room and staying indefinitely can be estimated from the known initial air concentration and assuming a breathing rate (highly conservative) of 20 liters/min for "reference man engaged in light activity" (see reference 17).

The time integral of activity concentration, assuming 10% air turnover/min and isotope decay, as above is:

$$.00649 \ \mathrm{Ci}/\mathrm{m}^3 \times 564 \text{ seconds} = 3.9 \ \mathrm{CiSec}/\mathrm{m}^3 \text{ for F-18}$$

Assuming a breathing rate of 20 liters/minute, a concentration of 1 Ci sec/cu meter results in .334 mCi inhaled. Thus, the total inhaled activity is:

.334 x 3.9 = 1.3 mCi for F-18

.334 x 3.1 = 1.03 mCi for C-11

.334 x 2.46 = .82 mCi for N-13

.334 x .93 = .31 mCi for O-15

The internal absorbed dose factors are estimated for ^{11}C and ^{15}O assuming that O_2, CO_2, CO, and HCN are 100% absorbed by the lungs, go directly into the blood, and stay there. The critical organ then is the blood with total dose factors of 89 mrad/mCi for ^{11}C and 10 mrad/mCi for ^{15}O. The corresponding whole-body dose factors are 13 mrad/mCi for ^{11}C and 1.4 mrad/mCi for ^{15}O.

The dose factors for ^{18}F as fluoride (which is the form that ^{18}F will take in the blood) is obtained from (ref. 16). The critical organ is bone, with a total dose factor of 200 mrad/mCi while the whole body dose is 73 mrad/mCi. Activity from the standard ^{13}N ammonia target cannot be dispersed in gaseous form. However, data for gaseous ^{13}N ammonia is included for completeness. The data from Lockwood (ref. 16) shows that the urinary bladder wall is the critical organ for ^{13}N ammonia uniformly distributed in the blood. The critical organ dose is 51 mrad/mCi and the whole body factor is 5.5 mrad/mCi. Internal absorbed doses are summarized in Table 1.

5.6 Case 6: Leak of Activity into the Cyclotron Ventilation System

A second circumstance to consider is the release of activity to the environment through the ventilation system. As before, we shall estimate the external exposure dose as a result of the gamma ray component as well as the dose due to inhaled activity.

A simple Gaussian plume atmospheric dispersion model will be used for this example. More complex scenarios such as multiple release points, use of meteorological data, and the complex modeling necessary for urban areas with high-rise buildings are outside the scope. The following comes directly from NCRP 123, “Screening Models for Releases of Radionuclides to Atmosphere, Surface Water and Ground.”

2.2.1 Isolated Point Source, No Wake Effects, H.2.5 h_b

This condition represents the case of the isolated point source, because the source is well above the perturbed flow around the neighboring buildings (Wilson and Britter, 1982). The values of C should be calculated by

$$C = \frac{fQP}{u} \tag{2.6}$$

where:

P= the Gaussian diffusion factor appropriate for the values *H* and *x* begins considered in (m^{-2})

Values of *P* as a function of downwind distance for various values of *H* are presented in Figure 2.2 [not shown in this document]. This figure was constructed using the 22.5° (16 point wind rose) sector-averaged form of the Gaussian plume model as represented by Equation 2.7 [below] (Fields and Miller 1980); *i.e.*

$$P = \frac{2.032}{x\sigma_z} \exp\left[-\frac{1}{2} \left(\frac{H}{\sigma_z} \right)^2 \right] \tag{2.7}$$

The constant 2.032 results from the crosswind integration of the basic Gaussian diffusion equation and replacing σ_y by one-sixteenth of the circumference of a circle of radius x: $\left(2\pi x 16^{-1}\right)$ [see, for example, Slade (1968)].

Note in Figure 2.2 [not shown in this document], however, that *P* is constant between 100 m and the distance where *P* is maximum for *H*. If the screening test cannot be passed using Equation 2.6, the use of a Gaussian-plume, wind-rose model incorporating a site-specific join frequency distribution of wind speed, wind direction, and atmospheric stability should be considered, *e.g.*, Moore, *et al*. (1979).

5.6.1 Example Daily Release

4Ci (148GBq) of ^{18}F is produced within a single production run. Assume the worst-case routine release of 1% of activity post filter.

where:

H = 11m
H_b = 4m
u = 3.65m/s
x = 50m
f = 0.25

For a single release of 40mCi (1.48GBq) of ^{18}F, 1.48 GBq is released over 30 minutes.

$$Q = 8.22 \times 10^{5} \text{ Bq/s}$$

$$\sigma_z = (0.06x) \big/ \left(\sqrt{1+0.00015x}\right)$$

$$\sigma_z = 2.989 \text{ m}$$

$$P = (2.032/x\sigma_z) \exp[- ½ (H/\sigma_z)^2]$$

$$P = 1.556 \times 10^{-5} \text{ m}^{-2}$$

$$C = (f * Q * P)/u$$

$$C = 0.876 \text{ Bq/m}^3$$

Exposure to an individual:

In NCRP123 Appendix B, Atmosphere Screening Factors, Table B.1 shows screening factors (SF) for the six pathways described for each parent radionuclide and each daughter contributing more than 10 percent of the dose of the parent. The screening factor listed for F-18 is 4.1E-06 Sv per Bq/m^3.

Multiplying the screening factor for F-18 and the average atmospheric concentration at the receptor together gives the dose to the receptor.

Dose to receptor = SF * C

Dose to receptor = 3.59 μSv (0.359 mrem)

Table 6 shows the estimated exposure to personnel resulting from activity vented into the cyclotron room, assuming:

- 1 Ci uniformly dispersed in 144-m^3 room volume.
- 500 ft^3/min ventilation exhaust rate.
- Person stays in the room indefinitely.

Table 6 Personnel Exposure from Activity Vented into the Cyclotron Room

Isotope (mrem)	Dose from External Activity (mrem)	Inhaled Activity (mCI)	Critical Organ Dose Factor (mr/mCI)	Dose to Critical Organ (mr)	Whole Body Dose Factor (mr/mCI)	Whole Body Dose (mr)
^{15}O	5.8	.31	10 Blood	3.1	1.4	.43
^{13}N	10.2	.82	51 Bladder	41.8	5.5	4.5
^{11}C	18.5	1.03	89 Blood	91.7	13	13.4
^{18}F	23.5	1.3	200 Skeleton	260	73	95

Table 7 shows the estimated exposure to personnel downwind resulting from activity vented in the exhaust stack and into the environment, assuming:

- 1 Ci vented in a single "puff."
- 500 ft^3/min ventilation flow rate.
- Stack height 25 m; stack diameter 1 ft.

Table 7 Personnel Exposure from Exhaust Stack

Isotope (mrem)	Dose from External Activity (mrem)	Inhaled Activity (mCI)	Critical Organ Dose Factor (mr/mCI)	Dose to Critical Organ (mr)	Whole Body Dose Factor (mr/mCI)	Whole Body Dose (mr)
^{15}O	<.080	<.00024	10 Blood	<.0024	1.4	<.00033
^{13}N	.080	.00024	51 Bladder	.0122	5.5	.00132
^{11}C	.080	.00024	89 Blood	.021	13	.0031
^{18}F	.080	.00024	200 Skeleton	.048	73	.0175

APPENDIX A: Target Removal

In addition to a film badge, which must be worn at all times while in the Eclipse room, a direct-reading dosimeter, disposable gloves, and a lab coat must always be worn during target removal and service. After opening the Eclipse shield, a survey of radiation levels must be taken of the area around the targets before attempting to remove any one of them. The magnet should be off to prevent damage to equipment or to persons from flying tools, and to prevent false readings on the radiation survey meter due to the fringe field. The exposure dose rate in roentgens/hr at a distance of one foot from the source and for a given quantity of radioactive material in the target is given by:

$$R = 5.7\ CE$$

where:

C = Activity in Ci
E = Energy in MeV/radioactive disintegration

which, in the case of positron-emitting isotopes is close to 2 times .511 MeV, or 1.022 MeV.

Thus, one Curie in the target will produce an exposure dose rate of approximately 5.7 roentgens/hr 1 foot away from the target. This does not take into account target body activation, which will vary with duration and frequency of target use.

Before a target body is removed from its beamline, the target material should be allowed to decay to a safe level (for a short lived gas such as ^{15}O), or the target should be emptied of radioactive material in the case (for a relatively long-lived product) such as ^{18}O water. The material can be unloaded and stored in a vial in a shielded enclosure. You should not open the shield or enter the shielded area until the radiation field has decayed to a safe level. Consult your local radiation safety official for radiation limits at your facility.

In the event of suspected leakage of a target, the target should be secured as quickly and as completely as possible. It should be placed in a non-spill receptacle and put into a shielded and vented hot cell or fume hood. Personnel coming in contact with this target must be checked for contamination before leaving the lab area. In the event of spill or contamination, the radiation safety officer must be notified and the area roped off and posted against unauthorized entry.

In all cases, proper operation of the Eclipse exhaust and the room exhaust should be verified. After completion of maintenance procedures, check personnel and their clothing for contamination before they leave the lab area.

APPENDIX B: Secondary Neutron Activation Products

Table 8 lists a number of pertinent neutron activation reactions (ref. 1). These are listed primarily for reference. In actual day-to-day operation the effects of neutron activation are negligible in comparison with direct proton activation of target components.

Table 8 Neutron Activation Reactions

Element	Component	Reaction	Product	Half-Life	Gamma Energy MeV
Copper	Magnet Coils	^{63}Cu (n, a)	Cobalt 60	5.27 yr	1.15. 1.33
Copper	Magnet Coils	^{63}Cu (n, g)	Copper 64	12 hr	8β
Iron	Magnet Yoke	^{54}Fe (n, p)	Manganese 54	312 day	.835
Iron	Magnet Yoke	^{56}Fe (n, a)	Manganese 56	2.58 hr	.847
Aluminum	Vacuum Tank Concrete	^{27}Al (n, a)	Sodium 24	15 hr	2.75
Aluminum	Vacuum Tank Concrete	^{27}Al (n, p)	Magnesium 27	9.5 month	.844
Silicon	Concrete	^{28}Si (n, p)	Aluminum 28	2.25 month	1.78
Sodium	Concrete	^{23}Na (n, g)	Sodium 24	15 hr	2.75
Antimony	Lead Shielding	^{121}Sb (n, g)	Antimony 122	2.7 day	.56
Antimony	Lead Shielding	^{123}Sb (n, g)	Antimony 124	60 day	.6, .7, 1.7

APPENDIX C: Proton Activation of Foils and Grids

Activation of cyclotron components from direct proton irradiation is unavoidable in the design of a proton accelerator. Activation risk has been minimized in the Eclipse through the use of tantalum, aluminum and carbon, which do not have significant long-lived activation products at 11 MeV. However materials used for the target windows, window support grids, and target bodies cannot always be made of these materials. The most important activation products, from the perspective of radiation protection, are listed below for Havar, Beryllium Copper, and Titanium foils and Copper support grids. Products with half-lives ranging from a few seconds to a few minutes are not included. Saturation activities for the various product isotopes are calculated. Cross sections at 11 MeV were derived from the CSISRS database~. Gamma ray energies and intensities were derived from several sources [9,10]. Activity at end of bombardment assumes 60 µA for 120 minutes.

For a given product, the activity A(t) at any time t after start of bombardment is given by:

$$A(t) = A_{sat}\left(1 - .5^{t/t_{half}}\right)$$

where:

> A_{sat} is the activity at saturation at the given beam current, and
> t_{half} is the half life in the same units as t.

For products with relatively short half-lives, such as Mn-52m and Cu-62, saturation is approached during a single bombardment of an hour or more, and the activity has completely decayed away at the start of bombardment the next day. For the longer-lived products, such as Tc-95g, saturation is not approached during bombardment, and the calculation can be done in terms of production rate, or mCi/µA per hr. The longer-lived isotopes do not represent a significant portion of the activity initially produced, but 24 hours later have not decayed significantly. Therefore they are the predominant contributors to long-term risk and waste. The calculation has been performed for the quantity produced in 1.67 µA-hrs, or a single bombardment of 60 µA and 2 hours. For the long-lived isotopes (multiple days half-life) an estimate of the annual amount produced is simply the production amount per bombardment multiplied by the number of runs anticipated in a year.

The calculations for 0.001" thick Havar are listed in Table 9. By far Mn-52m is the largest short lived activation product. Dose at 1 foot from a Havar foil immediately after a 60 µA, 120-minute bombardment can be estimated at >2 rem/hr. However, 3 hours later the activity in the foil has dropped to approximately 2 mCi, or a fraction of a percent of the initial activity. Fields one foot from the foil, unshielded, should be between 10 and 20 mrem/hr. These calculations are offered as a guideline. Safe working conditions are determined by your local radiation safety official. Moreover, no service should be attempted unless the area is surveyed and radiation fields are below a safe working level.

Havar														
Element	Naturally-Occurring Isotope		Weighted Amount (%)	Activation Reaction	Activation Product	Half Life	Half-life in hours	Cross Section (mb)	Decay Mode	TTY (mCi/uA hr, 25 u of Havar)	TTY (mCi/uA at sat, 25 u of Havar)	activity produced per run (mCi)	activity 3 hours later (mCi)	activity 24 hours later (mCi)
Cobalt	59	100%	42.00%	(p,n)	Ni-59	8e4 y	7.01E+08	400		2.08E-08		0.0	0.00	0.000
42%														
Chromium	50	4%	0.80%	NA	NA	NA	NA	NA						
20%	52	84%	16.80%	(p,n)	**Mn-52m**	21 m	3.50E-01	370	EC, Beta		5.74E+00	**337.8**	**0.89**	0.000
	52	84%	16.80%	(p,n)	**Mn-52g**	5.6 d	1.34E+02	80	EC, Beta	7.24E-03		**0.9**	**0.86**	**0.768**
	53	10%	2.00%	(p,n)	Mn-53	2e6 y	1.75E+10	200	NA					
	54	2%	0.40%	(p,n)	Mn-54	312 d	7.49E+03	100	EC	3.13E-05		0.0	0.00	**0.004**
Iron	54	6%	1.08%	NA	NA	NA	NA	NA						
18%	56	91%	16.38%	(p,n)	**Co-56**	77 d	1.85E+03	450	EC, Beta	1.75E-03		**0.2**	**0.21**	**0.208**
	57	2%	0.36%	(p,n)	Co-57	272 d	6.53E+03	400	EC	2.16E-05		0.0	0.00	**0.003**
Nickel	58	68%	8.84%	(p,a)	**Co-55**	17.5 h	1.75E+01	11	EC, Beta	4.23E-03		**0.5**	**0.45**	**0.196**
13%	60	26%	3.38%	(p,a)	Co-57	272 d	6.53E+03	11	EC	9.77E-06		0.0	0.00	**0.001**
	62	4%	0.52%	(p,n)	**Cu-62**	10 m	1.67E-01	770	Beta		3.10E-01	**18.6**	0.00	0.000
Tungsten	182	27%	0.81%	(p,n)	Re-182	2.3 d	5.52E+01	NA	EC					
3%	183	14%	0.42%	(p,n)	Re-183	70 d	1.68E+03	NA	EC					
	184	31%	0.93%	(p,n)	Re-184	38 d	9.12E+02	NA	EC					
	186	28%	0.84%	(p,n)	Re-186	3.8 d	9.12E+01	NA	Beta					
Manganese	55	100%	2.00%	(p,n)	Fe-55	2.7 y	2.37E+04	NA	EC					
2%														
Molybdenum	92	15%	0.30%	NA	NA	NA	NA	NA	NA					
2%	94	9%	0.18%	(p,n)	**Tc-94m**	53 m	8.83E-01	430	EC, Beta		3.59E-02	**1.7**	**0.16**	0.000
	94	9%	0.18%	(p,n)	Tc-94g	293 m	4.88E+00	90	EC, Beta		7.91E-03	**0.1**	**0.08**	**0.004**
	95	16%	0.32%	(p,n)	Tc-95m	61 d	1.46E+03	280	EC	1.97E-05		0.0	0.00	**0.002**
	95	16%	0.32%	(p,n)	**Tc-95g**	20 h	2.00E+01	700	EC		9.87E-02	**0.4**	**0.36**	**0.173**
	96	17%	0.34%	(p,n)	**Tc-96m**	52 m	8.67E-01	1000	EC		1.55E-01	**7.4**	**0.67**	0.000
	97	10%	0.20%	(p,n)	Tc-97	3e6 y	2.63E+10	NA	NA					
	98	24%	0.48%	(p,n)	Tc-98	4e6 y	3.50E+10	NA	NA					
	100	10%	0.20%	NA	NA	17 s	4.72E-03	NA	NA					
Total			99.58%									**367.7**	**3.68**	**1.358**

Table 9 Havar activation products

The largest contributor to long-lived activation products in Havar foils is Mn-52g. After 100 bombardments of 2-hr duration, significant fractions of a mCi are formed of all the isotopes that correspond to bold numbers in the far right hand column. Activity accumulation can be calculated, using the partial Thick Target Yield (TTY) from the table. An example is shown below.

EXAMPLE: Estimate the amount of Cobalt-56 activity present in a .001” Havar foil which has been used in a target for four weeks at 13 hrs/week on the average, at a beam current during bombardment of 60 µA.

SOLUTION: The yield from the table is 1.75×10^{-3} mCi/µA hr. Four weeks of 13 hrs/week bombardments gives 52 hrs of beam-on time, at 60 µA.

The amount of Cobalt-56 activity at the end of four weeks is given by:

$$A = 52hr \cdot 60\mu A \cdot 1.75 \times 10^{-3}\ {}^{mCi}/_{\mu Ahr} = 5.46 mCi$$

Because the half-life of ^{56}Co is 77 days, decay of the isotope and saturation effects can be neglected (agreement with the exact calculation is within 1.3%).

A similar calculation (see Table 10) has been performed for 0.002” thick 98% copper/2% beryllium foils, such as are used in the ^{11}C, ^{15}O and F_2 gas targets for the Eclipse HP. Copper-63, which has 69% abundance in nature, undergoes a p,n reaction to form ^{63}Zn. Copious amounts (>830 mCi after a 1-hr, 60 µA bombardment) of ^{63}Zn are produced, with a 38.4 min half life. After 6 hours, the activity is calculated to be 1.25 mCi, which will result in a field of several mrem/hr at one foot. Even 3 hours after bombardment, this field is likely to be 25 times higher. It is recommended to wait 6 hours after bombardment before handling Beryllium/Copper foils. The predominant long-lived activation product of copper bombardment is ^{65}Zn, with a 243.6 day half-life. Production of ^{65}Zn from one hundred bombardments is on the order of 2 mCi.

Table 10 Beryllium/Copper Activation Products

Beryllium Copper														
Element	Naturally-Occurring Isotope		Weighted Amount (%)	Activation Reaction	Activation Product	Half Life	Half-life in hours	Cross Section (mb)	Decay Mode	TTY (mCi/uA hr, 50 u of BeCu)	TTY (mCi/uA at sat, 50 u of BeCu)	activity produced per run (mCi)	activity 6 hours later (mCi)	activity 24 hours later (mCi)
Copper	63	69%	67.82%	(p,n)	**Zn-63**	38.4 m	6.40E-01	500	EC, Beta		3.08E+01	**830.0**	**1.25**	0.000
98%	65	31%	30.18%	(p,n)	**Zn-65**	243.6 d	5.85E+03	700	EC, Beta	2.82E-03		**0.1**	**0.05**	**0.051**
Beryllium	9	100%	2.00%	NA	NA	NA	NA	NA	NA					
2%														

The same isotopes (^{63}Zn and ^{65}Zn) are produced in the copper target support grid. However since the grid intercepts only 10-12% of the beam, overall production rates are lower. The calculations are shown in Table 11. It is recommended that grid-supported target windows not be serviced until 6 hours after the last bombardment. Production of ^{65}Zn over a year is on the order of 30 mCi.

Table 11 Copper Activation Products

Copper Hex Grid														
Element	Naturally-Occurring Isotope		Weighted Amount (%)	Activation Reaction	Activation Product	Half Life	Half-life in hours	Cross Section (mb)	Decay Mode	TTY (mCi/uA hr, >300 u of Cu)	TTY (mCi/uA at sat, >300 u of Cu)	activity produced per run (mCi)	activity 6 hours later (mCi)	activity 24 hours later (mCi)
Copper	63	69%	69%	(p,n)	Zn-63	38.4 m	6.40E-01	500	EC, Beta		5.46E+01	**347.9**	**0.52**	0.000
	65	31%	31%	(p,n)	Zn-65	243.6 d	5.85E+03	700	EC, Beta	5.05E-03		**0.1**	**0.04**	**0.036**
Interception	12%													

The calculations for a 0.001" thick titanium window (used in the ^{13}N targets) are shown in Table 12. In this case, the calculations assume a 10-minute, 60-µA run.

Table 12 Titanium Activation Products

Titanium														
Element	Naturally-Occurring Isotope		Weighted Amount (%)	Activation Reaction	Activation Product	Half Life	Half-life in hours	Cross Section (mb)	Decay Mode	TTY (mCi/uA hr, 25 u Ti)	TTY (mCi/uA at sat, 25 u of Ti)	activity produced per run (mCi)	activity 3 hours later (mCi)	activity 24 hours later (mCi)
	46	8%	8.00%	NA	NA	NA	NA	NA	NA					
	47	7%	7.00%	(p,n)	**V-47**	33 m	5.50E-01	315	EC		2.42E+01	**19.3**	**0.44**	0.000
	48	74%	74.00%	(p,n)	**V-48**	16 d	3.84E+02	500	EC, Beta	1.29E-01		**1.0**	**0.95**	**0.914**
	49	5%	5.00%	(p,n)	**V-49**	330 d	7.92E+03	>500	EC					
	50	5%	5.00%	NA	NA	NA	NA	NA	NA					

~ CSISRS database of nuclear reaction experimental data, National Nuclear Data Center, Brookhaven National Laboratory, Upton, NY. http://www.nndc.bnl.gov.

APPENDIX D Air Activation

A list of reactions on the constituents of air resulting from secondary neutron radiation from the target is listed below in Table 13 (taken from ref. 1):

Table 13 Reactions Resulting from Secondary Neutron Radiation

Reaction	Threshold*	Cross Section	Half-Life
^{16}O (n, p) ^{16}N	10 MeV	.04 b	7.5 sec
^{16}O (n, 2n) ^{15}O	18 MeV	.02 b	2 min
^{14}N (n, p) ^{14}C	0.5 MeV	.10 b	5730 y
^{14}N (n, p) ^{14}C	Thermal capture	1.81 b	5730 y
^{14}N (n, t) ^{12}C	4.3 MeV	.02 b	12 y (tritium)
^{14}N (n, 2n) ^{13}N	11.3 MeV	.02 b	10 min
^{14}N (g, n) ^{13}N	2.5 MeV	gamma	10 min

*The maximum neutron energy for any Eclipse target reaction is 8.5 MeV. The maximum gamma ray energy for any Eclipse target reaction is approximately 8.0 MeV.

APPENDIX E Dose Near Unshielded Gas Delivery Lines

The exposure dose at a point at distance h from an infinitely-long line source distribution of activity is given by (ref. 13):

$$D = 0.576\ C * E * pi/h$$

where:

D is expressed in rads/hr
C is expressed in Ci/m
E is expressed in MeV/decay (1.022 for positron emitters)
h is expressed in meters

Assuming that the far end of the line is at atmospheric pressure, and that pressure equilibrium has been reached in the line. Then, for 1 Ci in 150 cm^3 (i.e., a gas target load of 15 cm^3 at 10 atmospheres) contained in a long 32-mil bore tube the activity density is approximately 3.0E-3 Ci/m.

The exposure dose at a distance h = 1.00 meter from the line is:

$$D = 0.576 \ * \ 3.0E-3 \ * \ 1.022 \ * \ 3.1415/1.00$$
$$= 0.00555 \ \text{Rad}/\text{hr} \ \text{or} \ 5.55 \ \text{mr}/\text{hr}$$

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