PETNET Solutions

June 23, 2015

Decommissioning Branch
Region III - Division of Nuclear Materials and Safety
U.S. Nuclear Regulatory Commission
2443 Warrenville Road, Suite 210
Lisle, Illinois 60532-4352

Re: Request for approval of the submitted Decommissioning Plan for the Possession-Only Radioactive Material Licenses 41-32720-03 (Production) and 41-32720-04MD (Nuclear Pharmacy) – PETNET Solutions, Inc. St. Louis, MO Facility

To whom it May Concern,

In accordance with 10 CFR 30.36(g)(1), PETNET Solutions, Inc. is requesting approval of the proposed Decommissioning Plan (Attachment A) for the two RAM Licenses shown above. Upon NRC's approval, Chase Environmental Group (109 Flint Road, Oak Ridge, TN 37830) will commence the decommissioning process.

The decommissioning plan is comprehensive and inclusive to any and all PETNET licensed areas (i.e., production area, pharmacy area and the cyclotron vault), along with adjacent areas that might have been affected by operations. Decommissioning activities will begin as soon as possible followingthe NRC's approval of the Decommissioning Plan.

Should you have any questions please contact me at the numbers listed below or Ramón Davila at 865-332-6594 or ramondavila@siemens.com.

Sincerely,

Roger Moroney, CHP
Sr. Manager Radiation Protection/Corporate RSO
PETNET Solutions, Inc.
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Fax: 865-218-3018
william.moroney@siemens.com

cc: Tigran Sinanian, RPh, BCNP, Sr. Director of Manufacturing Operations
John Beyer, RPh, RSO, Regional Operations Director
Ramón Davila, MBA, RRPT, Regional Health Physicist

Attachment A

PETNET St. Louis – Decommission Plan

PETNET Solutions, Inc.
RDS-112 Cyclotron Facility
Decommissioning Plan

Saint Louis University Hospital
3635 Vista at Grand Blvd.
St. Louis, MO 63110

NRC Licenses:
41-32720-03 and 41-32720-04MD

May 25, 2015

Work to Be Performed Under
Chase Environmental Group's
Commonwealth of Kentucky
Radioactive Materials
License No. 201-605-90

Prepared:	Dave Culp	Field Services Manager	Date:	5/25/15
Approved:	Chris Echterling	Chase Radiation Safety Officer	Date:	6/12/15
Approved:		PETNET Representative	Date:	

CHASE
Environmental Group

Prepared by:
Chase Environmental Group, Inc.
109 Flint Road
Oak Ridge, TN 37830
865-481-8801

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ACRONYMS

ALARA	As Low As Reasonably Achievable
CRSO	Chase Corporate Radiation Safety Officer
DAC	Derived Air Concentration
$DCGL_{EMC}$	Derived Concentration Guideline Level – Elevated Measurement Comparison
$DCGL_W$	Derived Concentration Guideline Level – Wilcoxon Rank Sum
DQO	Data Quality Objective
DOT	Department of Transportation
DRS	Director, Radiological Services
DP	Decommissioning Plan
DSV	Default Screening Value
EPA	US Environmental Protection Agency
FSM	Field Services Manager
H_0	Null Hypothesis
H_A	Alternative Hypothesis
HEPA	High Efficiency Particulate Air (Filter)
HSA	Historical Site Assessment
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	Minimum Detectable Concentration
NARM	Naturally-Occurring and Accelerator-Produced Radioactive Material
NIST	National Institute of Standards and Technology
NRC	U.S. Nuclear Regulatory Commission
PET	Positron Emission Tomography
PIC	Pressurized Ion Chamber
PM	Project Manager
PPE	Personal Protective Equipment
RCS	Radiation Control Supervisor
RCT	Radiation Control Technician
RSM	Radiation Safety Manual
RWP	Radiation Work Permit
SLUH	Saint Louis University Hospital
TEDE	Total Effective Dose Equivalent
TLD	Thermoluminescent Dosimeter
USEI	US Ecology Idaho
WAC	Waste Acceptance Criteria

1.0 INTRODUCTION

PETNET Solutions (PETNET) has decided to terminate their NRC licenses for the cyclotron and nuclear pharmacy facility located at Saint Louis University Hospital. While the facility was operated under the PETNET radioactive materials license, the cyclotron and many of the facility assets are owned by SLH Vista, Inc. dba Saint Louis University Hospital (SLUH), who has financial responsibility for decommissioning. Upon completion of the decommissioning process, the space will be released back to SLUH.

The self-shielded 11 MeV Siemens RDS-112 cyclotron was used to produce short-lived radioactive materials for Positron Emission Tomography (PET) imaging. The cyclotron is located within a room in the basement of the hospital located at 3635 Vista at Grand Blvd., St. Louis, MO 63110. The facility was operated under US Nuclear Regulatory Commission (NRC) radioactive materials licenses 41-32720-03 (Production) and 41-32720-04MD (Nuclear Pharmacy Medical Distribution) issued to PETNET Solutions, Inc. Prior to the NRC assuming regulatory authority for accelerator produced radioactive materials the Missouri Department of Health and Senior Services regulated PETNET's operations under registration number IRM 145. The cyclotron was permanently shut down on January 30, 2015 and PETNET formally notified the NRC regarding their intent to decommission on February 15, 2015. All remaining radioactive materials, including sealed sources, were placed into storage at that time.

Based on previous decommissioning experience for RDS-112 cyclotrons, portions of the concrete floor underneath the cyclotron will be activated with Co-60, Eu-152, Eu-154, Cs-134 and other nuclides; and must be removed to facilitate decommissioning for unrestricted use. This Plan was developed using the guidance provided in NUREG-1757, "Consolidated NMSS Decommissioning Guidance" and NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual" (MARSSIM); and provides the approach, methods, and techniques for radiological decommissioning of impacted areas of the facility. These methods ensure technically defensible data are generated to aid in determining compliance with the criteria for unrestricted use specified in 10 CFR 20 Subpart E.

PETNET and SLUH have retained Chase Environmental Group, Inc. (Chase) to provide decommissioning services to include removing and disposing of cyclotron equipment, remediating activated building structures, and performing final status surveys to demonstrate compliance with the release criteria. Radiological work will be performed under Chase's Commonwealth of Kentucky radioactive materials license number 201-605-90 utilizing a reciprocal agreement with the NRC. A copy of the Chase license is available at www.chaseenv.com.

All licensed activities will be performed in accordance with this Decommissioning Plan (DP), Chase's Radioactive Materials License, and NRC regulations. On-site activities will commence shortly after NRC approval of this plan and are expected to take approximately three weeks to complete.

2.0 FACILITY DESCRIPTION

The cyclotron facility, located on the basement floor of the building, has a footprint of approximately 1,800 square feet and consists of a cyclotron room, laboratory areas, support areas, and an office. The various rooms located within the PETNET controlled area are presented in the table below.

Table 2-1: PETNET Controlled Areas

Room(s)	Floor Area (ft^2)	Restricted Area ?	Description
Cyclotron Room	496	YES	Contains cyclotron and support cabinets
Mechanical Room	168	YES	Contains cooling water system
Maintenance Room	56	YES	Used to build targets and repair components
Production Area	389	YES	Contains mini cells and laboratory fixtures
Dispensing Area	377	YES	Contains hot cells and laboratory fixtures
Receiving Area	146	NO	Shipping and receiving
Office	165	NO	

The general layout of the facility is presented below.

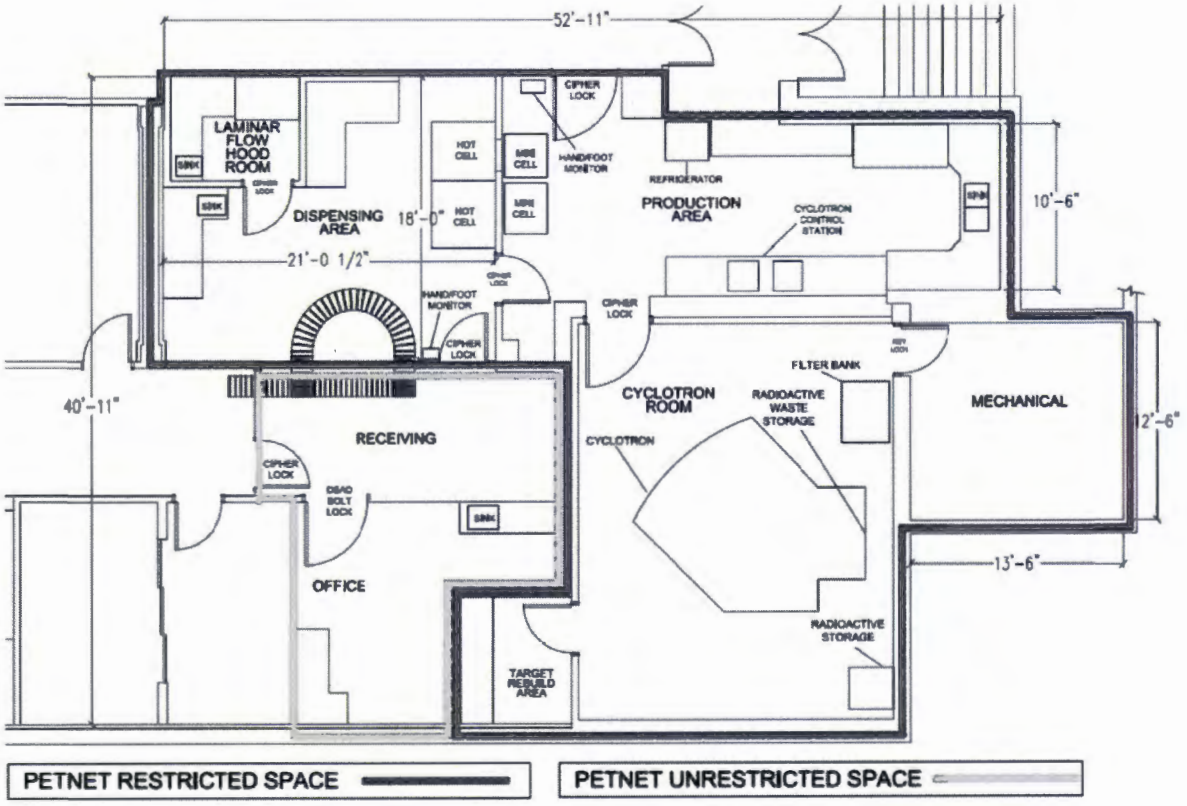

Figure 2-1: Facility Floor Plan

3.0 SITE OPERATING HISTORY

The facility was constructed in 1991 and the cyclotron first operated in September 1991. St. Louis University Hospital operated the facility until 2001, when PETNET commenced managing the facility under a lease agreement to operate and maintain the cyclotron for the production of radionuclides. In 2001 PETNET applied for and was granted a registration for accelerator produced radioactive materials from the Missouri Department of Health and Senior Services.

The Energy Policy Act of 2005 redefined "byproduct material" to include certain naturally-occurring and accelerator-produced radioactive material (NARM) under NRC jurisdiction. As a result of regulatory changes implementing the Act, in 2009 PETNET applied for a new NRC license for production of PET radionuclides at the facility. In the facility's recent operating history (within the past two years, the cyclotron has been producing approximately 20 doses per day of F-18 (FDG, Av-45) products, with lesser quantities of N-13 (NH_3 drugs).

4.0 NUCLIDES OF CONCERN

Potential contaminants include activation products from cyclotron operations. Residual radiopharmaceutical products can mathematically be eliminated from concern as they will have sufficient time to decay since shutdown.

Small quantities of H-3 are produced during the production of the radiopharmaceuticals, however the H-3 is confined to the O-18 recovery vials and the chemistry module waste vial. Due to the small amounts produced relative to the H-3 unrestricted release limits, and the careful control of the O-18 water for reprocessing off-site, H-3 is not considered a nuclide of concern for decommissioning. However, removable contamination measurements will be taken during initial facility characterization and counted by liquid scintillation counting to confirm this assumption.

4.1 CYCLOTRON COMPONENTS

Cyclotron component activation products are expected to include Ag-110m, Cd-109, Co-60, Eu-152, Mn-54, Zn-65 and others.

4.2 CONCRETE AND STEEL STRUCTURES

Many elements in concrete and structural steel can become activated by neutrons produced by (p,n) reactions, but only a few are long lived. Eu-152, Eu-154, Co-60, and Cs-134 typically dominate the dose models for building structures in these types of facilities. Other shorter-lived nuclides may be detectable, but typically contribute a minor fraction of the dose.

5.0 RADIOLOGICAL STATUS OF FACILITY

Generally accessible areas of the facility are expected to be free from residual surface contamination. Cyclotron structural components and the concrete floor underneath the cyclotron are expected to contain activation products from cyclotron operation. At the time of decommissioning, the machine will have been shut down for a sufficient period of time to allow for significant decay of short lived activation products from cyclotron operations. External dose rates are expected to be < 100 mrem/hr on contact and < 5 mrem/hr general area. The total activity of the cyclotron is expected to be less than 100 mCi and the total activity of the shields is expected to be less than 20 mCi. The concentrations of activation products in building structural concrete are expected to be low (tens to hundreds of pCi/g). The exact radiological status of the facility will be determined during facility characterization and cyclotron dismantlement.

6.0 RELEASE CRITERIA

The radiological release criteria are those specified in 10 CFR 20 Subpart E. Specifically the facility being released under this decommissioning effort will be surveyed in accordance with the guidance contained in MARSSIM to demonstrate compliance with the criteria of 10 CFR 20.1402, "Radiological Criteria for Unrestricted Use."

"A site will be considered acceptable for unrestricted use if the residual radioactivity that is distinguishable from background radiation results in a TEDE to an average member of the critical group that does not exceed 25 mrem (0.25 mSv) per year, including that from groundwater sources of drinking water, and the residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA). Determination of the levels which are ALARA must take into account consideration of any detriments, such as deaths from transportation accidents, expected to potentially result from decontamination and waste disposal."

7.0 DERIVED CONCENTRATION GUIDELINE LEVELS (DCGL)

The Derived Concentration Guideline Level (DCGL) is the surface or volumetric activity concentration that could result in a dose equal to the release criterion (the modeled dose to an occupant working in an area that contains residual radioactivity equal to the DCGL would be 25 mrem/yr TEDE). Due to the presence of various activation products in concrete, the determination of DCGLs is much more complex than simply directly applying screening values. As a result, two sets of DCGLs will be applicable for this project, a $DCGL_{surface}$ and $DCGL_{volumetric}$. The $DCGL_{surface}$ will be applied to surface contamination, and is equal to the default screening value (DSV) from NUREG-1757 for the most restrictive radionuclide, Co-60. For locations inside the cyclotron room that are likely to be activated, the $DCGL_{volumetric}$ will be an external dose rate combined with a conservative removable contamination limit to bound the dose contribution from internal sources.

In the case of non-uniform contamination, higher levels of activity are permissible over small areas. The $DCGL_{EMC}$ is derived separately for these small areas. For this facility, $DCGL_{EMC}$ is not expected to be used.

7.1 Surface DCGLs

The NRC has published DSVs for common radionuclides in NUREG-1757, Volume 1, Appendix B. These DSVs were calculated using NRC-approved DandD, version 2.1 software under default conditions. Some of the nuclides of concern are not listed in NUREG 1757, therefore DSVs were calculated using DandD v2.1 with an input total surface activity of 1 dpm/100 cm^2 and accepting the default parameter values of the building occupancy scenario. Copies of the

dose model output reports are presented in Appendix A. A summary of the DSVs is provided in the table below.

Table 7-1: Surface Activity DSVs

Nuclide	Half-Life	mrem/yr per dpm/100cm^2	DCGL[1] (dpm/100cm^2)
Co-60	5.3 years	3.55E-3	7.1E3
Cs-134	2.1 years	1.96E-3	1.3E4
Eu-152	13.6 years	1.97E-3	1.3E4
Eu-154	8.8 years	2.18E-3	1.1E4

For conservatism and convenience, the Co-60 DCGL will be applied to surface contamination measurements in areas where there is no building structure activation. An important assumption of the DandD dose model is that removable surface contamination is 10% of the total surface contamination. Therefore, removable contamination measurements shall not exceed 10% of the total surface contamination limit.

7.2 Volumetric DCGLs

The DSVs described above assume surface contamination at a depth not to exceed 1 cm and are therefore not appropriate for volumetric contamination that exists from activation of building structural surfaces. Volumetric contamination can be modeled using RESRAD-BUILD software. However, modeling of volumetric contamination presents challenges due to the conservative nature of the dose model, the difficulty in obtaining sufficient sample data to determine the actual radionuclide distribution, and the heterogeneous nature of activation of concrete and steel structures. Dose modeling is typically performed when the dose cannot be directly measured in the field. In the case of volumetrically activated concrete, almost all dose to the receptor is from the external component because the activation products are not available for inhalation or ingestion. However, surface contamination could consist of dusts from concrete remediation. The external component of dose can be directly measured, therefore the approach to dose modeling will consist of performing direct radiation measurements to ensure external doses are less than 24 mrem/yr and demonstrating through modeling that the internal dose contribution is less than 1 mrem/yr.

The external dose limit equivalent to 24 mrem/yr is easily calculated using the occupancy assumption of the building occupancy scenario as follows:

[1] DSVs were determined for each nuclide by dividing the release criterion of 25 mrem/yr by the output of the dose model in mrem/yr per dpm/100cm^2.

$$\frac{24\ \text{mrem}}{\text{year}} * \frac{\text{year}}{2{,}340\ \text{hours}} * \frac{1{,}000\ \mu\text{rem}}{\text{mrem}} = 10.3\ \frac{\mu\text{rem}}{\text{hour}}$$

The internal dose limit is met by establishing a removable surface activity limit that would result in 1 mrem/yr internal dose. The DandD software reports internal dose components for each nuclide, based on the assumption that 10% of the total surface activity is removable. Therefore, an input of 1 dpm/100 cm^2 total activity yields DandD output that includes internal exposure in mrem/yr per 0.1 dpm/100 cm^2 of removable surface activity. Calculations of the removable surface activity limit for each nuclide of concern are presented in the table below.

Table 7-2: Removable Activity Contribution to Total Dose

Nuclide	Annual Internal Dose[2] (mrem per 0.1 dpm/100 cm^2)			Removable Activity Equivalent to 1 mrem/yr Internal Dose[3] (dpm/100 cm^2)
	Inhalation	Ingestion	Total	
Co-60	4.87E-4	2.93E-5	5.16E-4	1.94E2
Cs-134	9.34E-5	7.21E-5	1.66E-4	6.04E2
Eu-152	5.12E-4	7.32E-6	5.19E-4	1.93E2
Eu-154	6.54E-4	1.06E-5	6.65E-4	1.50E2

The most conservative removable activity limit (Eu-154, 150 dpm/100 cm^2) is used as the removable surface activity DCGL.

7.3 Summary of DCGLs

DCGLs were determined using the most limiting nuclide. The following table summarizes all project DCGLs.

Table 7-3: Summary of DCGLs

Type of Activity	$DCGL_{Total}$	$DCGL_{Removable}$
Surface (areas without activated building structures)	7,100 dpm/100 cm^2	710 dpm/100 cm^2
Volumetric (areas with activated building structures)	10 µrem/hr	150 dpm/100 cm^2

[2] Annual doses are obtained from the DandD modeling reports presented in Appendix B.
[3] The removable activity equivalent to 1 mrem/yr is determined by dividing 0.1 dpm/100 cm^2 by the total annual internal dose per 0.1 dpm/100 cm^2 in mrem/yr.

7.4 Materials and Equipment

The radiological release limits for surface contaminated materials and equipment under the Chase radioactive materials license are:

- 5,000 dpm/100 cm^2 total surface contamination (averaged over 1 m^2)
- 15,000 dpm/100 cm^2 max total surface contamination (limited to 100 cm^2)
- 1,000 dpm/100 cm^2 removable surface contamination

Structural concrete and steel materials are expected to contain volumetric activation products and are therefore not suitable for release using the surface contamination limits. Because there are no established volumetric contamination release criteria, any materials that exhibit an increase in the audible count rate on a 2" x 2" sodium iodide detector will be packaged and shipped for disposal as radioactive waste. Materials with no increase in the audible count rate on a 2" x 2" sodium iodide detector and that meet the surface contamination limits may be released for unrestricted use.

8.0 ALARA GOALS

Personnel exposure shall be limited to total cumulative project exposure of 50 mrem per project person.

9.0 WORK PLAN LIMITATIONS

All work will be stopped and this plan revised if any of the conditions below occur.

- If air sample results inside the restricted area are $\geq$ 30% of the DAC
- If removable contamination sample results inside the restricted area are $\geq$ 60,000 dpm/100 cm^2 beta/gamma or as directed by the Chase Corporate Radiation Safety Officer (CRSO)
- If dose rates at 30 cm inside the restricted area are $\geq$ 500 mrem/hr or as directed by the CRSO
- If dose rates at the boundary of the restricted area are $\geq$ 2 mrem/hr (public dose limits) or as directed by the CRSO

10.0 NOTIFICATIONS

Chase personnel will notify the CRSO of conditions or situations that present a radiological hazard, concern or exceed limitations set forth in this work plan or the Chase Environmental Group's Radiation Safety Manual (RSM).

11.0 PLANNED ACTIVITIES

11.1 CYCLOTRON EQUIPMENT REMOVAL

Chase will provide the necessary training and health physics oversight to disassemble all items as required for movement, packaging, and characterization. Chase will perform radiological surveys to verify the radiological status of the facility prior to commencing physical work at the site. Surveys will consist of dose rate surveys of the cyclotron and vault as well as removable contamination surveys on equipment and structural surfaces, and in drains and ventilation ducts. After ensuring there is no removable contamination on accessible surfaces in the cyclotron room, disassembly of the cyclotron can begin. Subcontracted rigging services will be used to facilitate the disassembly and removal. PETNET facilities personnel will disconnect and de-energize all electrical/mechanical systems from the cyclotron, cabinets, auxiliary support systems and components; perform all system isolations; and perform required lock-out/tag-outs. Chase personnel will verify.

Chase will provide continuous radiological coverage and perform radiological surveys as newly exposed surfaces are revealed while workers disassemble, remove, and package the cyclotron and ancillary equipment. Materials that exhibit an increase in the audible count rate on a 2" x 2" will be packaged and shipped as radioactive waste. Materials with no increase in the audible count rate, and that meet the surface contamination limits, will be released for unrestricted use. During and after the movement of materials, Chase will perform dose rate measurements as well as removable contamination surveys to ensure external surfaces of all materials meet removable residual radioactivity limits prior to movement outside. Items that will be moved and dispositioned include the following:

- Cyclotron ~ 56,400 lb.
- Utility Cabinets (power supplies, controls, hydraulic/vacuum systems, power distribution) ~ 5,000 lb.
- Utility Support Structure ~ 8,000 lb.
- Water Recirculating system ~ 1,500 lb.
- Shielding (composite blocks) ~ 47,220 lb.
- Shielding (lead target region) ~ 12,491 lb.
- Lead Shielding (target region) ~ 1,700 lb.
- Lead Bricks/Plates used for Localized Shielding ~ 2,000 lb.

Material movement will be accomplished via a removable skylight located directly over the cyclotron utilizing a 120 ton crane located on Rutger Street. Materials will be characterized, classified, and packaged according to Department of Transportation (DOT) requirements and placed onto transport vehicles staged on Rutger Street. Radioactive materials will be attended by project personnel at all times during movement from the facility until loaded onto the transport vehicles.

11.2 REMEDIATION

As required, remediation methods that will be used include simple decontamination (i.e. wet wiping with a mild detergent) and removal of contaminated/activated material by dismantling systems and structures and/or cutting sections from the material. Cutting may consist of the use of reciprocating saws, band saws, high leverage shears, electric snips, tin snips, and/or ratcheting cable cutters. Concrete removal may consist of sawing and jack hammering. HEPA-filtered vacuums will be used to remove loose dry material from surfaces during remediation activities. All remediation activities will be conducted to control the spread of contamination and to maintain personnel exposures ALARA.

Portions of the concrete floor beneath the cyclotron are expected to be removed in order to meet the release criterion. A reasonable estimate of the amount of activated concrete is ~ 30 ft^3 (~ 4,500 lbs.) based on previous experience. Measurements with a 2"x2" NaI detector will be used to guide remediation efforts. The activated portions of the concrete pad will be removed by a subcontracted concrete cutting company utilizing wet sawing methods. Chase will provide the necessary training and continuous radiological coverage to the subcontracted concrete cutting company for cutting, removal, and packaging of the concrete floor. Chase will plug all drains and provide spill prevention and containment features. All removal activities will be conducted to control the spread of contamination and to maintain personnel exposures ALARA. Radiological surveys consisting of scan surveys, direct measurements, and removable contamination measurements will be performed in support of removal activities to ensure that contamination is not spread. A HEPA-filtered ventilation machine and HEPA filtered vacuums will be used during invasive activities and continuous air sampling for radioactive materials will be conducted.

During concrete removal, approximately three representative concrete samples will be collected for gamma spectroscopy analysis to determine the nuclide distribution. The distribution data will be used for dose modeling to support release of the facility for unrestricted use. Descriptions and results of dose modeling will be presented in the final status report.

12.0 MANAGEMENT ORGANIZATION

Chase will implement their Kentucky radioactive materials license at the site. As such, Chase will have complete control of areas. As the licensee, PETNET will will oversee decommissioning activities and SLUH will maintain responsibility for building maintenance, fire, and security functions. There will be clear separation of licensed activities between Chase, SLUH and PETNET; and activities will be coordinated such that no party violates the license of another party. Chase will clearly post and control areas as necessary to prevent inadvertent entry by unauthorized personnel.

The following personnel structure will be utilized for administration and implementation of this Plan. Each person is responsible for their own safety and has stop-work authority in the event they witness an operation that they feel presents an imminent radiological or safety hazard to employees, the environment, or the public.

12.1 CORPORATE RADIATION SAFETY OFFICER (CRSO)

Chase's CRSO is responsible for the corporate management of the radiological control and safety program and for directing the program to limit occupational radiation exposures to levels ALARA as specified in Chase's Radioactive Materials License.

The CRSO's responsibilities include, but not are limited to, the following:

- Establishing standards and guidelines for radiological services operations to comply with Chase policies and applicable federal and state regulatory requirements;
- Providing selection criteria for equipment, supplies and services for radiological controls and personnel exposure monitoring;
- Establishing standards for personnel protection to assure that exposures to ionizing radiation and radioactive contamination are maintained ALARA;
- Implementing the radiological control and safety audit program of individual projects;
- Establishing company policy to comply with state and federal statutes, rules, regulations and license conditions;
- Ensuring the quality of protective equipment for personnel and prescribing usage standards; and
- Establishing procedures for radiological protection and monitoring, including the ALARA program.

Chris Echterling is the CRSO and can be reached at 865-603-2618.

12.2 DIRECTOR, RADIOLOGICAL SERVICES (DRS)

The DRS reports to the Chase Board of Directors and is responsible for providing corporate and technical support to field projects including health physics, occupational safety, legal and/or administrative support. The DRS may choose to provide these support capabilities through permanent staffing or by subcontracting through outside organizations.

John O'Neil is the DRS and can be reached at 865-384-7555.

12.3 FIELD SERVICES MANAGER (FSM)

The FSM directs all aspects of operations including radiological activities. The FSM establishes policies and procedures to assure regulatory compliance and oversees all aspects of health physics operations to ensure regulatory compliance and adherence to the ALARA principle. As the alternate Radiation Safety Officer, the FSM will act as CRSO in the absence of the CRSO named on the Chase license.

The FSM is responsible for assigning PMs to individual projects and for providing technical support to projects. Technical support encompasses health physics, occupational safety, and /or administrative support. The FSM reports to the DRS.

Dave Culp is the FSM and can be reached at 865-207-3664.

12.4 PROJECT MANAGER (PM)

The Project Manager is responsible for project operations from initiation through completion. The PM's duties include the following:

- Maintaining compliance with conditions of site operating licenses, permits, rules, regulations and procedures of Chase, and state and federal agencies;
- Maintaining working conditions which assure health, safety and protection for all employees, visitors and the environment;
- Providing physical examinations for employees as required by company policy, local, state and federal regulations;
- Ensuring that employees are instructed regularly, or as required by law, on precautions, procedures and practices to be followed to minimize exposure to radioactive materials and to conduct operations safely;
- Notifying the CRSO promptly, of any operation or condition which appears to present a radiological hazard to employees, the public or the environment; or exceed limitations set forth in this plan, the Radiation Safety Manual or applicable procedures and work plans;

- Furnishing proper personnel protective equipment, ensuring that employees are instructed its proper use, and enforcing rules for the equipment's utilization;
- Ensuring that sufficient staffing for the project is present and that staffing consists of individuals able to conduct daily operations in compliance with regulatory requirements, and to maintain a safe working environment; and
- Maintaining project radiation exposures ALARA.

Dustin Miller is the PM and can be reached at 314-240-0507.

12.5 RADIATION CONTROL SUPERVISOR (RCS)

The RCS reports directly to the PM and is responsible for the implementation of the Radiation Protection Program (RPP) at the project. Responsibilities may include but are not limited to the following:

- Monitoring site conditions to ensure compliance with the RPP and the Chase Radioactive Materials License;
- Determining appropriate PPE;
- Ensuring that the CRSO is notified of conditions or situations that present a radiological hazard, concern, or exceed limitations set forth in the Radiation Safety Manual or applicable procedures and work plans;
- Issuing Radiation Work Permits (RWP); and
- Maintaining records related to the RPP in an auditable condition for the duration of the project.

Mike Culp is the RCS and can be reached at 865-850-2767.

12.6 RADIATION CONTROL TECHNICIANS (RCTS)

RCTs report to the RCS and act as the RCS's representatives in specifically implementing the RPP. Responsibilities may include but are not limited to the following:

- Performing and documenting radiological surveys;
- Maintaining, inspecting, and performing operational checks of field instrumentation;
- Identifying and controlling radiation protection hazards; and
- Performing job coverage duties, (i.e., surveys, contamination control, air sampling, sample analysis, environmental sampling, custody control, etc.).

12.7 RADIATION WORKERS

Radiation workers are individuals who have received training for unescorted accesses into Restricted Areas to perform work where they may receive exposure to ionizing radiation. A Radiation Worker's responsibilities include, but are not limited to, the following:

- Obeying all posted, verbal, and Radiation Work Permit (RWP) instructions;
- Wearing dosimetry as required;
- Tracking and controlling one's own radiation exposure;
- Minimizing exposure;
- Not eating, drinking, or smoking in areas where dispersible radioactive material may be present;
- Utilizing contamination control techniques to prevent the spread of contamination;
- Properly utilizing anti-contamination clothing and respiratory protection equipment;
- Adhering to personnel monitoring requirements when leaving a contaminated area;
- Notifying radiological control personnel in the event of a spill.

13.0 PROJECT TRAINING REQUIREMENTS

This section describes the minimum training that Chase and subcontractors will possess prior to conducting licensed activities. SLUH/PETNET may conduct additional training at their discretion to satisfy the conditions of their site procedures or to allow unescorted access into their facility.

13.1 RADIOLOGICAL TRAINING

Radiological training will be completed and documented in accordance with Section 7 of the Chase Radiological Safety Manual (RSM). The PM will maintain a copy of each individual's certification in the project file.

13.2 PROJECT SPECIFIC TRAINING

Prior to conducting licensed activities, personnel will attend an initial project specific training session conducted by the PM. The training session will include the following items:

- Review of the Work Plan
- Discussion regarding the scope of work and planned work activities
- Review of chemical, physical, and radiological hazards

- Types and use of available personal protective equipment
- Project security control and operational work zones
- Emergency response and site evacuation procedures
- Air monitoring and medical monitoring procedures
- Project communications
- General safe work practices
- Decontamination procedures
- Radiation Work Permits
- Review of applicable regulatory standards as applied to project operations

13.3 GENERAL SAFETY BRIEFINGS

General safety meetings will be held by the PM at the beginning of each work shift until project completion. The purpose of these meetings will be to discuss project status, potential problem areas, general safety concerns, and to reiterate Plan requirements. Additional meetings will be held if conditions warrant.

13.4 VISITOR ORIENTATION

All non-essential personnel and visitors will be briefed on the Plan requirements and will be trained in accordance with the Chase RSM Section 7 "Personnel Training." Section 7 specifies visitors must be escorted at all times and receive at a minimum General Orientation training. General orientation shall be administered to all personnel, contractors, and visitors requiring access to restricted areas. The scope of orientation shall be commensurate with the activities being performed and the risks involved. General orientation training shall be valid only for the particular project at which it is administered. Escorts shall have a minimum of Radiation Worker training.

Additionally, all visitors must receive training and/or briefings in accordance with SLUH's policies on parking and entrance into the facility

13.5 TRANSPORTATION TRAINING

Persons who prepare hazardous materials for transportation or are otherwise responsible for safely transporting hazardous material will be trained in accordance with the requirements of 49 CFR 172, subpart H. The PM will maintain copies of certifications on-site in the project file.

14.0 PROJECT TASK MANAGEMENT

Licensed activities will be conducted under the provisions of the Chase radioactive materials license and in accordance with this DP. Activities involving licensed material shall be conducted in accordance with written and approved procedures, radiation work permits (RWP), and/or survey packages to ensure adequate worker protection and to comply with the radioactive materials license and this DP.

15.0 RADIATION PROTECTION

Radiological work will be performed according to the Chase radioactive materials license radiation protection program (RPP). The RPP will be implemented commensurate with the scope and extent of licensed activities at the site. This program and associated operating procedures are the primary means used to administratively establish safe radiation work practices and ensure compliance with NRC requirements. Selected sections of particular relevance to this project are discussed below.

15.1 RADIATION WORK PERMITS (RWP)

RWPs will be prepared, reviewed and authorized in accordance with the RWP procedure that addresses request, initiation, development, issuance, and termination of an RWP. The RWP contains the information listed below.

- Job description
- Permit Start and Expiration dates
- Work locations
- Radiation and contamination levels
- Airborne radioactivity concentrations
- Personnel Protective Equipment requirements
- Dosimetry requirements
- Respiratory Protection requirements
- Additional permits that may be required
- Health Physics coverage requirements
- Instructions to workers
- Access Controls
- ALARA Limits

- Radiation Area Controls
- Contamination Control
- Special Conditions

The RCS will initiate the RWP and provide a description on the RWP of existing and/or anticipated radiological conditions. RWP development will include specific identification of the radiological conditions and radiological protection requirements (e.g. clothing, respiratory protection, dosimetry, monitoring, and training). Also, hold points and special instruction may be described on the RWP. The RWP form contains items such as the job description, location, known radiological conditions, protective clothing requirements, respiratory protection, dosimetry, training, health physics monitoring requirements, and any other special instructions. RWP development also includes creating a sign-in/out sheet for use by the authorized users. After development, the RWP must be approved for issuance by the RCS. Issuance includes a review of the RWP with the authorized users, as required. A pre-job meeting may also be prerequisite to issuance of the RWP.

During use, a copy of the RWP will be maintained at the worksite, and authorized users will be required to sign-in/out when participating in the subject activity, indicating their understanding of the requirements of the RWP. RWPs will be terminated upon completion of the activity by signature on the RWP and completion of a form indicating the reason for termination and confirmation of final radiological survey of the activity or area. Upon termination of the RWP, the RWP package will be completed and filed. The package generally contains the completed RWP, sign-in sheets, applicable radiological surveys, and any other documents pertinent to the job. If radiological conditions or requirements change, appropriate changes to the RWP may be made by the RCS or designee. Alternatively, a new RWP may be issued.

15.2 DOSIMETRY

Each individual who will perform work under the Chase radioactive material license during this project will be monitored for external doses by whole body thermo-luminescent dosimeters (TLDs). Self-reading pocket dosimeters may be required by the RWP or at the discretion of the RCS.

15.3 AIR SAMPLING

Airborne particulate sampling will be performed during all cyclotron disassembly and concrete cutting operations to assess the potential for internal exposures. A limiting airborne concentration limit of 8E-9 μCi/ml will be used to estimate

doses from airborne radioactivity, based on the most limiting DAC value of the nuclides of concern (Eu-154).

15.4 PERSONAL PROTECTIVE EQUIPMENT (PPE)

Personnel conducting concrete cutting operations will wear, at a minimum, Tyvek coveralls, rubber overshoes, latex or rubber gloves, and safety glasses. Engineering controls are expected to be sufficient to control airborne radioactivity levels. Additional PPE requirements may be required by the RWP or at the discretion of the RCS.

16.0 RADIOACTIVE MATERIALS MANAGEMENT

Chase will characterize materials for radioactivity by using a combination of activation profiles for similar cyclotrons obtained using high purity germanium gamma spectroscopy equipment, on-site measurements using a 2" x 2" sodium iodide based portable gamma spectroscopy system, and dose-to-curie inferences using Microshield™ software.

Radioactive materials will be transported via DOT approved carriers and manifested by qualified shippers to the appropriate waste processing and/or disposal sites. Chase will provide packaging suitable for Class 7 Hazardous Materials, as applicable, and a qualified Hazmat shipper to oversee containerization and blocking/bracing of the material by the rigging contractor. Chase will provide all packaging materials and prepare the packages for transport over public highways in accordance with appropriate DOT regulations based on field measurements and dose-to-curie inferences. Shipping papers will reflect that Chase is the shipper and PETNET is the generator; Chase will sign the Shipper's Certification required for shipments containing hazardous material.

Depending upon the characterization results when the cyclotron is disassembled, components may be designated for disposal at a licensed facility or at the US Ecology Subtitle C disposal site in Grandview Idaho (USEI).

17.0 ALTERNATE DISPOSAL PROCEDURES

In order to make the USEI option available, PETNET is requesting NRC approval (via approval of this DP) of this disposal method according to 10CFR20.2002. The required exposure assessments have been completed by USEI and approved by the US Environmental Protection Agency (EPA) as part of the permitting process.

Wastes will be handled in a controlled manner by trained individuals to ensure doses to workers and the public are maintained ALARA. Wastes will be packaged at the St. Louis facility by qualified radiation workers in compliance with the

Chase radioactive materials license, transported directly to USEI according to US DOT requirements using properly licensed/permitted carriers and transport vehicles, and received/handled by USEI according to the provisions of their EPA Permit. No other facilities will be affected by these disposal procedures.

The acceptance criteria for particle accelerator produced radioactive material are presented in Table C.3: Particle Accelerator Produced Radioactive Material of the USEI waste acceptance criteria (WAC) and reproduced in the table below.

Table 17-1: US Ecology Idaho Waste Acceptance Criteria

Acceptable Material	Activity or Concentration
Any particle accelerator produced radionuclide.	All materials shall be packaged in accordance with USDOT packaging requirements. Any packages containing iodine or volatile radionuclides will have lids or covers sealed to the container with gaskets. Contamination levels on the surface of the packages shall not exceed those allowed at point of receipt by USDOT rules. Gamma or x-ray radiation levels may not exceed 10 millirem per hour anywhere on the surface of the package. All packages received shall be directly disposed in the active cell. All containers shall be certified to be 90% full.

18.0 QUALITY ASSURANCE PROGRAM

Due to the limited scope of the planned activities, project-specific quality requirements are included in this plan and a separate Quality Assurance Project Plan (QAPP) is not warranted. This plan will be supported by the Chase corporate Quality Assurance (QA) program and meet the guidelines of MARSSIM Section 9. QA criteria are applied in a graded manner to achieve a balance between the rigor of application of quality assurance measures and the scale, cost, and complexity of the work involved.

Accountability for quality is everyone's responsibility, extending from the PM through established lines of authority to all project personnel, who are responsible for the requisite quality of their own work. Quality assurance will be implemented by personnel conducting their activities to meet requirements and expectations according to established plans and procedures that reflect the way business is to be conducted on the project.

All project personnel are responsible for executing their work and ensuring that quality-affecting activities within their purview are performed in conformance with applicable plans and procedures. All personnel have the authority and responsibility to stop his/her own work and the responsibility to report such

conditions when continuation will produce or conceal results that are not in accordance with prescribed requirements, and/or pose imminent radiological or safety hazard to employees, the environment, or the general public. Project personnel have sufficient freedom, authority, access, and responsibility to:

- Identify quality problems, deficiencies, nonconformance's, and noncompliance with regulatory and performance objectives
- Initiate, recommend, or provide solutions through designated channels
- Verify implementation of the solutions
- Assure that deficient work is stopped or is proceeding under controlled conditions until proper disposition of the unsatisfactory condition is accomplished

18.1 NONCONFORMANCE CONTROL AND CORRECTIVE ACTION

All project personnel shall be responsible for notifying their supervisor, the Project Manager, and/or the Quality Assurance Manager (QM) of conditions or items that do not meet specified requirements. Chase policy defines the controls, which address the following measures:

- Identification or segregation of the nonconformance;
- Documentation of the nonconformance;
- Evaluation of the nonconformance;
- Disposition and justification provisions;
- Notification to affected personnel or organizations, and;
- Verification of disposition.

In the case of significant conditions adverse to quality, the PM together with the QM shall treat these conditions in accordance with Chase procedure QAP 16.1, "Corrective Action."

All project personnel are encouraged to identify any activity, process, or procedure that could lead to a potential non-conformances or a condition adverse to quality. Chase procedure QAP 16.1, "Corrective Action" also provides the reporting and evaluation requirements for preventative actions resulting in the elimination of potential quality problems. All non-conformances, corrective actions, and preventative actions shall be documented and maintained in accordance with Chase procedures "Document Control" and "Quality Assurance Records," QAP 6.1 and 17.1.

18.2 QUALITY ASSURANCE AUDITS

The Chase CRSO shall be responsible for planned and periodic audits of project activities. These audits shall be scheduled in a manner that will provide sufficient coverage and coordination of activities throughout the duration of the project. These audits will verify compliance with the requirements specified in this plan, related procedures, plans, and regulatory requirements. These audit activities also provide a mechanism to identify opportunities for continuous improvement. Due to the limited scope and duration of this project, an audit is not expected to be performed.

In addition to this audit activity, the PM or designee shall perform periodic surveillances to monitor and document compliance with this plan and standard radiological and safety practices. Identified departures from specified requirements shall be treated as non-conformances and corrected.

18.3 SAMPLE CHAIN-OF-CUSTODY

The sample chain-of-custody (COC) maintains the integrity of the sample; that is, there is an accurate record of sample custody during collection, transport, and analysis. This ensures that samples are neither lost nor tampered with, and that the sample analyzed in the laboratory is actually and verifiably the sample taken from a specific location in the field. Samples sent off-site for analysis will use an approved Chain of Custody Procedure.

18.4 QUALITY ASSURANCE SURVEYS

Quality Assurance surveys will consist of duplicating the final status survey protocol for building structural surfaces at a rate of 5% to include scans, static measurements, and smears.

19.0 SURVEY INSTRUMENTATION

19.1 INSTRUMENTATION SPECIFICATIONS

The radiation detection instrumentation to be used is summarized in the table below. With approval of the CRSO, alternate instruments may be used that provide adequate detection sensitivity to meet data quality objectives.

Table 19-1: Instrumentation Specifications

Meter Model	Detector Type	Detector Model	Use
Ludlum 2241-3	Gas Flow Proportional	Ludlum 43-37	Scans, Direct Measurements
Ludlum 2241-3	Gas Flow Proportional	Ludlum 43-68	Scans, Direct Measurement
Ludlum 3	Geiger Mueller	Ludlum 44-9	Beta Gamma Surveys, Frisking
Ludlum 2241	2 x 2 Sodium Iodide	Ludlum 44-10	Gamma Scans
Ludlum 3	3 x 3 Sodium Iodide	Ludlum 44-103	Gamma Scans
Ludlum 2929	ZnS(Ag) Adhered to Plastic Scintillation Material	Ludlum 43-10-1	Airborne Radioactivity and Removable Contamination Measurements
Packard Tri-Carb	Liquid Scintillation	N/A	H-3 Removable Contamination Measurements
Eberline RO-2A	Ion Chamber	N/A	External Dose Rate Measurements
Bicron MicroRem	Tissue Equivalent Gamma Scintillation	N/A	External Dose Rate Measurements
Victoreen 450P	Pressurized Ion Chamber	N/A	External Dose Rate Measurements

Laboratory and portable field instruments will be calibrated at least annually with National Institute of Standards and Technology (NIST) traceable sources.

Functional checks will be performed at least daily when in use. The background, source check, and field measurement count times for radiation detection instrumentation will be specified by procedure to ensure measurements are statistically valid. Background readings will be taken as part of the daily

instrument check and compared with the acceptance range for instrument and site conditions. If an instrument fails a functional check, all data obtained with the instrument since the last satisfactory check will be reviewed to determine the validity of the data.

19.2 MINIMUM DETECTABLE CONCENTRATIONS

Minimum counting times for background determinations and counting times for measurement of total and removable contamination will be chosen to provide a minimum detectable concentration (MDC) below radiological limits. NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual" (MARSSIM) instrumentation performance protocols will be used. MARSSIM equations relative to building surfaces have been modified to convert to units of dpm/100 cm^2. Count times and scanning rates for surface contamination are determined using the following equations:

19.3 STATIC COUNTING

Static counting MDC at a 95% confidence level is calculated using the following equation, which is an expansion of NUREG-1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions", Table 3.1 (Strom & Stansbury, 1992):

$$MDC_{static} = \frac{3 + 3.29\sqrt{B_r \cdot t_s \cdot (1 + \frac{t_s}{t_b})}}{t_s \cdot E_{tot} \cdot \frac{A}{100 cm^2}}$$

Where:

MDC_{static} = minimum detectable concentration level in dpm/100cm^2

B_r = background count rate in counts per minute

t_b = background count time in minutes

t_s = sample count time in minutes

E_{tot} = total detector efficiency for radionuclide emission of interest (includes combination of instrument efficiency and surface efficiency)

A = detector probe area in cm^2

19.4 RATEMETER SCANNING - SURFACE

Scanning MDC at a 95% confidence level is calculated using the following equation which is a combination of MARSSIM equations 6-8, 6-9, and 6-10:

$$MDC_{scan} = \frac{d'\sqrt{b_i}\left(\frac{60}{i}\right)}{\sqrt{p}\cdot E_{tot}\cdot\frac{A}{100cm^2}}$$

Where:

MDC_{scan} = minimum detectable concentration level in dpm/100 cm^2

d' = desired performance variable (1.38)

b_i = background counts during the residence interval

i = residence interval

p = surveyor efficiency (0.5)

E_{tot} = total detector efficiency for radionuclide emission of interest (includes combination of instrument efficiency and surface efficiency)

A = detector probe area in cm^2

19.5 RATEMETER SCANNING – VOLUMETRIC

Sodium iodide detectors will be used qualitatively for scanning activated structures to guide remediation. Areas of elevated activity identified by an audible increase in the count rate will be evaluated with external dose rate measurements.

19.6 SMEAR COUNTING

Smear counting MDC at a 95% confidence level is calculated using the following equation, which is NUREG-1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions", Table 3.1 (Strom & Stansbury, 1992):

$$MDC_{smear} = \frac{3+3.29\sqrt{B_r\cdot t_s\cdot(1+\frac{t_s}{t_b})}}{t_s\cdot E}$$

Where:

MDC_{smear} = minimum detectable concentration level in dpm/smear

B_r = background count rate in counts per minute

t_b = background count time in minutes

t_s = sample count time in minutes

E = instrument efficiency for radionuclide emission of interest

20.0 SURVEY DOCUMENTATION

Survey packages will be developed for each survey area that contain specific survey instructions. Survey package preparation and completion will be approved by the PM or designee to ensure all survey requirements and Data Quality Objectives (DQOs) are met. As applicable, each survey package will contain:

- Survey unit number
- Maps of the survey unit surfaces
- Overview maps detailing survey locations and placement methodology
- General survey requirements
- Instrument requirements with associated Minimum Detectable Concentrations (MDCs), count times and scan rates
- Survey Instruction Sheets
- Percentage of surface requiring scan surveys
- Number of measurements required
- Additional specific survey instructions
- Survey Data Sheets
- Sampling protocols
- Chain of Custody Forms
- Signature of Preparer, Surveyor and Reviewer

21.0 CHARACTERIZATION SURVEYS

The survey protocol for building surfaces will consist of performing the scanning portion of the final status survey protocol, with judgmental smears and static measurements on surfaces with the highest probability for residual radioactivity.

The purpose of scanning is to identify locations of elevated activity. Where elevated activity is identified, a static measurement and smear will be taken at the location of highest activity identified during the scan. Where elevated activity is identified, the boundary of the elevated area will be marked to aid in locating the area for remedial actions. Sodium iodide scans will be performed to detect elevated activity as a result of activation based on the audible response. These

areas will then be evaluated for compliance with the dose criterion with a MicroRem meter or a Pressurized Ion Chamber (PIC) and smears.

The survey protocol for building system surveys will consist of performing removable contamination measurements on internal surfaces of ventilation and drain systems consistent with the final status survey protocols contained in this plan.

If the initial characterization survey results indicate that contamination is not present in excess of the release criteria, then data from the survey may be used as part of the final status survey. For areas that are partially contaminated, the characterization survey data may be used as part of the final status survey measurements provided that 1) the data used is only from areas with contamination levels below the release criteria, and 2) decontamination work is controlled such that the survey location could not have become cross-contaminated.

22.0 REMEDIAL ACTION SURVEYS

Remediation will be conducted to control the spread of contamination and keep personnel exposures ALARA. Remedial action surveys are conducted in support of remediation activities to help determine when the area is ready for a final status survey and to provide updated estimates for final status survey planning. Remedial action surveys serve to monitor the effectiveness of decontamination efforts and ensure that surrounding areas are not cross-contaminated from remediation actions.

Remedial action surveys will consist of scan surveys, direct measurements, dose rate measurements, and removable contamination measurements. These will be conducted following remediation activities to establish the success or failure of the efforts to decontaminate the applicable survey area. Results of the survey will be the decision basis for continued remediation or conduct of final status surveys.

Remedial action surveys will be designed to meet the objectives of the final status surveys. To the extent allowed by MARSSIM, the results of the remedial action surveys will be used to supplement the final status survey.

23.0 FINAL STATUS SURVEYS

Final status surveys are performed to demonstrate that residual radioactivity in each survey unit satisfies the predetermined criteria for release for unrestricted use. The final status survey will be conducted using the Data Quality Objective (DQO) process. Characterization and remedial action survey data will be used as final status survey data to the extent possible.

Final status surveys will be conducted by performing required scan surveys, total direct surveys, dose rate measurements and removable contamination measurements as discussed further in this section. All survey data shall be documented on survey maps and associated data information sheets.

23.1 BACKGROUND DETERMINATION

The use of reference background areas or paired background comparisons is not necessary for beta surface contamination measurements. The background rate for NaI and PIC detectors will be determined in a non-impacted area of similar construction. Background will be subtracted from survey unit measurements and used to calculate actual MDCs for measurements.

23.2 DATA QUALITY OBJECTIVES

The following is a list of the major data quality objectives (DQOs) for the survey design described in this plan:

- Static measurements will be taken to achieve an MDC_{static} of less than 50% of DCGL.
- Scanning will be conducted at a rate to achieve an MDC_{scan} of less than 50% of the DCGL.
- Removable surface activity measurements will be conducted to achieve an MDC_{smear} of less than 50% of the DCGL.
- Individual measurements will be made to a 95% confidence interval.
- Decision error probability rates will initially be set at 0.05 for both α and β.
- The null hypothesis (H_0) and alternative hypothesis (H_A) are that of NUREG-1505 scenario A:
    - H_0 is that the survey unit does not meet the release criteria
    - H_A is that the survey unit meets the release criteria
- Characterization and remedial action support surveys will be conducted under the same quality assurance criteria as final status surveys such that the data may be used as final status survey data to the maximum extent possible.

23.3 AREA CLASSIFICATIONS

Based on experience decommissioning RDS 112 cyclotron facilities, facility areas have been classified as impacted areas or non-impacted areas.

23.3.1 Non-Impacted Area

Non-impacted areas are areas without residual radioactivity from licensed activities and are not surveyed during final status surveys. Non-impacted areas include areas outside the cyclotron facility and building structural surfaces above a 2-meter height.

23.3.2 Impacted Areas

Impacted areas are those areas that have potential residual radioactivity from licensed activities. Impacted areas are subdivided into Class 1, Class 2 or Class 3 areas. Class 1 areas have the greatest potential for contamination and therefore receive the highest degree of survey effort for the final status survey using a graded approach, followed by Class 2, and then by Class 3. Impacted sub-classifications are defined as follows:

- Class 1 Area: Areas with the highest potential for contamination, and meet the following criteria: (1) impacted; (2) potential for delivering a dose above the release criterion; (3) potential for small areas of elevated activity; and (4) insufficient evidence to support classification as Class 2 or Class 3.
- Class 2 Area: Areas that meet the following criteria: (1) impacted; (2) low potential for delivering a dose above the release criterion; and (3) little or no potential for small areas of elevated activity.
- Class 3 Area: Areas that meet the following criteria: (1) impacted; (2) little or no potential for delivering a dose above the release criterion; and (3) little or no potential for small areas of elevated activity.

Initial area classifications are:

- Cyclotron Room Lower Surfaces (< 2 meter height) – Class 1
- Impacted rooms other than the Cyclotron Room - Class 3

23.4 SURVEY UNITS

A survey unit is a geographical area of specified size and shape for which a separate decision will be made whether or not that area meets the release criteria. A survey unit is normally a portion of a building or site that is surveyed, evaluated, and released as a single unit. Areas of similar construction and composition will be grouped together as survey units and tested individually against the DCGLs and the null hypothesis to show compliance with the release criteria. Survey units will be homogeneous in construction, contamination potential, and contamination distribution.

The number of discrete sampling locations needed to determine if a uniform level of residual radioactivity exists within a survey unit does not depend on the survey unit size. However, the sampling density should reflect the potential for small, elevated areas of residual radioactivity. Survey units will be sized according to the potential for small, elevated areas of residual radioactivity. Recommended maximum survey unit sizes for building structures, based on floor area, is as follows:

- Class 1: up to 100 m^2
- Class 2: 100 m^2 to 1000 m^2
- Class 3: no limit

23.5 SURFACE SCANS

Scanning is used to identify locations within the survey unit that exceed the investigation level. These locations are marked and receive additional investigations to determine the concentration, area, and extent of the contamination. For Class 1 areas, scanning surveys are designed to detect small areas of elevated activity that are not detected by the measurements using the systematic pattern. The percentage of accessible building structural surfaces to be scanned based on classification is:

- Class 1: 100%
- Class 2: 50%
- Class 3: 10%

The percentage of survey area scan surveyed may be increased based on suspected elevated activity. For Class 2 and Class 3 areas, the surfaces to be scan surveyed will be those with the highest potential to contain residual contamination.

If elevated activity is detected during the scan surveys, then the location shall be marked and total (or dose rate) and removable surface activity measurements and will be taken to quantify the activity. However, total surface activity measurements (or dose rates) are in addition to the measurements required for the statistical test.

All survey units will be scanned using a gas flow proportional detector. Additionally, surfaces in the Cyclotron Room will be scanned for volumetric activation products using a 2" x 2" NaI detector.

23.6 TOTAL SURFACE ACTIVITY MEASUREMENTS

Direct surveys (static measurements) will be taken on building surfaces and system internals to the extent practical in impacted areas utilizing instrumentation of the best geometry based on the surface at the survey location. Additionally, locations of elevated activity identified and marked during the scan survey will require direct survey measurements.

23.7 DOSE RATE MEASUREMENTS

Dose rate measurements will be performed with a tissue equivalent Bicron MicroRem meter or a pressurized ion chamber (PIC). These instruments are selected due to their flat energy response. Additionally, a one-minute count will be performed at each location using a 2" x 2" sodium iodide detector.

Dose rate measurements will be taken on building surfaces in the cyclotron room with a potential for activation and in a background reference area. The background reference area will be selected outside the cyclotron room near concrete surfaces, yet away from the activated structures. Dose rate measurements will be taken at each cyclotron room calculated sample location, at areas of elevated activity identified during scans, and in the background reference area. At each location, a measurement will be taken at 1 m from the surface to determine the dose rate at the midpoint of a receptor.

23.8 SOLID SAMPLES

Solid samples of building structures will be collected and sent to an external laboratory for gamma spectroscopy analysis to confirm nuclide distributions for dose modeling calculations.

23.9 NUMBER OF SAMPLES

A minimum number of samples are needed to obtain sufficient statistical confidence that the conclusions drawn from the samples are correct. The number of samples will depend on the Relative Shift (the ratio of the concentration to be measured relative to the statistical variability of the contaminant concentration).

The minimum number of samples is obtained from MARSSIM tables or calculated using equations in Section 5 of MARSSIM.

23.9.1 Determination of the Relative Shift

The number of required samples will depend on the ratio involving the activity level to be measured relative to the variability in the concentration. The ratio to be used is called the Relative Shift, Δ/σ_S and is defined in MARSSIM as:

$$\Delta / \sigma_S = \frac{DCGL - LBGR}{\sigma_S}$$

Where:

DCGL = derived concentration guideline level

LBGR = concentration at the lower bound of the gray region. The LBGR is the average concentration to which the survey unit should be cleaned in order to have an acceptable probability of passing the test

σ_S = an estimate of the standard deviation of the residual radioactivity in the survey unit

The actual preliminary calculations are provided below:

Surface Activity (Static Measurements):

$$\Delta / \sigma_S = \frac{7{,}100 - 3{,}550}{1{,}000} = 3.6$$

Volumetric Activity (Dose Rates):

$$\Delta / \sigma_S = \frac{10 - 5}{2} = 2.5$$

Since MARRSIM Table 5.5 does not include relative shifts above 3 and the number of samples required decreases with an increasing relative shift, the relative shift for structural surface activity was conservatively set at 3.

23.9.2 Determination of Acceptable Decision Errors

A decision error is the probability of making an error in the decision on a survey unit by passing a unit that should fail (α decision error) or failing a unit that should pass (β decision error) The decision errors are 0.05 for both α and β errors.

23.9.3 Number of Data Points for Surface Activity (Sign Test)

The number of direct measurements for a particular survey unit, employing the Sign Test, is determined from MARSSIM Table 5.5, which is based on the following equation (MARSSIM equation 5-2):

$$N = \frac{\left(Z_{1-\alpha} + Z_{1-\beta}\right)^2}{4\left(SignP - 0.5\right)^2}$$

Where:

N	= number of samples needed in the survey unit
$Z_{1-\alpha}$	= percentile represented by the decision error α
$Z_{1-\beta}$	= percentile represented by the decision error β
SignP	= estimated probability that a random measurement will be less than the DCGL when the survey unit median is actually at the LBGR

Note: SignP is determined from MARSSIM Table 5.4

MARSSIM recommends increasing the calculated number of measurements by 20% to ensure sufficient power of the statistical tests and to allow for possible data losses. MARSSIM Table 5.5 values include an increase of 20% of the calculated value. The following calculations were made to determine this number:

$$N = \frac{\left(1.645 + 1.645\right)^2}{4\left(0.998650 - 0.5\right)^2} = 11$$

$Z_{1-\alpha}$ and $Z_{1-\beta}$ are equal to 1.645 using the error rate of 0.05 from MARSSIM Table 5.2. SignP is equal to 0.998650 from MARSSIM Table 5.4. Adding an additional 20% to account for data losses resulted in a value of 14. Therefore, the determined number of samples per survey unit for planning purposes is **14**.

23.9.4 Number of Data Points for Volumetric Activation (WRS Test)

The number of direct measurements for a survey unit and the background reference area, employing the WRS Test, is determined from MARSSIM Table 5.3, which is based on the following equation (MARSSIM equation 5-1):

$$N = \frac{\left(Z_{1-\alpha} + Z_{1-\beta}\right)^2}{3\left(P_r - 0.5\right)^2}$$

Where:

N = number of samples needed in the survey unit and in the background reference area

$Z_{1-\alpha}$ = percentile represented by the decision error α

$Z_{1-\beta}$ = percentile represented by the decision error β

P_r = probability that a random measurement from the survey unit exceeds a random measurement from the background reference area by less than the DCGL when the survey unit median is equal to the LBGR above background.

Note: P_r is determined from MARSSIM Table 5.1

MARSSIM recommends increasing the calculated number of measurements by 20% to ensure sufficient power of the statistical tests and to allow for possible data losses. MARSSIM Table 5.3 values include an increase of 20% of the calculated value. The following calculations were made to determine this number:

$$N = \frac{\left(1.645 + 1.645\right)^2}{3\left(0.961428 - 0.5\right)^2} = 17$$

$Z_{1-\alpha}$ and $Z_{1-\beta}$ are equal to 1.645 using the error rate of 0.05 from MARSSIM Table 5.2. P_r is equal to 0.961428 from MARSSIM Table 5.1. Adding an additional 20% to account for data losses resulted in a value of 21. This number is the total number of measurements for the survey unit and background reference area combined. Therefore, the determined number of samples for the survey unit and for the background reference area for planning purposes is **11**.

23.10 SAMPLE LOCATIONS

Determination of Class 1 survey unit sample locations is accomplished by first determining sample spacing and then systematically plotting the sample locations from a randomly generated start location. The random starting point of the grid provides an unbiased method for obtaining measurement locations to be used in the statistical tests. Class 1 survey units have the highest potential for small areas of elevated activity, so the areas between measurement locations may be adjusted to ensure that these areas can be detected by scanning techniques.

Similar systematic spacing methods are used for Class 2 and Class 3[4] survey units. The use of a systematic grid allows the decision-maker to draw conclusions about the size of the potential areas of elevated activity based on the area between measurement locations.

Sampling locations are established in a unique pattern beginning with the random start location and the determined sample spacing. After determining the number of samples needed in the survey unit, sample spacing is determined from MARSSIM equation 5-8:

$$L = \sqrt{\frac{A}{N}} \text{ for a square grid}$$

Where:

L = sample spacing interval

A = the survey unit area

N = number of samples needed in the survey unit

Maps will be generated of the survey unit's permanent surfaces included in the statistical tests (floors, walls, ceilings, fixed cabinetry, etc.). A random starting point will be determined using computer-generated random numbers coinciding with the x and y coordinates of the total survey unit. A square grid pattern will be plotted across the survey unit surfaces based on the random start point and the determined sample spacing. A measurement location will be plotted at each intersection of the grid plot.

[4] MARSSIM recommends random sampling (random x, random y) for Class 3 areas. However, in this survey design, Class 3 areas are sampled on a systematic grid pattern in the same manner as MARSSIM recommends for Class 1 and Class 2 areas.

23.11 REMOVABLE CONTAMINATION MEASUREMENTS

Removable contamination measurements (smears) will be collected on building structural surfaces at each sample location. Additionally, removable contamination measurements will be collected for building system internals. An area of approximately 100 cm^2 shall be wiped if possible. If an area of less than 100 cm^2 is wiped, a comment shall be added to the survey data sheet estimating the surface area wiped to allow for area correction of the results. Swabs may be used when system or component access points are not large enough to allow for disc smears.

23.12 SURVEYS OF BUILDING MECHANICAL SYSTEM INTERNALS

Surveys of various building system components will need to be performed. Survey design for these systems is out of the scope of MARSSIM. For the purposes of identifying potential residual contamination within these systems, the following survey protocol has been established. Surveys of ventilation exhausts will consist of scan surveys, total activity measurements, and removable contamination measurements of accessible ventilation exhaust points and at locations of potential collection/buildup. Removable contamination surveys of the internal surfaces of sink drains, sink drain traps and floor drains will be collected, since scan surveys and static measurements are not practical due to their small geometry.

23.13 SURVEY INVESTIGATION LEVELS

Investigation levels are used to flag locations that require special attention and further investigation to ensure areas are properly classified and adequate surveys are performed. These locations are marked and receive additional investigations to determine the concentration, area, and extent of the contamination. The survey investigation levels for each type of measurement are listed by classification in the table below.

Table 23-1: Survey Investigation Levels

Survey Unit Classification	Flag Direct Measurement or Sample Result When:	Flag Scanning Measurement Result When:	Flag Removable Measurement Result When:
Class 1	>50% of DCGL	>MDC	> 100 dpm/100 cm^2
Class 2	>50% of DCGL	>MDC	> 100 dpm/100 cm^2
Class 3	>MDC	>MDC	> 100 dpm/100 cm^2

24.0 INTERPRETATION OF SURVEY RESULTS

The statistical guidance contained in Section 8 of MARSSIM will be used to determine if areas are acceptable for unrestricted release, and whether additional surveys or sample measurements are needed.

24.1 DATA VALIDATION

Field data will be reviewed and validated to ensure:

- Completeness of forms and that the type of survey has correctly been assigned to the survey unit.
- The MDCs for measurements meet the established data quality objectives; independent calculations will be performed for a representative sample of data sheets and survey areas.
- Instrument calibrations and daily functional checks have been performed accurately and at the required frequency.

24.2 PRELIMINARY DATA REVIEW

A preliminary data review will be performed for each survey unit to identify any patterns, relationships, or potential anomalies. Additionally, measurement data are reviewed and compared with the DCGLs and investigation levels to identify areas of elevated activity and confirm the correct classification of survey units. If an area is misclassified with a less restrictive classification, the area will be upgraded and surveyed accordingly.

The following preliminary data reviews will be performed for each survey unit:

- Calculations of the survey unit mean, median, maximum, minimum, and standard deviation for each type of reading.
- Comparison of the actual standard deviation to the assumed standard deviation used for calculating the number of measurements. If the actual standard deviation is greater than estimated, the minimum number of samples shall be calculated using the actual standard deviation to ensure a sufficient number of samples have been obtained.
- Comparison of survey data with applicable investigation levels.

24.3 DETERMINING COMPLIANCE FOR SURFACE ACTIVITY

For Class 1 areas, if it is determined that all total activity results are less than the applicable DCGL, then the survey unit passes and no further statistical tests are required. If the average of the total activity results is above the $DCGL_W$, the

survey unit fails. If any total activity measurement is greater than the $DCGL_W$, and the average is less than the $DCGL_W$, the Sign Test is performed.

Table 24-1: Sign Test Summary of Statistical Tests

Survey Result	Conclusion
All measurements less than the DCGL	Survey unit meets release criterion
Average greater than DCGL	Survey unit does not meet release criterion
Any measurement greater than DCGL and the average less than DCGL	Conduct Sign Test

For Class 2 and Class 3 areas, data results are initially compared to the investigation levels. These investigation levels are provided to help ensure that survey units have been properly classified. If all data results in Class 2 or 3 areas are less than the investigation levels, then the survey unit is determined to meet the release criterion. If these investigation levels are exceeded, then an investigation is performed to verify the initial assumptions for classification and determine the appropriate resolution (e.g., additional scans or survey unit reclassification).

Removable contamination measurements will be compared directly to the applicable removable DCGL. No contingency is established for elevated removable contamination. Therefore, if any removable contamination is detected which exceeds the removable contamination limit, the survey unit is determined not to meet the release criterion. However, if all removable contamination measurements are less than the removable contamination limit, then compliance shall be determined based on total activity measurements.

24.4 DETERMINING COMPLIANCE FOR VOLUMETRIC ACTIVITY

For volumetric activity, the Wilcoxon Rank Sum (WRS) test is used to evaluate external dose rate measurements. If the highest result of any location inside the cyclotron room is less than $DCGL_w$ above the lowest result of any location from the background reference area, the survey unit passes and Wilcoxon Rank Sum (WRS) test is not required. If the survey unit average dose rate is more than the $DCGL_w$ above the background reference area average dose rate, the survey unit fails. If the survey unit average dose rate is less than the $DCGL_w$ above the background reference area average dose rate, the WRS Test will be performed.

Table 24-2: WRS Summary of Statistical Tests

Survey Result	Conclusion
Difference between largest survey unit measurement and smallest reference area measurement is less than DCGL	Survey unit meets release criterion
Difference of survey unit average and reference area average is greater than DCGL	Survey unit does not meet release criterion
Difference between any survey unit measurement and any reference area measurement greater than DCGL and the difference of survey unit average and reference area average is less than DCGL	Conduct WRS test

Removable contamination measurements will be compared directly to the applicable removable DCGL. No contingency is established for elevated removable contamination. Therefore, if any removable contamination is detected which exceeds the removable contamination limit, the survey unit is determined not to meet the release criterion. However, if all removable contamination measurements are less than the removable contamination limit, then compliance shall be determined based on dose rate measurements.

25.0 MECHANICAL SYSTEM SURVEY DATA ANALYSIS

If any measurement exceeds the applicable DCGL, then the survey unit does not meet the release criterion and is considered contaminated. Remediation or removal of the affected system components may be required. If all measurements are less than the applicable DCGL, then the system meets the release criterion and is considered releasable. Results of mechanical system surveys will be compared directly with the DCGL. This comparison will consider the applicable DCGL as a maximum value, rather than an average.

26.0 ALTERNATE SCENARIO ANALYSIS

The results of gamma spectroscopy analysis of reinforced concrete samples collected during remediation will be used to model the resultant doses from removal, recycling, and disposal of activated structures after unrestricted release of the facility. These analyses are in addition to the dose evaluations based on the building occupancy scenario and will be provided in the final status report.

27.0 FINAL STATUS REPORT

A Final Status Report summarizing project activities will be prepared and submitted to PETNET for review and approval. Subsequently, PETNET will submit the report to the NRC to request unrestricted release of the facility and license termination. The guidance provided in NUREG-1757 will be used to prepare the report.

28.0 SCHEDULE

On-site licensed activities are scheduled to occur shortly after NRC approval of this DP. On-site project activities are expected to take approximately three weeks.

29.0 REFERENCES

1. NRC Regulations
2. PETNET radioactive materials license numbers 41-32720-03 (Production) and license 41-32720-04MD (Medical Distribution)
3. PETNET Registration of non-medical radioactive material number IRM 145.
4. Chase Environmental Group Commonwealth of Kentucky radioactive materials license number 201-605-90
5. Chase Environmental Group Radiation Safety Manual
6. NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual" (MARSSIM)
7. NUREG-1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions"
8. NUREG-1757, Volume 1 "Consolidated NMSS Decommissioning Guidance," September, 2002

DandD Building Occupancy Scenario

DandD Version: 2.1.0
Run Date/Time: 8/27/2007 11:06:19 AM
Site Name: N/A
Description: DSV Determination
FileName:C:\Documents and Settings\Dave Culp\My Documents\Co-60 DSV.mcd

Options:

Implicit progeny doses NOT included with explicit parent doses
Nuclide concentrations are distributed among all progeny
Number of simulations: 100
Seed for Random Generation: 8718721
Averages used for behavioral type parameters

External Pathway is ON
Inhalation Pathway is ON
Secondary Ingestion Pathway is ON

Initial Activities:

Nuclide	Area of Contamination (m^2)	Distribution
60Co	UNLIMITED	CONSTANT(dpm/100 cm**2)
Justification for concentration: DSV Determination		Value 1.00E+00

Chain Data:

Number of chains: **1**

Chain No. 1: **60Co**
Nuclides in chain: **1**

Nuclide	Chain Position	Half Life	First Parent	Fractional Yield	Second Parent	Fractional Yield	Ingestion CEDE Factor (Sv/Bq)	Inhalation CEDE Factor (Sv/Bq)	Surface Dose Rate Factor ((Sv/d)/(Bq/m^2))	15 cm Dose Rate Factor ((Sv/d)/(Bq/m^3))
60Co	1	1.93E+03					7.28E-09	5.91E-08	2.03E-10	6.26E-12

Initial Concentrations:

Note: All reported values are the upper bound of the symmetric 95% confidence interval for the 0.9 quantile value

Nuclide	Surface Concentration (dpm/100 cm**2)
60Co	1.00E+00

Model Parameters:

General Parameters:

Parameter Name	Description	Distribution
To:Time In Building	The time in the building during the occupancy period	CONSTANT(hr/week)
Default value used		Value 4.50E+01
Tto:Occupancy Period	The duration of the occupancy exposure period	CONSTANT(days)
Default value used		Value 3.65E+02
Vo:Breathing Rate	The average volumetric breathing rate during building occupancy for an 8-hour work day	CONSTANT(m**3/hr)
Default value used		Value 1.40E+00
RFo*:Resuspension Factor	Effective resuspension factor during the occupancy period = RFo * Fl	DERIVED(1/m)
Default value used		
GO*:Ingestion Rate	Effective secondary ingestion transfer rate of removable surface activity from building surfaces to the mouth during building occupancy = GO * Fl	DERIVED(m**2/hr)
Default value used		
Tstart:Start Time	The start time of the scenario in days	CONSTANT(days)
Default value used		Value 0.00E+00
Tend:End Time	The ending time of the scenario in days	CONSTANT(days)
Default value used		Value 3.65E+02
dt:Time Step Size	The time step size	CONSTANT(days)
Default value used		Value 3.65E+02
Pstep:Print Step Size	The time steps for the history file. Doses will be written to the history file every n time steps	CONSTANT(none)
Default value used		Value 1.00E+00
AOExt:External Exposure Area	Minimum surface area to which occupant is exposed via external radiation during occupancy period	CONSTANT(m**2)
Default value used		Value 1.00E+01
AOInh:Inhalation Exposure Area	Minimum surface area to which occupant is exposed via inhalation during occupancy period	CONSTANT(m**2)
Default value used		Value 1.00E+01
AOIng:Secondary Ingestion Exposure Area	Minimum surface area to which occupant is exposed via secondary ingestion during occupancy period	CONSTANT(m**2)
Default value used		Value 1.00E+01
AO:Exposure Area	Minimum surface area to which occupant is exposed during the occupancy period	DERIVED(m**2)
Default value used		
Fl:Loose Fraction	Fraction of surface contamination available for resuspension and ingestion	CONSTANT(none)
Default value used		Value 1.00E-01
Rfo:Loose Resuspension Factor	Resuspension factor for loose contamination	CONTINUOUS LOGARITHMIC(1/m)
Default value used		Value / Probability 9.12E-06 / 0.00E+00 1.10E-04 / 7.67E-01 1.46E-04 / 9.09E-01 1.62E-04 / 9.50E-01 1.85E-04 / 9.90E-01 1.90E-04 / 1.00E+00
GO:Loose Ingestion	The secondary ingestion transfer rate of loose removable surface activity from	

Rate	building surfaces to the mouth during building occupancy	CONSTANT(m**2/hr)
Default value used		Value 1.10E-04

Correlation Coefficients:

None

Summary Results:

90.00% **of the** 100 **calculated TEDE values are <** 3.55E-03 **mrem/year .**
The 95 % Confidence Interval for the 0.9 **quantile value of TEDE is** 3.50E-03 **to** 3.61E-03 **mrem/year**

Detailed Results:

Note: All reported values are the upper bound of the symmetric 95% confidence interval for the 0.9 **quantile value**

Concentration at Time of Peak Dose:

Nuclide	Surface Concentration (dpm/100 cm**2)
60Co	9.37E-01

Pathway Dose from All Nuclides (mrem)

All Pathways Dose	External	Inhalation	Secondary Ingestion
3.61E-03	3.09E-03	4.87E-04	2.93E-05

Radionuclide Dose through All Active Pathways (mrem)

Nuclide	All Pathways Dose
60Co	3.61E-03
All Nuclides	3.61E-03

Dose from Each Nuclide through Each Active Pathway (mrem)

Nuclide	External	Inhalation	Secondary Ingestion
60Co	3.09E-03	4.87E-04	2.93E-05

DandD Building Occupancy Scenario

DandD Version: 2.1.0
Run Date/Time: 1/18/2009 3:06:24 PM
Site Name: N/A
Description: DSV Determination
FileName:C:\Documents and Settings\Dave Culp\My Documents\Cs-134 DSV.mcd

Options:

Implicit progeny doses NOT included with explicit parent doses
Nuclide concentrations are distributed among all progeny
Number of simulations: 100
Seed for Random Generation: 8718721
Averages used for behavioral type parameters

External Pathway is ON
Inhalation Pathway is ON
Secondary Ingestion Pathway is ON

Initial Activities:

Nuclide	Area of Contamination (m^2)	Distribution
134Cs	UNLIMITED	CONSTANT(dpm/100 cm**2)
Justification for concentration: DSV Determination		Value 1.00E+00

Chain Data:

Number of chains: **1**

Chain No. 1: **134Cs**
Nuclides in chain: **1**

Nuclide	Chain Position	Half Life	First Parent	Fractional Yield	Second Parent	Fractional Yield	Ingestion CEDE Factor (Sv/Bq)	Inhalation CEDE Factor (Sv/Bq)	Surface Dose Rate Factor ((Sv/d)/(Bq/m^2))	15 cm Dose Rate Factor ((Sv/d)/(Bq/m^3))
134Cs	1	7.53E+02					1.98E-08	1.25E-08	1.31E-10	3.86E-12

Initial Concentrations:

Note: All reported values are the upper bound of the symmetric 95% confidence interval for the 0.9 quantile value

Nuclide	Surface Concentration (dpm/100 cm**2)
134Cs	1.00E+00

Model Parameters:

General Parameters:

Parameter Name	Description	Distribution
To:Time In Building	The time in the building during the occupancy period	CONSTANT(hr/week)
Default value used		Value 4.50E+01
Tto:Occupancy Period	The duration of the occupancy exposure period	CONSTANT(days)
Default value used		Value 3.65E+02
Vo:Breathing Rate	The average volumetric breathing rate during building occupancy for an 8-hour work day	CONSTANT(m**3/hr)
Default value used		Value 1.40E+00
RFo*:Resuspension Factor	Effective resuspension factor during the occupancy period = RFo * Fl	DERIVED(1/m)
Default value used		
GO*:Ingestion Rate	Effective secondary ingestion transfer rate of removable surface activity from building surfaces to the mouth during building occupancy = GO * Fl	DERIVED(m**2/hr)
Default value used		
Tstart:Start Time	The start time of the scenario in days	CONSTANT(days)
Default value used		Value 0.00E+00
Tend:End Time	The ending time of the scenario in days	CONSTANT(days)
Default value used		Value 3.65E+02
dt:Time Step Size	The time step size	CONSTANT(days)
Default value used		Value 3.65E+02
Pstep:Print Step Size	The time steps for the history file. Doses will be written to the history file every n time steps	CONSTANT(none)
Default value used		Value 1.00E+00
AOExt:External Exposure Area	Minimum surface area to which occupant is exposed via external radiation during occupancy period	CONSTANT(m**2)
Default value used		Value 1.00E+01
AOInh:Inhalation Exposure Area	Minimum surface area to which occupant is exposed via inhalation during occupancy period	CONSTANT(m**2)
Default value used		Value 1.00E+01
AOIng:Secondary Ingestion Exposure Area	Minimum surface area to which occupant is exposed via secondary ingestion during occupancy period	CONSTANT(m**2)
Default value used		Value 1.00E+01
AO:Exposure Area	Minimum surface area to which occupant is exposed during the occupancy period	DERIVED(m**2)
Default value used		
Fl:Loose Fraction	Fraction of surface contamination available for resuspension and ingestion	CONSTANT(none)
Default value used		Value 1.00E-01
Rfo:Loose Resuspension Factor	Resuspension factor for loose contamination	CONTINUOUS LOGARITHMIC(1/m)
Default value used		Value / Probability 9.12E-06 / 0.00E+00 1.10E-04 / 7.67E-01 1.46E-04 / 9.09E-01 1.62E-04 / 9.50E-01 1.85E-04 / 9.90E-01 1.90E-04 / 1.00E+00
GO:Loose Ingestion	The secondary ingestion transfer rate of loose removable surface activity from	

Rate	building surfaces to the mouth during building occupancy	CONSTANT(m**2/hr)
Default value used		Value 1.10E-04

Correlation Coefficients:

None

Summary Results:

90.00% **of the** 100 **calculated TEDE values are <** 1.96E-03 **mrem/year .**
The 95 % Confidence Interval for the 0.9 **quantile value of TEDE is** 1.95E-03 **to** 1.97E-03 **mrem/year**

Detailed Results:

Note: All reported values are the upper bound of the symmetric 95% confidence interval for the 0.9 **quantile value**

Concentration at Time of Peak Dose:

Nuclide	Surface Concentration (dpm/100 cm**2)
134Cs	8.49E-01

Pathway Dose from All Nuclides (mrem)

All Pathways Dose	External	Inhalation	Secondary Ingestion
1.97E-03	1.81E-03	9.34E-05	7.21E-05

Radionuclide Dose through All Active Pathways (mrem)

Nuclide	All Pathways Dose
134Cs	1.97E-03
All Nuclides	1.97E-03

Dose from Each Nuclide through Each Active Pathway (mrem)

Nuclide	External	Inhalation	Secondary Ingestion
134Cs	1.81E-03	9.34E-05	7.21E-05

DandD Building Occupancy Scenario

DandD Version: 2.1.0
Run Date/Time: 8/27/2007 11:10:30 AM
Site Name: N/A
Description: DSV Determination
FileName:C:\Documents and Settings\Dave Culp\My Documents\Eu-152 DSV.mcd

Options:

Implicit progeny doses NOT included with explicit parent doses
Nuclide concentrations are distributed among all progeny
Number of simulations: 100
Seed for Random Generation: 8718721
Averages used for behavioral type parameters

External Pathway is ON
Inhalation Pathway is ON
Secondary Ingestion Pathway is ON

Initial Activities:

Nuclide	Area of Contamination (m^2)	Distribution
152Eu	UNLIMITED	CONSTANT(dpm/100 cm**2)
Justification for concentration: DSV Determination		Value 1.00E+00

Chain Data:

Number of chains: **1**

Chain No. 1: **152Eu**
Nuclides in chain: **2**

Nuclide	Chain Position	Half Life	First Parent	Fractional Yield	Second Parent	Fractional Yield	Ingestion CEDE Factor (Sv/Bq)	Inhalation CEDE Factor (Sv/Bq)	Surface Dose Rate Factor ((Sv/d)/(Bq/m^2))	15 cm Dose Rate Factor ((Sv/d)/(Bq/m^3))
152Eu	1	4.87E+03					1.75E-09	5.97E-08	9.53E-11	2.78E-12
152Gd	2	3.94E+16	1	0.2792			4.34E-08	1.01E-06	0.00E+00	0.00E+00

Initial Concentrations:

Note: All reported values are the upper bound of the symmetric 95% confidence interval for the 0.9 quantile value

Nuclide	Surface Concentration (dpm/100 cm**2)
152Eu	1.00E+00
152Gd	0.00E+00

Model Parameters:

General Parameters:

Parameter Name	Description	Distribution
To:Time In Building	The time in the building during the occupancy period	CONSTANT(hr/week)
Default value used		Value 4.50E+01
Tto:Occupancy Period	The duration of the occupancy exposure period	CONSTANT(days)
Default value used		Value 3.65E+02
Vo:Breathing Rate	The average volumetric breathing rate during building occupancy for an 8-hour work day	CONSTANT(m**3/hr)
Default value used		Value 1.40E+00
RFo*:Resuspension Factor	Effective resuspension factor during the occupancy period = RFo * Fl	DERIVED(1/m)
Default value used		
GO*:Ingestion Rate	Effective secondary ingestion transfer rate of removable surface activity from building surfaces to the mouth during building occupancy = GO * Fl	DERIVED(m**2/hr)
Default value used		
Tstart:Start Time	The start time of the scenario in days	CONSTANT(days)
Default value used		Value 0.00E+00
Tend:End Time	The ending time of the scenario in days	CONSTANT(days)
Default value used		Value 3.65E+02
dt:Time Step Size	The time step size	CONSTANT(days)
Default value used		Value 3.65E+02
Pstep:Print Step Size	The time steps for the history file. Doses will be written to the history file every n time steps	CONSTANT(none)
Default value used		Value 1.00E+00
AOExt:External Exposure Area	Minimum surface area to which occupant is exposed via external radiation during occupancy period	CONSTANT(m**2)
Default value used		Value 1.00E+01
AOInh:Inhalation Exposure Area	Minimum surface area to which occupant is exposed via inhalation during occupancy period	CONSTANT(m**2)
Default value used		Value 1.00E+01
AOIng:Secondary Ingestion Exposure Area	Minimum surface area to which occupant is exposed via secondary ingestion during occupancy period	CONSTANT(m**2)
Default value used		Value 1.00E+01
AO:Exposure Area	Minimum surface area to which occupant is exposed during the occupancy period	DERIVED(m**2)
Default value used		
Fl:Loose Fraction	Fraction of surface contamination available for resuspension and ingestion	CONSTANT(none)
Default value used		Value 1.00E-01
Rfo:Loose Resuspension Factor	Resuspension factor for loose contamination	CONTINUOUS LOGARITHMIC(1/m)
Default value used		Value / Probability 9.12E-06 / 0.00E+00 1.10E-04 / 7.67E-01 1.46E-04 / 9.09E-01 1.62E-04 / 9.50E-01 1.85E-04 / 9.90E-01 1.90E-04 / 1.00E+00

GO:Loose Ingestion Rate	The secondary ingestion transfer rate of loose removable surface activity from building surfaces to the mouth during building occupancy	CONSTANT(m**2/hr)
Default value used		Value 1.10E-04

Correlation Coefficients:

None

Summary Results:

90.00% **of the** 100 **calculated TEDE values are <** 1.97E-03 **mrem/year .**
The 95 % Confidence Interval for the 0.9 **quantile value of TEDE is** 1.92E-03 **to** 2.03E-03 **mrem/year**

Detailed Results:

Note: All reported values are the upper bound of the symmetric 95% confidence interval for the 0.9 **quantile value**

Concentration at Time of Peak Dose:

Nuclide	Surface Concentration (dpm/100 cm**2)
152Eu	9.74E-01
152Gd	8.82E-16

Pathway Dose from All Nuclides (mrem)

All Pathways Dose	External	Inhalation	Secondary Ingestion
2.03E-03	1.51E-03	5.12E-04	7.32E-06

Radionuclide Dose through All Active Pathways (mrem)

Nuclide	All Pathways Dose
152Eu	2.03E-03
152Gd	8.00E-18
All Nuclides	2.03E-03

Dose from Each Nuclide through Each Active Pathway (mrem)

Nuclide	External	Inhalation	Secondary Ingestion
152Eu	1.51E-03	5.12E-04	7.32E-06
152Gd	0.00E+00	7.84E-18	1.64E-19

DandD Building Occupancy Scenario

DandD Version: 2.1.0
Run Date/Time: 8/27/2007 11:14:10 AM
Site Name: N/A
Description: DSV Determination
FileName:C:\Documents and Settings\Dave Culp\My Documents\Eu-154 DSV.mcd

Options:

Implicit progeny doses NOT included with explicit parent doses
Nuclide concentrations are distributed among all progeny
Number of simulations: 100
Seed for Random Generation: 8718721
Averages used for behavioral type parameters

External Pathway is ON
Inhalation Pathway is ON
Secondary Ingestion Pathway is ON

Initial Activities:

Nuclide	Area of Contamination (m^2)	Distribution
154Eu	UNLIMITED	CONSTANT(dpm/100 cm**2)
Justification for concentration: DSV Determination		Value 1.00E+00

Chain Data:

Number of chains: **1**

Chain No. 1: **154Eu**
Nuclides in chain: **1**

Nuclide	Chain Position	Half Life	First Parent	Fractional Yield	Second Parent	Fractional Yield	Ingestion CEDE Factor (Sv/Bq)	Inhalation CEDE Factor (Sv/Bq)	Surface Dose Rate Factor ((Sv/d)/(Bq/m^2))	15 cm Dose Rate Factor ((Sv/d)/(Bq/m^3))
154Eu	1	3.21E+03					2.58E-09	7.73E-08	1.02E-10	3.04E-12

Initial Concentrations:

Note: All reported values are the upper bound of the symmetric 95% confidence interval for the 0.9 quantile value

Nuclide	Surface Concentration (dpm/100 cm**2)
154Eu	1.00E+00

Model Parameters:

General Parameters:

Parameter Name	Description	Distribution
To:Time In Building	The time in the building during the occupancy period	CONSTANT(hr/week)
Default value used		Value 4.50E+01
Tto:Occupancy Period	The duration of the occupancy exposure period	CONSTANT(days)
Default value used		Value 3.65E+02
Vo:Breathing Rate	The average volumetric breathing rate during building occupancy for an 8-hour work day	CONSTANT(m**3/hr)
Default value used		Value 1.40E+00
RFo*:Resuspension Factor	Effective resuspension factor during the occupancy period = RFo * Fl	DERIVED(1/m)
Default value used		
GO*:Ingestion Rate	Effective secondary ingestion transfer rate of removable surface activity from building surfaces to the mouth during building occupancy = GO * Fl	DERIVED(m**2/hr)
Default value used		
Tstart:Start Time	The start time of the scenario in days	CONSTANT(days)
Default value used		Value 0.00E+00
Tend:End Time	The ending time of the scenario in days	CONSTANT(days)
Default value used		Value 3.65E+02
dt:Time Step Size	The time step size	CONSTANT(days)
Default value used		Value 3.65E+02
Pstep:Print Step Size	The time steps for the history file. Doses will be written to the history file every n time steps	CONSTANT(none)
Default value used		Value 1.00E+00
AOExt:External Exposure Area	Minimum surface area to which occupant is exposed via external radiation during occupancy period	CONSTANT(m**2)
Default value used		Value 1.00E+01
AOInh:Inhalation Exposure Area	Minimum surface area to which occupant is exposed via inhalation during occupancy period	CONSTANT(m**2)
Default value used		Value 1.00E+01
AOIng:Secondary Ingestion Exposure Area	Minimum surface area to which occupant is exposed via secondary ingestion during occupancy period	CONSTANT(m**2)
Default value used		Value 1.00E+01
AO:Exposure Area	Minimum surface area to which occupant is exposed during the occupancy period	DERIVED(m**2)
Default value used		
Fl:Loose Fraction	Fraction of surface contamination available for resuspension and ingestion	CONSTANT(none)
Default value used		Value 1.00E-01
Rfo:Loose Resuspension Factor	Resuspension factor for loose contamination	CONTINUOUS LOGARITHMIC(1/m)
Default value used		Value / Probability 9.12E-06 / 0.00E+00 1.10E-04 / 7.67E-01 1.46E-04 / 9.09E-01 1.62E-04 / 9.50E-01 1.85E-04 / 9.90E-01 1.90E-04 / 1.00E+00
GO:Loose Ingestion	The secondary ingestion transfer rate of loose removable surface activity from	

Rate	building surfaces to the mouth during building occupancy	CONSTANT(m**2/hr)
Default value used		Value 1.10E-04

Correlation Coefficients:

None

Summary Results:

90.00% **of the** 100 **calculated TEDE values are <** 2.18E-03 **mrem/year .**
The 95 % Confidence Interval for the 0.9 **quantile value of TEDE is** 2.12E-03 **to** 2.26E-03 **mrem/year**

Detailed Results:

Note: All reported values are the upper bound of the symmetric 95% confidence interval for the 0.9 **quantile value**

Concentration at Time of Peak Dose:

Nuclide	Surface Concentration (dpm/100 cm**2)
154Eu	9.62E-01

Pathway Dose from All Nuclides (mrem)

All Pathways Dose	External	Inhalation	Secondary Ingestion
2.26E-03	1.59E-03	6.54E-04	1.06E-05

Radionuclide Dose through All Active Pathways (mrem)

Nuclide	All Pathways Dose
154Eu	2.26E-03
All Nuclides	2.26E-03

Dose from Each Nuclide through Each Active Pathway (mrem)

Nuclide	External	Inhalation	Secondary Ingestion
154Eu	1.59E-03	6.54E-04	1.06E-05