SECTION 3

PROTECTION OF WORKERS, THE PUBLIC, AND THE ENVIRONMENT AND RADIOACTIVE WASTE MANAGEMENT

Mallinckrodt Inc.

Phase 1 Plan
For C-T Decommissioning

NRC Docket: 40-06563
NRC License : STB-401

January 9, 2002

3. PROTECTION OF WORKERS, THE PUBLIC AND THE ENVIRONMENT AND RADIOACTIVE WASTE MANAGEMENT 3-i

3. PROTECTION OF WORKERS, THE PUBLIC AND THE ENVIRONMENT AND RADIOACTIVE WASTE MANAGEMENT

3.1 HEALTH AND SAFETY PROGRAM

This section describes the measures to protect workers, the public, and the environment during remediation. A Health and Safety Program for the C-T Project will be developed to ensure the safety of all contractors and Mallinckrodt employees, visitors, and members xof the public during decommissioning. In recognition that both the amount of radioactivity and the general safety hazards will be reduced as the project progresses, the Health and Safety Program may be modified to be commensurate with the activities being performed. Mallinckrodt will review and approve the Health and Safety Program, and any revisions that are made during the project. Any such adjustment to the requirements of this health and safety program shall be made in accordance with §2.6 herein.

The Health and Safety Program will consist of the following three parts:

- Industrial Safety Program
- Radiation Protection Program
- Environmental Safety Program

3.2 INDUSTRIAL SAFETY PROGRAM

An industrial safety program will be adopted for the C-T Project that will augment the existing Mallinckrodt Health, Safety and Environmental Guidelines by adding procedures specific to decommissioning, if required. The total package of procedures and administration will then form the Industrial Safety Program for the C-T Project. Table 3-1 lists typical industrial safety procedures that will be used for the planned remediation activities.

Table 3-1 **Industrial Safety Procedures**
Accident Investigation, Reporting, and Recordkeeping
Safety Color Code for Marking Physical Hazards
Working with Hazardous Chemicals
Welding and Thermal Cutting
Posting Requirements
Safety Training
Selection and Use of Personal Protective Equipment
Aerial Work Platforms
Scaffolds
Housekeeping
Equipment Lockout/Tagout
Operation of Lifting and Handling Equipment

Table 3-1 (cont'd)
Hand and Portable Power Tools
Electrical
Fall Protection
Guarding Floor Holes and Openings, Wall Holes and Openings
Fire Protection Program
Permit-Required Confined Space Entry Program
Excavation, Trenching, and Shoring
Drum Handling Procedure
Operation of Motorized Vehicles and Mechanized Equipment
Handling, Use, and Storage of Compressed Gas Cylinders
Hot Work
Heat Stress Program

3.2.1 Industrial Safety Training

The St. Louis Plant site-wide industrial safety program will be used for training all unescorted individuals involved in decommissioning activities at the C-T project. The purpose of the program is to promote an awareness of the potential risks, and to provide knowledge and proficiency in industrial safety consistent with the assigned tasks. Training takes place on a continuing basis.

Personnel involved in the C-T Project will be trained to perform their assigned responsibilities safely. On-the-job training and equipment-specific training will supplement the Mallinckrodt site-wide training program. Training in the proper use of specialized equipment is given before the individual uses that equipment. Credit may be given for applicable training received off-site.

The primary objectives of the industrial safety training program for the C-T Project:

- provide information on the industrial safety and hygiene hazards associated with working at the C-T Project ,and the steps to be taken to provide a safe work environment including those hazards unique to building demolition;

- enable each person to comply with plant rules and respond properly to warnings and alarms under normal and accidental conditions; and,

- enable individuals to recognize potential site specific hazards and to take appropriate measures to prevent personal injury or damage to facilities and equipment

The industrial safety training program will be reviewed and revised as needed to meet changing conditions and ensure that instructions are sufficiently well understood to permit practical application. The status and extent of the training of each individual will be documented to verify that workers are adequately trained for each assigned job.

The industrial safety training program includes:

- weekly shop-talks on pertinent industrial safety information, injury statistics, and specific safety issues;

- specific training on specialized equipment including the use of cranes, forklift trucks, front-end loaders, and scissors lifts;

- general industrial safety training including proper lifting, hearing conservation, eye protection, slips and falls, hazardous material handling, and use of power tools; and,

- specialized training including first aid, CPR, fire fighting, use of respirators, and HAZWOPER.

- safety work permits addressing confined space entry, asbestos removal, lock-out/tag-out, etc.

3.3 RADIATION PROTECTION PROGRAM

The Radiation Protection Program will consist of procedures to protect workers, the public, and the environment from ionizing radiation.

A radiation protection program will be adopted for the C-T Project that addresses the following topics. The contractor will be required to implement the program with oversight by the Site RSO.

- health and safety protection measures and policies

- instrumentation, calibration, and equipment

- use of air samplers, monitoring policy methods, frequency, and procedures

- contamination control and personnel decontamination

- external exposure control

- airborne releases and monitoring

- Safety Work Permits, including ALARA

- engineering controls

- transportation

- accident response

- posting and labeling
- records and reports
- potential sources of contamination exposure
- The accident analysis is presented in Attachment 6

3.3.1 Radiation Safety Training

All unescorted individuals involved in decommissioning activities for the C-T Project will be required to complete the Mallinckrodt radiation safety training course or the contractor equivalent course. The purpose of the training is to increase awareness of the potential radiation risks during decommissioning, and to provide a level of proficiency in personal radiation protective measures consistent with assigned tasks. On-the-job training, as deemed necessary by the contractor ES&H personnel, will be used to complement the formal radiation safety training.

All individuals will be trained before entering a controlled area to perform work. The safety performance of each individual will be reviewed annually, and workers will be retrained every two years. Credit may be given for applicable training received off-site, but plant-specific training is required for all decommissioning personnel. Training and examination results will be formally documented.

The primary objectives of the radiation safety training program are to comply with the instruction requirements of 10 CFR 19.

The radiation safety training will be reviewed and revised as appropriate to meet changing conditions and ensure that instructions are sufficiently well understood to permit practical application.

The radiation safety training program includes the following topics:

- radiation fundamentals - basic characteristics of radiation and contamination
- radiation exposure limits, administrative control levels, and controls - external radiation exposure control methods, procedures, and equipment
- radiation contamination limits and controls - contamination and internal radiation exposure control methods, procedures, and equipment
- contaminated materials associated with decommissioning work - potential radiological problems

- radiological work planning - integrating radiation safety and operational requirements to ensure safe conduct of work
- emergency procedures and systems - work related information and actions
- biological effects of radiation - basic understanding of biological effects and methods of assessment
- the Radiation Protection Program
- workers rights and responsibilities
- radiation exposure reports which workers may request pursuant to 10 CFR 19.13
- ALARA

3.3.2 Radiation Protection Instrumentation

Instrumentation utilized for personnel monitoring will be calibrated and maintained in accordance with radiation safety procedures. These procedures utilize the manufacturers calibration guidance. Portable instruments are calibrated on a semi-annual basis or as required due to maintenance. Specific requirements for instrumentation include traceability to NIST standards, field checks for operability, background radioactivity checks, operation of instruments within established environmental bounds (i.e., temperature and pressure), training of individuals, scheduled performance checks, calibration with isotopes with energies similar to those to be measured, quality assurance tests, data review, and record keeping. Where applicable, activities of sources utilized for calibration are also corrected for decay. All calibration and source check records are completed, reviewed, signed off and retained in accordance with Quality Assurance Program requirements. A list of typical radiation instrumentation and minimum detectable activities (MDA) is given in Table 3-2. Typical personnel monitoring equipment is shown in Table 3-3. In the event an instrument of the type listed in Table 3-2 is employed on C-T decommissioning, its background count rate or exposure rate and its lower limit of detection will be estimated for its application. Alternative instrumentation must also be able to measure adequately to assess compliance with radiological protection requirements.

Table 3-2
Typical Instruments for Performing Radiation Protection Surveys

Instrument Type	Radiation Detected	Scale Range	BKG	Typical MDA 95% confidence Level
Scintillation (Ludlum 2224) Scaler/Ratemeter with 43-89 probe	Alpha Beta Beta	0-500,000 cpm	<10 cpm <300 cpm	100 dpm/100 cm^2 500 dpm/100 cm^2 4500 dpm/100 cm^2 (scan)
Micro-R Meter (Ludlum) 1" x 1" NaI Detector	Gamma	0-3,000 μR/h or 0-5,000 μR/h	7 μR/h	1-2 μR/h
Ion Chamber (Victoreen)	Gamma	0.1-300 mR/h	<0.1 mR/h	<0.2 mR/h
3" x ½" NaI Scintillation Detector Digital Scaler	Gamma	0-500,000 cpm	3,000 cpm avg shielded 9,000 cpm avg unshielded	250 cpm 500 cpm
435 cm^2 gas flow (43-27) Digital Scaler	Alpha	0-500,000 cpm	<10 cpm	20 dpm/100 cm^2
100 cm^2 gas flow (43-68) Digital Scaler	Alpha Beta Beta	0-500,000 cpm	<10 cpm <300 cpm	100 dpm/100 cm^2 500 dpm/100 cm^2 4500 dpm/100 cm^2 (scan)
60 cm^2 gas flow (43-4) Digital Scaler	Alpha	0-500,000 cpm	<10 cpm	200 dpm/100 cm^2
60 cm^2 Count Rate Meter (PRM-6)	Alpha	0-500,000 cpm	<100 cpm	350 dpm/100 cm^2
50 cm^2 Personnel Room Monitor (Ludlum 177)	Alpha	0-500,000 cpm	<100 cpm	500 dpm/100 cm^2
Ludlum 2" GM Tube (Pancake)	Beta Gamma	0-500,000 cpm 720 cpm = 0.2 mR/h	<200 cpm	70 cpm
Bicron AB-100 Scintillation Probe	Beta	0-500,000 cpm	<200 cpm	200 dpm/100 cm^2

Notes:

1) Instrument MDAs are based upon static measurements, one minute count times unless otherwise noted.
2) Instrument MDAs depend upon background.

Table 3-3
Typical Equipment for Performing Personnel Monitoring

Equipment Description	Purpose
Personal Air Samplers (BZ) Gillian or equivalent	Breathing zone air monitoring
Area Air Samplers SAIC or equiv.	High volume air monitoring
Area Air Samplers SAIC or equiv.	Work area low volume air monitoring
Personnel Dosimetry TLD or equiv.	Deep dose, eye dose, skin dose
Alpha Firsker Ludlum 43-68 or equiv.	Contamination monitoring
Alpha Frisker Ludlum 177	Contamination monitoring
Beta Frisker Bicron AB100	Contamination monitoring
Micro-R meter Ludlum or equiv.	Exposure rate
Ion Chamber	Dose rate

3.3.3 ALARA

An objective of radiation protection during decommissioning is to achieve as low as reasonable exposure to regulated radioactive material and radiation from it. The most effective emphasis will be to consider during preparation of each radiation safety work permit whether any particular action and or engineered control beyond good health physics practice would be reasonable to specify to try to reduce exposure. The radiation safety work permit form shall specify that ALARA be considered.

An ALARA analysis is presented in Attachment 5.

3.4 ENVIRONMENTAL SAFETY PROGRAM

An Environmental Safety Program will be developed and implemented as required to monitor air and water effluents discharged from the C-T Project during decommissioning. The program will be reviewed and approved by Mallinckrodt prior to implementation. Samples will be routinely collected or measurements routinely made at on-site and site boundary or off-site locations to determine the extent of environmental discharges during remediation. Monitoring locations will be chosen commensurate with remediation activities.

In recognition that both the amount of radioactivity and the general environmental hazards will be reduced as decommissioning progresses, the Environmental Safety Program can be modified to be commensurate with the activities being performed by following the procedure described in Section 2.6.

3.4.1 Effluent Air Monitoring

No effluent air monitoring is anticipated, since no point sources of effluent air are expected to exist. However, in the event a decontamination process exhaust ventilation or similar point discharge of potentially radioactive effluent air were employed, its effluent air would be sampled and analyzed for regulated radioactive particulate.

3.4.2 Environmental Air Monitoring

Environmental sampling stations will be provided during demolition or decontamination activities as required by 10 CFR Part 20 (Appendix B limits) to verify there are no adverse impacts to on-site workers and the public. Each environmental sampling station will be equipped with an air sampler.

Collection and analysis of the continuous air samples will be performed during demolition or decontamination activities as required by 10 CFR Part 20. The samples will be analyzed for gross alpha and gross beta activity to interpret the uranium and thorium series. The analytical instruments will be calibrated using standards traceable to the National Institute of Science & Technology (NIST).

3.4.3 Liquid Effluent Monitoring

It will be the policy of Mallinckrodt during the C-T Project to minimize the production of contaminated aqueous liquids. There are three possible sources of contaminated aqueous liquids: sink and shower water, decontamination fluids, and water used for dust suppression. Mallinckrodt expects sink and shower water to contain insignificant amount of regulated radioactive material in readily dispersible biological material, and thereby may be discharged to sanitary sewerage in accordance with 10 CFR Part 20.2003 without monitoring. Should rain water or surface water be collected, it will ordinarily be used for dust suppression of solid waste destined for NRC-approved disposal. In the event other aqueous waste potentially containing significant concentration of regulated radioactive material were considered for discharge to sewerage, Mallinckrodt would, beforehand, filter it to remove non-dispersible solids, sample and analyze it, estimate the concentration in sewage, compare it with the 10 CFR Part 20, Appendix B, Table 3, monthly average concentration limit, and estimate the total radioactivity inventory discharged.

3.4.4 Direct Radiation Monitoring

The environmental safety program is designed to assure that direct radiation in unrestricted areas does not exceed limits in 10 CFR 20.1301. The objective of direct radiation monitoring is to verify the effectiveness of the environmental safety program in meeting the limits.

The monitoring of penetrating radiation will be performed using standard environmental thermoluminescent dosimeters that are placed at various locations around the perimeter of the restricted remediation area. These dosimeters will be collected by Health and Safety personnel and analyzed quarterly by a qualified contract vendor to measure the integrated gamma dose for each location.

3.4.5 Action Levels

The following action levels will be established in procedures to aid in compliance with environmental safety regulations in 10 CFR 20.

Medium	Action Level (fraction of limit)	Regulation 10 CFR 20
Environmental air	≤ 0.75	App. B, Table 2, col 1
Effluent water	≤ 0.6	App. B, Table 2, col 2
Sewage	≤ 0.6	App. B, Table 3
Gamma radiation	≤ 0.5	Part 20.1301(a)(1)
Gamma radiation	≤ 0.5	Part 20.1301(a)(2)

If an action level is exceeded, the Mallinckrodt RSO and the contractor ES&H manager will be notified and corrective action will be implemented as appropriate. Investigation may include additional measurements or analysis to assess compliance with the regulation and to ensure that the total radiological dose from inhalation and irradiation by external gamma-rays does not exceed 100 mrem/yr.

3.5 SECURITY DURING DECOMMISSIONING

Extensive security is provided at the Mallinckrodt St. Louis Plant. The entire perimeter is fenced, and all entrances are controlled by card-readers or the site security force. Normal Mallinckrodt security will be maintained at the St. Louis Plant during decommissioning activities. In addition, areas undergoing remediation will be cordoned and posted as required to control personnel access and vehicle access to the portion of the site under remediation. Cordoning may be relocated as remediation progresses and will remain in place until the activities are completed and the area is surveyed and made safe for general access.

3.6 RADIOACTIVE WASTE MANAGEMENT

Radioactive waste from the C-T project decommissioning activities will be managed in accordance with the requirements of this C-T Project Waste Management Plan. This plan ensures that radioactive waste from the C-T project will be handled, stored and disposed of in accordance with applicable regulatory requirements.

3.6.1 Management of Radioactive Waste

Two general categories of radioactive waste will exist during decommissioning: equipment and building material.

- ***Equipment***
  Equipment, will be segregated based on planned disposition and or on contamination level. Equipment may be released in accordance with Section 2.2.

- ***Building Material***
  Building material, including roofing material, block, brick, structural steel, concrete, piping, wiring, and associated components, will be segregated based on planned disposition and or on contamination level. Building material will be released in accordance with Sections 2.2. Non-process building slabs may be surveyed and released in accordance with criteria in Section 2.2.

Decommissioning activities involving pavement and streets, soil, C-T process building slabs, and underground structures and utilities, will not be performed during Phase I decommissioning. The Phase II Plan will discuss surface/subsurface waste management.

3.6.2 Regulatory Requirements

Processing and disposal of radioactive waste will be performed in accordance with the relevant requirements of 10 CFR 20, 10 CFR 40, 10 CFR 71, DOT regulations, and 49 CFR 172-178 and the applicable disposal site waste acceptance criteria.

3.6.3 Projected Quantities of Radioactive Waste

Table 3-4 provides a summary level breakdown of the estimated volume of radioactive waste that is expected to be generated during decommissioning of the buildings and equipment of the C-T process and support areas.

Table 3-4 Estimated Phase I Waste Volumes (Buildings and Equipment)	
Type	**Volume (ft^3)**
Equipment	
Tanks, Pumps, Piping, etc.	38,280
Building Materials	
Reinforced Concrete	28,300
Brick	11,500
Concrete Block	11,000
Structural and other Steel	12,100
Doors, Windows and Roofing	19,200
Exterior, Interior Walls	5,000
Asbestos containing	500
Total Volume	125,880

The buildings and equipment have been characterized and the resulting wastes to be generated are projected to have average concentrations of natural uranium and natural thorium significantly below 0.05 weight percent. The resulting wastes to be generated will thus meet the unimportant quantity of source material as defined in 10 CFR 40.13.

In order to estimate relative radioactivity of long-lived, key radionuclides on buildings for the purpose of deriving maximum acceptable areal radioactivity density on buildings (DCGL) and equipment (FC 83-23, Table 1), characterization surveyors scanned for maximal radioactivity on building surfaces and, where it was found, sampled it by scraping or scabbling. Seventy-four samples were collected on interior and exterior surfaces of process, support, and peripheral buildings. An index was derived for each sample to enable it to be compared with the maximum acceptable "unimportant" radioactivity concentration corresponding to 0.05 wt% source material.[1] The index of each sample was derived by the equation

$$\text{Index} = \frac{U}{167} + \frac{Th}{55}$$

where U = uranium concentration in sample, the average of U^{238}, Th^{230}, and Ra^{226}, background not subtracted (pCi/g sample)

167 = radioactivity concentration equivalent of 0.05 wt% uranium (pCi/g)

Th = thorium concentration in sample, the average of Th^{232}, Ra^{226}, and Th^{228}, background not subtracted (pCi/g sample)

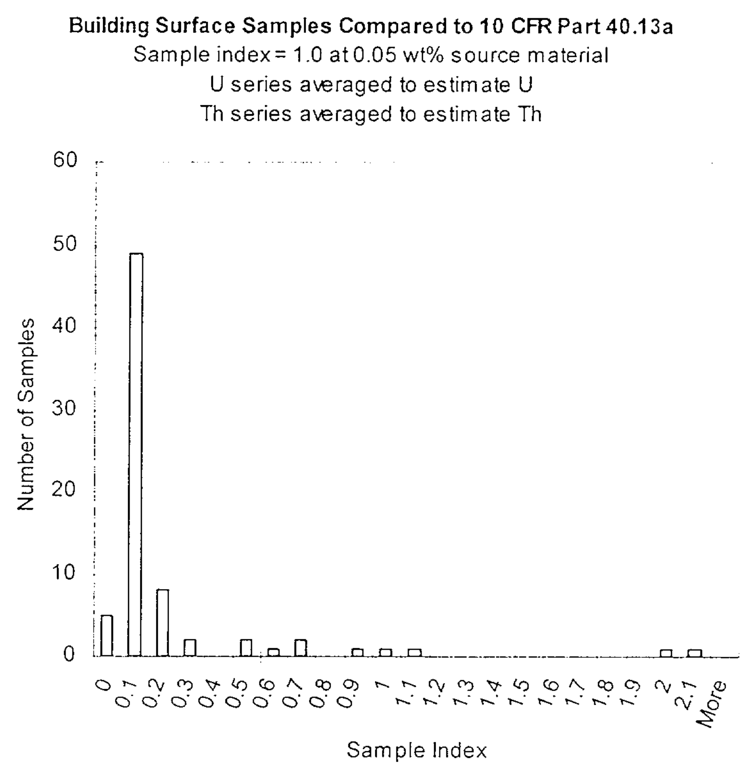
[1] 10 CFR Part 40.13a.

55 = radioactivity concentration equivalent of 0.05 wt% thorium (pCi/g)

Estimating U and Th concentration this way considers possible analytical inaccuracy, nearness to radioactive equilibrium, and importance of long-lived progeny.

A histogram of the indices of the 74 samples, Figure 3-1, reveals that the index of only 5 samples is more than 0.8 of the upper bound of an unimportant concentration, 0.05 wt% source material. Four of those samples were collected in CT process building 238. The other was collected in a laboratory hood in building 250. Since the shallow, surficial contamination is at most 0.1 wt% source material in a few samples, the dismantled building material, that is not reasonably separable, will contain an unimportant concentration, well below 0.05 wt% source material.

Figure 3-1



The bulk waste will be surveyed before disposition to confirm that it is an unimportant quantity. These samples and confirmatory measurements will demonstrate that source material concentration in bulk waste will be an unimportant quantity in accordance with 10 CFR Part 40.13(a). Therefore, classification for disposal under 10 Part 61.55 or characterization under Part 61.56 would not seem relevant. If it were, the waste would be Class A in accordance with 10 CFR Part 61.55(a)(6).

3.6.4 Waste Disposition

Equipment and other materials known to be contaminated will be disposed of by transfer to a licensed disposal facility, transfer to a disposal facility authorized to receive an unimportant quantity of source material, or will be decontaminated, surveyed and released under NRC-approved criteria.

3.6.4.1 Equipment

Most equipment and other items will be managed and disposed of at either a licensed disposal facility or a disposal facility authorized to receive an unimportant quantity of source material. In the event equipment or other item is to be released for unrestricted use and removal, it must first be surveyed and found compliant with Materials License STB-401, condition 16.

3.6.4.2 Building Material

In the event a building or other structure is to be demolished, before dismantlement, it will be decontaminated only to the extent necessary for health and safety control. The waste material will be characterized before release to a carrier for transport and for receipt by the disposal site operator. See Section 2.2 for the relevant release criteria.

3.6.5 Temporary On-Site Storage of Radioactive Waste Prior to Shipping

Contaminated material may be staged on-site temporarily to (a) stage material for sampling and analysis: (b) accumulate sufficient quantities for economical shipment; and (c) coordinate shipments between the carrier and the disposal site. Contaminated building material and equipment may be stored in designated staging areas.

Buildings and equipment have been characterized and the resulting wastes to be generated are projected to have average concentrations of natural uranium and natural thorium significantly below an exempted quantity of source material as defined in 10 CFR 40.13. Mallinckrodt anticipates that no more than about 20,000 cubic feet (approximately 750 cubic yards) of waste materials will be in temporary storage at any given time and for no longer than three months. Two locations currently envisioned to be utilized for storage are: 1) within the Plant 5 area, or 2) within the Plant 7 area as described on DP Figure 1-3 "C-T Production Process & Support Areas, Rev.1". Positive control is maintained in a two-fold manner: 1) An active 24-hour security system is in place for the entire Mallinckrodt facility (ref. §3.5, "Security During Decommissioning"), and 2) each temporary radioactive material storage area will be enclosed and/or roped-off and appropriately posted as required. It is expected that radiation levels at access points to temporary storage areas will be up to several times background, with the average being less that 50 μR/hr and the maximum less than 100 μR/hr. Thus, the low radiation level beside waste in storage will ensure compliance with 10 CFR 20.1301. In addition, appropriate training will be provided to workers regarding the waste materials temporarily stored on-site.

3.6.6 Mixed Waste

Characterization efforts performed to date have not identified any mixed wastes. Mallinckrodt does not anticipate that mixed waste will result from decommissioning efforts. If mixed waste were discovered, Mallinckrodt has a permit to manage hazardous waste on-site in accordance with a RCRA Part B permit with the State of Missouri. In the event mixed waste is identified during remediation activities, Mallinckrodt will characterize the wastes, identify a disposal method, assess the effect on the schedule, assess related disposal costs, modify handling procedures, as needed, and will notify the NRC

3.6.7 Records

Mallinckrodt will maintain records of waste material released from the C-T decommissioning area or controlled areas. The Administrative Controls Plan presents the record retention requirements in Attachment 2.

SECTION 4

PLANNED FINAL STATUS SURVEY

Mallinckrodt Inc.

Phase 1 Plan
For C-T Decommissioning

NRC Docket: 40-06563
NRC License : STB-401

January 9, 2002

4. PLANNED FINAL STATUS SURVEY

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This section will provide a summary of the planned Final Status Survey (FSS). This survey will demonstrate that equipment and building surfaces meet the criteria for unrestricted release.

Table 1-2,"C-T Process and Support Buildings" lists the buildings which contain equipment to be included in the final status survey. Section 1.6.4, "Current Radiological Status," further describes the current radiological condition of the equipment.

Removable equipment which will be released for unrestricted use will be surveyed in accordance with the *Guidelines*[1] and must meet both *average* and *maximum* areal density limits[2] in the *Guidelines*. Equipment which is located in buildings slated for demolition and equipment which is suspect or known to contain internal contamination will be disposed of in accordance with Section 2.2.4, and is not subject to FSS.

Installed apparatus or building components will be subject to the building surface release criteria in the survey unit in or on the building in or on which it is installed. Otherwise, it will be disposed in accordance with Section 2.2.4, and is not subject to FSS.

Table 1-5, "Disposition Of C-T Process And Support Area Buildings," lists the buildings which were used for the C-T Process and support areas or that are located in Plant 5. Section 1.6.4, "Current Radiological Status," further describes the current radiological status of the buildings. Appendix A, "Surface Survey Area Results," classifies these buildings according to contamination levels consistent with Section 4.4.1.2, "Building Surfaces". FSS results for all buildings other than those addressed under FUSRAP or demolished will be included in the Final Status Survey Report.

Buildings slated for demolition are listed in Table 1-5, "Disposition Of C-T Process And Support Area Buildings". A building to be demolished is not subject to a FSS.

To the extent practical, guidance from the Draft Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM), NUREG 1727 Appendix E, and NUREG-1505 has been incorporated in this Final Status Survey Plan. First, release criteria (denoted "derived concentration guideline level" or "DCGL" in MARSSIM) have been established and are described in Section 2.2, "Decommissioning Criteria," and referenced in Section 4.1, "Release Criteria." Next, proper instrumentation has been chosen and described in Section 4.2, "Instrumentation". The background determination is described in Section 4.3, "Background". This section describes prior activities to determine background

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1 NRC. "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material." Table 1. April 1993.

2 The average areal density limit may also be referred to as "maximum acceptable average areal density, or MAAAD. The maximum areal density limit may also be referred to as the maximum acceptable areal density, or MAAD.

values and the remaining activities to be performed. Finally, Section 4.4, "Survey Methodology" describes the approach which will be taken to perform the FSS.

Section 5 describes the Quality Assurance Program as it applies to the Final Status Survey.

4.1 RELEASE CRITERIA

Residual radioactive material release criteria (DCGL or MAAAD[3]) have been determined for buildings and equipment. Residual radioactive material is material which is regulated as a radioactive material and does not include naturally occurring radioactive material.

Equipment release criteria are based on the limits found in NRC "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material". A composite criterion is derived using the sum-of-fractions convention. Measurements to assess compliance with equipment release *Guidelines* will be interpreted in equivalent radiation units, *e.g.*, dpm $\alpha/(\text{min}\cdot 100\ \text{cm}^2)$ or dpm $\beta/(\text{min}\cdot 100\ \text{cm}^2)$. These derivations are described in Appendix H.

The building surface DCGL were derived using RESRAD-Build assuming a dose limit of 25 mrem/y to a worker assumed to be exposed as defined for the Worker-Building Occupation Scenario. This derivation is described in Appendix C, "Derivation of Nuclide Limits for Building Surfaces". As with equipment, radiation measurements will be interpreted in equivalent units as described in Appendix D for comparison with the derived limits. The $DCGL_W$ applicable to building surfaces and installed apparatus is specified in Section 2.2, " Decommissioning Criteria".

4.2 INSTRUMENTATION

An FSS will consist of scanning, direct stationary surveying, and analysis of material samples. Therefore, scanning instruments, direct measuring instruments and laboratory instruments will be used in conducting the final radiation surveys and sample analyses. All instruments will be appropriate for the type of survey and the concentration of radioactivity to be measured. QA/QC procedures to be used with instrumentation are 2described in Section 5.7, "Equipment Maintenance and Calibration."

Typical instrumentation for remediation and final surveys is listed in Tables 4-1 and 4-2. Other instrumentation may be used provided it meets quality objectives for calibration, operability,[4] and detection capability.[5] Table 4-1 lists the instrumentation to be used for survey activities, along with typical parameters and detection sensitivities for the

[3] *DCGL* ≡ derived concentration guideline level, for buildings
MAAAD ≡ maximum acceptable average areal density, for removable equipment

[4] DP §5.7 Equipment Maintenance and Calibration

[5] DP Appendix D, Lower Limit of Detection

instrumentation and survey technique. Fixed and actual MDC will be derived in accordance with MDC methodology in Appendix D to reflect conditions at the time a final status survey is conducted. The combination of instrumentation and technique will be chosen to provide a detection sensitivity to satisfy survey objectives. Sensitivities for scanning techniques are based on movement of the detector over the surface at one detector-width per second and use of audible indicators to sense changes in instrument count rate.

Table 4-1
Typical Instruments for Performing Final Radiation Status Surveys

Instrument Type	Radiation Detected	Scale Range	Typical Background	Typical MDA 95% confidence Level[2,3,4]	Usage
Scintillation (Ludlum 2224) Scaler/Ratemeter with 43-89 probe	Alpha Beta Beta	0-500,000 cpm	<10 cpm ≈750 cpm closed ≈1500 cpm open	100 dpm/100 cm^2 (direct) 1600 dpm/100 cm^2 (direct) 5100 dpm/100 cm^2 (scan)	General Characterization[5]; FSS[6]; Internal Surface-Q[7]
Micro-R Meter (Ludlum) 1" x 1" NaI Detector	Gamma	0-3,000 μR/h or 0-5,000 μR/h	7 μR/h	1-2 μR/h	General Characterization[5]; Internal Surface-Q[7]
3" x ½" NaI Scintillation Detector Digital Scaler	Gamma	0-500,000 cpm	2,500 cpm avg shielded[7] 7,000 cpm avg unshielded	250 cpm 500 cpm	General Characterization[5]; Internal Surface-NQ[7]
100 cm^2 gas flow (43-68) Digital Scaler[1]	Alpha Beta Beta	0-500,000 cpm	<10 cpm ≈750 cpm closed ≈1500 cpm open	100 dpm/100 cm^2 (direct) 1600 dpm/100 cm^2 (direct) 7100 dpm/100 cm^2 (scan)	General Characterization[5]; FSS[6]; Internal Surface-Q[7]
Ludlum 2" GM Tube (Pancake)	Beta Gamma	0-500,000 cpm 720 cpm = 0.2 mR/h	<200 cpm	2300 cpm	General Characterization[5]; Internal Surface-Q[7]
5" plastic scintillator w/ multi-channel analyzer	Beta	N.A.	TBD[1]	TBD[1]	General Characterization[5]
Bicron AB-100 Scintillation Probe	Alpha Beta Beta	0-500,000 cpm	<10 cpm ≈750 cpm closed ≈1500 cpm open	70 dpm/100 cm^2 (direct) 850 dpm/100 cm^2 (direct) 3900 dpm/100 cm^2 (scan)	General Characterization[5]; FSS[6]; Internal Surface-Q[7]

Notes:
[1] TBD = to be determined prior to initial use and calibration.
[2] Instrument MDA are based upon static measurements, one minute count times unless otherwise noted.
[3] Instrument MDA depends upon background.
[4] Release instrument (i.e., FC 83-23)
[5] Characterization instrument; may include non-quantitative uses
[6] FSS instrument, or may be used in support of FSS
[7] This denotes instruments that may be used for internal surveys of equipment. Q denotes quantitative. NQ denotes non-quantitative results which cannot be used as the only FSS method.

Direct measurements for gross α activity will be performed using a scintillation or gas proportional detector.

Direct surface measurements for gross beta activity will be performed using scintillation probes or gas proportional probes. Special consideration will be given to the use of large area probes on the order of 100 cm^2. Scanning shall be performed using scintillation probes or gas proportional probes such as the Bicron Surveyor M with A100 or B100 probes and gas proportional floor monitors for beta activity. Removable activity measurements for equipment shall be performed using a low background gas flow proportional counter, an instrument such as a Ludlum 2200 with a 43-10 detector or an AB-100 scintillation detector, or equivalent.

Table 4-2
Typical Laboratory Instruments for Performing Final Radiation Status Surveys

Instrument Type	Radiation Detected	Scale Range	Typical Background	Typical MDA 95% confidence Level[4,5]	Instrument Usage
Ludlum Model 2929	Alpha Beta	0-99,999,999 cpm	0.2 cpm 45 cpm	3 dpm 50 dpm	Free Release[2] General Characterization[3] Internal Surface-Q[6]
Tennelec LB5100 Computer Based Auto Sample Counter	Alpha Beta	0-99,999,999 cpm	<0.3 cpm 1.5 cpm	0.4 dpm 1.5 dpm	Free release[2] Environmental General Characterization[3] Internal Surface-Q[6]
Waste Counter – Computer Linked MCA 3" x 3" NaI (Tl) Detector	Gamma	-	TBD[1] pCi/g Total U TBD[1] pCi/g Th (Nat)	TBD[1] pCi/g U TBD[1] pCi/g Th (Nat)	Waste Characterization
5" Slide-Drawer ZnS Scintillation Counter	Alpha	0-500,000 cpm	<0.3 cpm	2 dpm	Free release[2] Environmental Internal Surface-Q[6] General Characterization[3]

Notes: [1] TBD = to be determined prior to initial use and calibration.
[2] Release instrument (*i.e.*, FC 83-23)
[3] Characterization instrument
[4] FSS instrument, or will be used in support of FSS
[5] Instrument MDA depends on background
[6] This denotes instruments that may be used for internal surveys of equipment. Q denotes quantitative. NQ denotes non-quantitative results which cannot be used as the only FSS method.

The methods of interpreting stationary and scanning sensitivity[6] to beta radiation are in Appendix D. Those methods will be used to estimate the lower limit of detection (LLD or MDA) of each instrument before it is used to perform a final status survey (*a priori* LLD). LLD are estimated in Tables 4-1 and 4-2 for instruments for which representative background data are available.

[6] also called lower limit of detection (LLD), minimum detectable concentration (MDC), or minimum detectable areal density (MDAC)

4.3 BACKGROUND

Because the nuclides of interest are naturally occurring and are measurable in most materials, when β or γ is measured, then β or γ background values will be needed for the media to be surveyed, either equipment or building surfaces. As part of the characterization, background surveys were made for equipment and building surfaces. Background measurements were taken on site or in the immediate vicinity of the site in areas that were not affected by site operations. To determine brick backgrounds, bricks were removed from areas not affected by site operations, cut in half to expose fresh clay material, and surveyed. Table 4-3 shows the background levels which have been determined. Background values which must still be determined are denoted by TBD and values which are assumed to be 0 are also noted.

Table 4-3
Background Values of Some Materials of Construction

Material	Number of Samples	Average Background[a] (β/min/100 cm^2)	Standard Deviation (β/min/100 cm^2)
Asphalt	42	254	166
Brick	90	638	140
Concrete	70	180	79
Concrete Block	51	299	62
Concrete Block Bldg 101	21	560	62
Ceramic Tile	33	591	44
Counter Top - Bldg 250 Lab	10	403	77
Ceiling Tile	30	510	49
Roofing	N/A	TBD	N/A
Red Clay Tile	1	638	N/A
Tar/Roofing	N/A	Assumed 0	N/A
Vinyl Tile	N/A	Assumed 0	N/A
Wood	N/A	Assumed 0	N/A

[a] The data represent net β areal density, *i.e.*, open window minus closed window for an AB100 detector.

In the event additional background measurements of these materials are needed, they will be supplemented. Background radioactivity of these or other materials of construction is to be determined as needed.

In general, the sensitivity of a survey is greatest when the number of measurements is split between the background and the survey unit. In the event that the number of background data points acquired during characterization is insufficient, additional background data points will be taken as needed and as practicable.

4.4 SURVEY METHODOLOGY

The FSS will be designed to detect elevated areas and determine the distribution of radionuclides in accordance with the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM), NUREG –1727 Appendix E, and NUREG-1505 to the extent practical. First, areas will be classified according to potential for residual contamination or known contamination level. Next, these areas will be grouped into survey units as described in Section 4.4.2. Then, the required number of measurements and the survey locations will be determined. Finally, the surveys will be performed and the data will be analyzed.

4.4.1 Classification

The intent of the plan is to focus most of the survey effort on areas where the likelihood of exceeding the DCGL is greatest. Therefore, for the purposes of establishing the sample density and the sampling pattern, equipment and building surfaces will be classified according to the potential for residual contamination.

4.4.1.1 Equipment

Equipment is defined as material that is not a structural component of the building, is not permanently attached to, nor is an integral part of a building or structure. Examples of items that are not part of a building or structure include furniture, office machines, instruments, and appliances that are not built-into nor attached to the structure, stocks of chemicals, reagents, metals, and other supplies, a motor vehicle, and any other item that would not normally be conveyed with a building when it is sold. [7]

Equipment will first be classified as either Non-Impacted or Impacted. Non-Impacted equipment is equipment which has no reasonable potential for containing residual radioactive material and will not need any level of survey coverage. Equipment will be considered Impacted if it has been used or stored in areas where it may have been in contact with uncontained radioactive material.

Impacted equipment will be further subdivided into one of two categories for the purpose of demonstrating compliance with FC 83-23 as:

- **Category 1 Equipment**: Equipment which has become contaminated, or is highly suspect, requiring comprehensive or full survey.

- **Category 2 Equipment**: Equipment which is possibly contaminated but for which there is no direct evidence of contamination. At least a confirmatory/verification-type survey is required for unrestricted release of equipment in this category.

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[7] DG-4006, § B, p. 2.

Equipment to be released for use will be surveyed for unrestricted release. Otherwise, Category 1 equipment located in buildings slated for demolition, other equipment classified initially as Category 1, and equipment ultimately reclassified as Category 1 will either be decontaminated, if needed, and surveyed for unrestricted release as described in Section 2.2 or will be disposed of in accordance with Section 2.2.4, and will not be included in the FSS.

The equipment has been classified according to the potential for residual contamination as delineated in Appendix A. If the equipment is classified Impacted, the level of impact is noted.

Some high-value items such as inconel furnace tubes, filter presses, and perhaps steel beams or trusses may be reclaimed from a C-T process building. When so, each will be treated as removable equipment, subject to the *Guidelines* for equipment release.

Any equipment whose survey results exceed the maximum acceptable **average** areal density of regulated radioactive contamination specified in DP section 2, Table 2.1, shall be reclassified to Category 1.

4.4.1.2 Building Surfaces

Parts of a building or structure are subject to the decommissioning standards. Some examples are structural members, floors, walls, ceilings, doors, windows, sinks, hoods, lighting fixtures, built-in laboratory benches, built-in furniture,[8] ventilation duct external surfaces,[9] and installed apparatus.

Building surfaces will first be classified as either Non-Impacted or Impacted. Non-Impacted areas are areas that have had no reasonable potential for containing residual radioactive material and do not need any level of survey coverage. These areas have not been radiologically impacted from site operations. Impacted areas are areas which have potential for containing residual radioactive material.

Impacted areas will be further subdivided into one of three classifications:

- **Class 1 Areas**: These areas have, or had, a potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiological surveys). Areas containing contamination in excess of the DCGL prior to remediation shall be classified as Class 1 areas. Examples of Class 1 areas include: 1) site areas previously subjected to remedial actions, 2) locations where leaks or spills are known to have occurred, 3) waste storage sites, and 4) areas with contaminants in discrete solid pieces of material having high specific activity.

[8] DG-4006, §B, p. 2.
[9] NUREG-1727 Appendix E, § 10.3.

- **Class 2 Areas**: These areas have, or had, a potential for radioactive contamination or known contamination, but are not expected to exceed the DCGL. Examples of areas that might be classified as Class 2 for the final status survey include: 1) locations where radioactive materials were present in an unsealed form (*e.g.*, process facilities), 2) potentially contaminated transport routes, 3) upper walls and ceilings of buildings or rooms subjected to airborne radioactivity, 4) areas where low concentrations of radioactive materials were handled, and 5) areas on the perimeter of former contamination control areas.

- **Class 3 Areas**: Any Impacted areas that are not expected to contain any residual radioactivity, or are expected to contain levels of residual radioactivity at a very small fraction of the DCGL, based on site operating history and previous radiological surveys. Examples of areas that might be classified as Class 3 include buffer zones around Class 1 or Class 2 areas, and areas with very low potential for residual contamination but insufficient information to justify a Non-Impacted classification.

For the purposes of establishing the sample density and sampling pattern, the buildings have been classified according to the potential for residual contamination as delineated in Appendix A.

4.4.2 Survey Unit Definition

To facilitate survey design and assure that the number of survey data points for the media are relatively uniformly distributed within areas of similar contamination potential, and to focus most of the survey effort on areas where the likelihood of exceeding the DCGL is greatest, the media will be divided into survey units which have a common history or other similar characteristics, or are naturally distinguishable from other portions of the site.

4.4.2.1 Building Surfaces

A building surface survey unit will not include areas that have different classifications. The survey unit characteristics will be generally consistent with exposure pathway modeling that was used to convert dose into the DCGL. For indoor areas, where rooms are classified as Class 1 areas, each room may be designated as a survey unit. Indoor areas may also be subdivided into several survey units of different classification, such as separating floors and lower walls from upper walls and ceilings (and other upper horizontal surfaces) or subdividing a large warehouse based on floor area. This FSS applies to floors and walls. In the event the measurements on the walls exceed 0.5 of the $DCGL_W$, the ceiling shall be subject to survey at the appropriate classification.

Survey units will be limited in size based on classification, exposure pathway modeling assumptions, and site-specific conditions. The suggested maximum areas for survey units are as follows:

Table 4-4 Maximum Areas for Structure Survey Units	
Area Classification	Typical Maximum
Class 1	100 m^2 floor areas
Class 2	100 to 1000 m^2
Class 3	no limit

Installed apparatus or structural components of similar contamination potential in a survey unit may be surveyed as a survey unit of its own. The number of data points will be distributed approximately uniformly among the items in such a survey unit. In the event there are fewer items of equipment in a survey unit than the number of measurements recommended in a survey unit, measurement density does not have to exceed one per m^2, provided no measurement exceeds the $DCGL_W$. One or a few items of installed apparatus may be surveyed by at least one measurement at locations based on judgment and released without restriction provided every measurement is less than the $DCGL_W$.

4.4.3 Hypothesis Formulation

The decision that the DCGLs are met is based on a hypothesis test. Usually, the null hypothesis will be that the survey unit exceeds the release criterion.[10] This will require that significant evidence exists such that the residual radioactivity in the survey unit is less than the release criterion to reject the null hypothesis (and pass the survey unit). If the evidence is not significant at level α, the null hypothesis of a non-complying survey unit will be accepted and the survey unit will fail.

Alternatively, the tested hypothesis may be that measurements in a survey unit do not exceed background + $DCGL_W$, *i.e.*, scenario B,[11] and apply alternate, appropriate statistical test(s).

4.4.4 Selection of LBGR and Tolerable Decision Error

The number of measurements in a survey unit is a function of the LBGR, Types 1 and 2 decision error rates, and standard deviation of residual radioactivity. A reasoned balance between values of these parameters and number of measurements will be sought. A decision maker should balance costs of survey design, measurements, analyses, and reporting against consequence of decision error. When estimating consequence of decision error, radiological risk is assumed to be linearly proportional to radiological dose, and dose is linearly proportional to average radioactivity concentration or areal density.

[10] NUREG-1505, pp. 2-14 & 2-15.
[11] NUREG-1505, §2.5.

The following additional factors will be considered when deliberating selection of LBGR and Types 1 and 2 decision error rates other than default LBGR = 0.5 x $DCGL_W$ or $\alpha = \beta = 0.05$:

- conservatism in derivation of the DCGL,
- the residual radioactivity concentration or areal density range in the survey unit,
- cost-benefit of additional remediation, measurement, analysis, and or reporting versus concentration, dose, and risk reduction,
- physical distribution of residual source as it may affect remediation and logical survey unit boundary,
- appropriate LBGR and gray region, Δ, relative to background, variability in background, and multiple materials backgrounds in the survey unit;[12] increased α error may be tolerable when Δ/σ is small in order to avoid an unreasonably large number of measurements,
- difficult or adverse measurement conditions,
- interference in measurements, *e.g.* K^{40} interference in beta radiation measurement,
- whether measurement error can be reduced reasonably,
- and safety considerations.

The LBGR is the minimum concentration or areal density differentiable from the DCGL, *i.e.*, the minimum increment from the DCGL where one should begin to control false negative decision error. If a scenario A hypothesis is posed, the LBGR is bounded on the upside by the DCGL and on the downside by background.

Tolerable decision error rates are based on consideration of the consequence of making an incorrect decision about whether a survey unit complies with radiological criteria for release. The same value of both Type 1 (α) and Type 2 (β) errors will be selected,[13] although it is recognized that any value of β would be acceptable with respect to radiological safety.[14] The target for α and β will be 0.05. Selection of any greater value, not to exceed 0.15, will depend on consideration of these factors and documentation of the reasons in the survey report and requires NRC approval.

4.4.5 Determination of Number of Stationary Measurements

The number of measurements required for each survey unit will be determined by three considerations. The first is the sensitivity required by statistical analysis to support the hypothesis test, as discussed in Section 4.4.5.1 below. The second, applicable only to Category 1 equipment and Class 1 building surfaces, is the sampling density required for elevated area detection, discussed in Section 4.4.5.2 below. The number of stationary measurements for a given survey unit will be the larger of the two numbers determined in Sections 4.4.5.1 and 4.4.5.2, subject to selection of LBGR and tolerable decision error

[12] NUREG-1505, p. 3-17.
[13] MARSSIM, p. D-26,
[14] NUREG-1727 Appendix E, §7.2.

according to Section 4.4.4.

4.4.5.1 Number of Samples Required for Central Tendency Analysis

Two different approaches are used for central tendency analysis. The first applies to the situation in which the radionuclides of interest are present in the background. The second applies to the situation in which they are not present in the background (i.e., where the background concentration is zero or is assumed to be zero). These situations are discussed separately below.

4.4.5.1.1 Nuclides of Interest Present in Background

The number of data points required when the contaminant is present in the background will be determined in accordance with section 5.5.2.2 of MARSSIM:

- Estimate the relative shift, Δ/σ, where σ is the expected standard deviation of the survey unit measurements and Δ is the width of the "gray region." The gray region, as used in MARSSIM, can be considered as a region of central tendency nuclide concentration that corresponds to dose. Within the gray region, the probabilities of either a Type I decision error (deciding the unit meets DCGL when it does not) or a Type II decision error (deciding the unit exceeds DCGL when it does not) exceed desired limits. The upper bound of the gray region necessarily corresponds to the release criterion concentration. However, setting the boundary too low drives down the central tendency concentration below which one is highly confident in a decision that a unit exceeds DCGL. Setting the value too high drives up the number of measurements required to achieve the desired decision error probabilities. If decontamination is not performed, for purposes of estimating the number of data points, the value of σ will be estimated from characterization survey data or from background measurements, whichever is representative of the survey unit. If decontamination is performed, variability, σ, will be estimated from either 1) post-remediation survey, 2) characterization survey after deleting measurements exceeding the DCGL, or 3) background survey data. The determination of Δ/σ may be iterative.

- Determine P_r, the probability that a measurement performed at a random location in the survey unit will result in a larger value than a measurement performed at a random location in the reference area. P_r is determined from Table 5.1 of MARSSIM using the estimated relative shift, Δ/σ, determined in step 1.

- Determine the decision error percentiles, $Z_{1-\alpha}$ or $Z_{1-\beta}$, corresponding to the desired error probabilities α and β using Table 5.2 of MARSSIM. Tolerable decision error, α and β, will be decided in accordance with §4.4.4 herein.

- Calculate the total number of measurement points (survey unit plus reference area) for the WRS Test using Equation 4-1 (Equation 5-1 from MARSSIM) with input parameter values determined in steps 1 through 3.

$$N = \frac{\left(Z_{1-\alpha} + Z_{1-\beta}\right)^2}{3\left(P_r - 0.5\right)^2} \qquad \text{[Equation 4-1]}$$

Where $Z_{1-\alpha}$ and $Z_{1-\beta}$ are the percentiles represented by the selected decision error levels; P_r is the probability that a measurement performed at a random location in the survey unit will result in a larger value than a measurement performed at a random location in the reference area; and N is the number of data points to be obtained from each reference area + survey unit combination.

Equation 4-1 assumes equal numbers of background and survey measurements. As noted in the discussion on background in section 4.3, the number of background measurements may be constrained. There will be some advantage in sensitivity achievable by collecting a larger number of measurements from the survey unit, although there will be diminishing returns as the number of measurements from the survey unit increases. To calculate the number of samples in this case, Equation 4-2 (based on NUREG-1505, Equation 9-6) is first used:

$$N = \frac{\left(Z_{1-\alpha} + Z_{1-\beta}\right)^2}{12c(1-c)\left(P_r - 0.5\right)^2} \qquad \text{[Equation 4-2]}$$

where $Z_{1-\alpha}$ and $Z_{1-\beta}$ are the percentiles represented by the selected decision error levels; P_r is the probability that a measurement performed at a random location in the survey unit will result in a larger value than a measurement performed at a random location in the reference unit by less than the $DCGL_W$; c is the fraction of samples from the survey unit; and N is the number of data points to be obtained from each reference area + survey unit combination.

In any survey there will be some missing or unusable data. The rate of missing or unusable measurements, R, expected to occur in survey units or reference areas will be accounted for during survey planning. To assure sufficient data points to attain the desired power level with the statistical tests and allow for possible lost or unusable data, the number of data points will be increased by 20% (R=0.2), and rounded up, over the values calculated in the final step. In the event it is not practical to collect 1.2·N measurements, as few as N measurements will be acceptable without verification by a retrospective power curve.

The required number of measurements determined in the first iteration may exceed reasonable bounds. The process can be repeated using more suitable values of Δ/σ, α, and β as appropriate.

4.4.5.1.2 Nuclides of Interest Not Present in Background

The general approach to determining the number of required data points when the contaminant is not present in the background parallels, to some extent, the approach used for the situation when the contaminant is present in the background, described above. However, because background concentrations need not be considered, the formulation for the number of measurements differs somewhat.

- First, the relative shift Δ/σ is determined as described above.
- Next, Sign p is determined. Sign p is the probability that a random measurement from the survey unit will be less than shift, Δ. Given the relative shift, Sign p is determined using the previously determined value of Δ/σ and Table 5.4 of MARSSIM.
- Determine the decision error percentiles corresponding to the desired error probabilities α and β using Table 5.2 of MARSSIM. On the first iteration, a value of 0.05 will be used for α and β. Larger values may be used on subsequent iterations.
- Finally, the number of data points is calculated using Equation 4-3 (equation 5-2 from MARSSIM)

$$N = \frac{\left(Z_{1-\alpha} + Z_{1-\beta}\right)^2}{4\left(Sign\ p - 0.5\right)^2} \qquad \text{[Equation 4-3]}$$

The number of anticipated data points will be increased by 20% to assure sufficient power of the tests and to allow for possible data losses. In the event it is impractical to collect 1.2·N measurements, as few as N measurements will be acceptable without confirmation by a retrospective power curve. Because the presence of nuclides of interest in background are not a consideration for this case, detection difficulties should not arise. However, if they do, the required number of measurements determined in the first iteration can be adjusted by repeating the process using more suitable values of Δ/σ, α, and β as appropriate.

4.4.5.2 Determining Data Points for Small Areas of Elevated Activity

The measurements described in section 4.4.5.1 are designed to test whether the DCGL are met based on measures of central tendency. Scanning measurements are the primary method for detecting small areas of elevated contamination. Scanning will be performed for survey units containing building surface classes 1 and 2. For survey units containing building surface class 2, the scanning is confirmatory only. For survey units containing building surface class 1, individual measurements that may exceed DCGL are anticipated. The interest in these units is to assure that the concentrations and areal extent of elevated contamination are sufficiently constrained to assure that unrestricted release dose criteria are met. To the extent that scanning measurements are not sufficiently sensitive to assure this, they are supplemented by stationary measurements.

The minimum detectable areal density by scanning (scan MDC) needs to be less than the maximum tolerable areal radioactivity density for any given area of contamination smaller than a grid cell for systematic measurement, *i.e.*, the $DCGL_{EMC}$. Thus the required scan MDC must be less than or equal to the $DCGL_w$ times the area factor. Suppose an area of elevated contamination equals a systematic grid cell area. In that instance, the required scan MDC must be less than or equal to $DCGL_w$ times the area factor corresponding to grid cell area. When the scan MDC is estimated, the area factor corresponding to this estimated, or *actual* scan MDC is

$$\text{area factor corresponding to actual scan MDC} = \frac{\text{actual scan MDC}}{DCGL_W}$$

The contaminated area corresponding to the area factor that corresponds to the *actual* scan MDC is the largest area of elevated radioactivity potentially causing 25 mrem/yr that is detectable by that scan instrument MDC.
The corresponding number of areas of elevated radioactivity in a survey unit, n_{EA}, is derived by the equation:

$$n_{EA} = \quad \text{m} \quad \frac{\text{survey unit area}}{\text{area corresponding to actual scan MDC}}$$

If n_{EA} is less than $n_{wilcoxon}$,[15] calculate systematic grid spacing with one of the equations:

For a triangular grid, $L = \sqrt{\frac{A}{0.866 \cdot n}}$, or. Equation 4-4

For a square grid, $L = \sqrt{\frac{A}{n}}$. Equation 4-5

If n_{EA} is greater than $n_{wilcoxon}$, calculate systematic grid spacing with one of the equations:

For a triangular grid, $L = \sqrt{\frac{A}{0.866 \cdot n_{EA}}}$, or. Equation 4-6

For a square grid, $L = \sqrt{\frac{A}{n_{EA}}}$. Equation 4-7

[15] $N_{wilcoxon}$ = number of measurements needed to provide desired confidence in a Wilcoxon Rank Sum test

4.4.6 Survey Locations

The number of data points required and the area of the survey unit will determine the location of the data points.

4.4.6.1 Building Surfaces Survey Locations

A scale drawing of the survey unit will be prepared, along with the overlying planar reference coordinate system or grid system. Any location within the survey area is thus identifiable by a unique set of coordinates. The maximum length, X, and width, Y, dimensions of the survey unit are then determined.

All structure surfaces for a specific survey unit will be included on a single reference grid system for purposes of identifying survey locations. Measurements and samples in Class 3 survey units and reference areas will be taken at random locations. These locations will be determined by generating sets of random numbers (2 values, representing the X axis and Y axis distances). Each set of random numbers will be used to provide coordinates, relative to the origin of the survey unit reference grid pattern. Coordinates identified in this manner, which do not fall within the survey unit area or which cannot be surveyed, due to site conditions, will be replaced with other survey points determined in the same manner.

Class 2 areas will be surveyed on a random-start systematic pattern. The number of survey locations calculated to satisfy statistical tests, will be used to determine the spacing, L, of a systematic pattern by the following equations, Equation 4-4 for a triangular grid or Equation 4-5 for a square grid. In the equations, L is the spacing, A is the area of the survey unit, and n is the number of calculated survey locations.

$$L = \sqrt{\frac{A}{0.866n}} \qquad \text{Equation 4-4}$$

$$L = \sqrt{\frac{A}{n}} \qquad \text{Equation 4-5}$$

The choice of grid pattern will be made by considering elevated area contamination potential and general shape of the survey unit. The grid which is most practical to survey and evaluate will be chosen.

After L is determined, a random coordinate location will be identified, as described previously, for a survey pattern starting location. Beginning at the random starting coordinate, a row of points will be identified, parallel to the X axis, at intervals of L. For a triangular grid, a second row of points will then be developed, parallel to the first row, at a distance of 0.866•L from the first row. Survey points along that second row will be midway (on the X-axis) between the points on the first row. This process will be repeated to identify a pattern of survey locations throughout the survey unit. If identified points fall outside the survey unit or at locations which cannot be surveyed, additional points will be determined using the random process described above, until the desired total number of points is identified.

For Class 1 areas, a systematic pattern will be installed on the survey unit. The starting point for this pattern will be selected at random, as described above for Class 2 areas. The same process as described above for Class 2 areas applies to Class 1, only the estimated number of samples may be different.

In addition to the survey locations identified for statistical evaluations and elevated measurement comparisons, it is likely that data will also be obtained from judgment locations, selected due to unusual appearance, location relative to contamination areas, high potential for residual activity, general supplemental information, *etc.* These data points selected based on professional judgment will not be included with the data points from the random-start triangular grid for statistical evaluations; instead they will be compared to the investigation levels described in section 4.4.7.2. Measurement locations selected based on professional judgment violate the assumption of unbiased measurements used to develop the statistical tests.

In the event a grid node for systematic survey were to occur where the intended surface is not accessible, that measurement may be relocated either by 1) random selection, 2) offset by <0.2 of the systematic grid spacing, or 3) onto an accessible, intervening surface of similar contamination potential.

4.4.7 Surveys

4.4.7.1 Equipment Surveys[16]

Survey Specifications: Removable equipment will be surveyed in accordance with the *Guidelines*[17] as interpreted herein. Equipment surface release limits for regulated radioactive material above background is specified in Section 2.2.2.

Removable equipment to be released for use will be surveyed for unrestricted release. Any equipment that must be decontaminated in order to be released will be classified as Category 1. Otherwise, Category 1 equipment located in buildings slated for demolition, other equipment classified initially as Category 1, and equipment ultimately reclassified as Category 1 will either be surveyed individually to decide eligibility for release, will be decontaminated, if needed, and surveyed for unrestricted release, or else will be disposed

[16] Ref. §4.4.1.1 definition. See §4.0, ¶2 also.
[17] NRC. "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material." Table 1. April 1993.

in accordance with Section 2.2.4.

Equipment survey specifications will identify the measurements to be made and will be subject to approval by Mallinckrodt's RSO.

An equipment survey specification will be influenced by several factors:

- operational history;
- radioactive contamination potential, *i.e.*, Category (ref. §4.4.1.1)
- whether access to each potentially contaminated part of the equipment is reasonably achievable;
- whether the equipment can reasonably be dismantled;
- whether radioactive solids, *e.g.*, sediment, may be accumulated in the equipment; and
- whether the effect of a coating such as paint or grease over radioactive contamination that attenuates radiation emitted and prevents detection of underlying radioactive contamination.

Each item of impacted removable equipment will be assigned a number and category (ref. Appendix A). A sketch or drawing of the equipment will be made. An appropriate combination of direct, scan measurements and/or wipes will be specified taking into account the considerations and factors specified above. Although guidance will be provided, the exact number and location of survey points may be left to the judgment of the surveyor on-site. The surveyor will annotate the sketch or drawing to show where scans and measurements were made or samples were taken. These labels will also be put on data sheets to identify where measurements were made and or samples were taken.

If the equipment to be surveyed is comprised of numerous pieces, then at least one measurement per piece of equipment should be obtained. Judgment with respect to factors mentioned will help decide where and how much of impacted equipment to survey.

Guidance for access to the inside of equipment: The radioactivity on the interior surfaces of pipes, drain lines, air ducts, or other places difficult to access shall be evaluated by making measurements at traps and other appropriate access points, provided that contamination at these locations is likely to be representative of contamination on the interior of the pipes, drain lines, or ductwork. Smear measurements shall be taken along with the direct measurements of access points.

Inaccessible Surface: If the surface of removable equipment that is inaccessible to direct measurement or sampling is suspected to be or is likely to be contaminated, then it shall be presumed to be contaminated in excess of the *Guideline* limits.

Disposition: If during the course of a survey for equipment, if survey results, direct, scan or removable, are greater than the maximum acceptable average or removable criterion, then additional action such as opening system components is required in order to survey to ensure that no excessive, hidden contamination exists. If such a survey is not feasible, then the equipment is to be disposed.

If contamination greater than the maximum acceptable areal density in Table 2.1 is confirmed, the equipment will be disposed of in accordance with section 3.6 or will be decontaminated and categorized as Category 1 for survey again.

Affected removable equipment known to be contaminated or no longer useful may not go through a complete survey; if not, it will be disposed by an NRC-approved transfer to a disposal facility.

4.4.7.2 Building Surface Surveys

This class includes surfaces of buildings, structures, and installed apparatus. After the number of required samples has been established and the location of the samples is determined, the survey strategy will be developed using the following guidelines. The survey methods focus on beta radiation measurements, however alpha radiation measurements may be used in lieu of beta measurements if conditions warrant.

4.4.7.2.1 Class 1 Areas

Surface scans will be performed over 100% of structure surfaces for alpha or beta radiation which might be emitted from the potential radionuclide contaminants. Locations of direct radiation, above the $DCGL_{EMC}$, will be identified and evaluated. Results of initial and follow-up direct measurements and sampling at these locations will be recorded and documented in the Final Status Survey Report. Measurements of total contamination will be performed at locations identified by scans and at previously determined locations (Section 4.4.5.2).

4.4.7.2.2 Class 2 Areas

Surface scans will be performed for β radiation over at least 10% of building surfaces in a Class 2 survey unit. Scanning will be done on reasonably accessible surfaces. In the event a physical constraint were identified that would make it unreasonable to scan as much as 10% of a survey unit, the survey report would describe the area and condition that justifies lower scan coverage.

4.4.7.2.3 Class 3 Areas

Scans of Class 3 area surfaces will be performed for beta radiations at designated locations. These locations will be determined through the use of historical knowledge and contractor experience. Direct (stationary) measurements of total beta contamination will be performed at locations identified by the scans and at the randomly selected locations, chosen in accordance with Section 4.4.5.2.

4.4.8 Data Analysis

The evaluation of final status survey results is performed in four stages. The first stage will consist of a preliminary review of the data. The second stage of evaluation will consist of an evaluation of elevated measurements against investigation levels. The third stage of data evaluation will consist of statistical analysis to determine if DCGL have been satisfied. The last stage of data evaluation will consist of concluding whether the results of the survey meet the design objectives. Based on results from the first three stages, resurvey, reclassification, remedial action, or some combination of these measures may be required. The survey will not be complete until the conclusion that survey objectives have been met can be supported. Each of these stages is discussed in greater detail in the following subsections.

Final status surveys will be managed in the following way. In summary:

- Each final status survey design will be documented and approved for execution.
- Final status survey measurements will be recorded.
- Final status survey data of record will be examined for legibility, completeness, conformance with the survey design specifications, and apparent errors.
- Original survey design specifications and survey records will be retained.
- If the data are acceptable, tests of compliance with DP §2.2 release criteria will be performed. These would test each measurement for compliance with the *elevated measurements criterion* and would test the systematic measurements in the survey unit for compliance with collective data criteria.
- A final status survey report of the data, screening tests, and compliance tests is examined.
    - If either the final status survey records, checks, or a test is not accepted, contingent actions described in DP §4.4.9, "Contingencies," will be considered.
    - Else, if the data, screening tests, and compliance tests for that survey unit are accepted, a final status survey report for that survey unit will be written.

4.4.8.1 Preliminary Data Review

The first stage of data evaluation will consist of a preliminary review of the data to check quality and reasonableness, including:

- legibility of recorded data,
- assessment of completeness of the data,
- verification of instrument selection and calibration,
- verification of survey technician training qualifications,
- an initial judgment about the overall quality of the data. This would aim to identify gross errors in data recording,

- conversion of raw data to standard units where appropriate. For building surfaces, the units are (dpm/100 cm^2),
- whether the number of points taken are in accordance with the specific survey plan requirements,
- whether the locations of systematic grid points correspond to what was prescribed in the survey design,
- whether systematic survey locations were inaccessible and locations substituted by the field team were adequate for use as systematic grid points in the population statistical analysis,
- whether scan data were adequately processed, (Scan data sheets will be reviewed to determine whether any high reading requiring investigation has been adequately verified with a direct confirmatory measurement), and
- whether bias data points were taken in accordance with FSS design instructions and their location accurately documented on drawings and in the database.

If, during this initial review, any measurement is considered inadequate, additional survey information may be collected. Once this review is complete and the analyst is satisfied, the survey records will be "locked," completing preliminary data review and acceptance of the survey. Permanent archive files of all survey data taken on connection with the C-T project will be maintained to provide for their security, organization, and availability to authorized reviewers, including the NRC.

4.4.8.2 Evaluation of Measurements Individually

In the second stage of evaluation, scan results and individual measurements from installed equipment and building surfaces will be compared to an appropriate investigation level for evidence of a small area of elevated radioactivity. An investigation level is a radioactivity concentration, areal density, or index that is used to indicate when additional investigation may be necessary. An investigation level depends on survey unit classification. A scan result which exceeds the corresponding investigation threshold listed in Table 4.5 shall be confirmed by direct measurement. Scan measurement results will remain as paper records. The direct measurement data only will be recorded for further analysis and classification.

The data set for the survey unit will be processed within a database using screening software developed and verified for the project. The screening software will perform the following comparison tests:

Data Screening Tests
Min/Max screen
Background screen
$DCGL_w$ screen
EMC limit screen

A brief description of each test applied follows.[18]

- Min/Max Screening. Recorded data points in the survey unit will first be processed to derive the difference between the largest survey value and the smallest applicable background value. If that difference is less than the $DCGL_W$, a class 1 or class 2 survey unit will be rated clean and no further computation will be needed. A class 3 survey unit will be passed on to the background screen. **If a class Screening 2 or 3 survey unit fails** this test, it will be evaluated for additional analysis, remediation, or other appropriate action. A class 1 unit that fails this test will be passed on to perform the remaining tests.

- Background Screening. All class 3 survey units will be processed through background screening. Each data point that fails the test will be flagged as an exception. In a class 3 survey unit, residual, regulated radioactivity is not expected. Therefore, the investigation level is set to flag any measurement that is just above the range expected for background or just above the detection limit for the measurement method, whichever is greater (ref. Table 4.5).

- Elevated Measurement Comparison (EMC) Limit Screening. In a class 1 survey unit, measurements above the $DCGL_W$ are expected, so a special derived concentration guideline for elevated measurements, $DCGL_{EMC}$, will be calculated and supplied in the final status survey design. The derived concentration guideline level for the elevated measurement comparison (EMC) is determined as described in Appendix C. The area on which $DCGL_{EMC}$ is derived *a priori* will be the systematic grid cell area derived in section 4.4.5.

  The background reference level for each data point will be the average of background measurements associated with the same reference material Matrix Code. Any value exceeding the EMC criterion in table 4.5 will be flagged for investigation.

  In the event that an area of elevated radioactivity is identified, an additional test is performed to ascertain whether the overall radioactivity concentration in the survey unit is greater than the release limit.[19] Following determination of the size of the elevated area and radioactivity concentration therein, a sum-of-fractions rule[20] should be used to ascertain whether the radioactivity concentration over all the survey unit is less than the $DCGL_W$. To pass the test, the combined contribution from the elevated area and the remainder of the survey unit must conservatively be less than 0.95 instead of unity (1).

- $DCGL_W$ Limit Screening. The net radioactivity of each survey point within the survey unit will be compared with the $DCGL_W$. The reference level for each data point will be the average of background measurements associated with the same matrix. Each value exceeding $DCGL_W$ will be flagged as an exception, point by point. In addition, the mean value for the reference area and the survey unit will

[18] MARSSIM, Table 8.2.
[19] MARSSIM, §8.5.1.
[20] MARSSIM equation 8.2.

be compared for class 1 and class 2 survey units. If the average radioactivity observed in a survey unit exceeds background average, such that the difference between the two is greater than the $DCGL_W$, the entire survey unit will be flagged for additional analysis, remediation, or other appropriate action as to its reclassification.

If a test is failed, an analyst will decide about reclassification, remediation or release of the survey unit. Exceptions will be handled on a case-by-case basis. For example, the analyst may opt to reapply the tests after accounting for an unusually high level of K-40, if it is present. The resolution of each flagged datum will be documented to provide a clear understanding of how the survey unit was ultimately released in the final status survey report.

4.4.8.3 Investigation Levels

Individual measurements from equipment and building surfaces will be compared to investigation levels for evidence of small areas of elevated activity. Scan results for those units subject to scanning will also be compared to investigation levels. The levels established for investigation will depend upon the survey unit classification. Lower investigation thresholds will be set for those units having lower potential for elevated areas. Depending on the outcome of the elevated measurement test and other tests, resurvey, reclassification, partial or complete remediation, or some combination of these measures may be required. (If only partial remediation is required, resurvey of some portion of the unit after supplementary remediation will also be required. To the extent practical and appropriate, original survey data from portions of the unit outside the supplementary remediation area will be used in conjunction with new survey data from the supplementary remediation area in new tests to determine whether the unit meets release criteria.)[21,22].The results of all investigations will be documented in the final status survey report. Investigation levels are described in greater detail below.

A building class 3 area is expected by history or previous radiation survey to have no residual radioactivity or no more than a small fraction of the $DCGL_W$. Therefore, investigation levels are set to flag measurements that are just above the range expected for background levels or just above detection limits for the measurement method, whichever is greater.

In a building surface class 2 survey unit, measurements of net levels above the $DCGL_W$ are not expected. Therefore, investigation levels are set to flag measurements exceeding the $DCGL_W$ on a net basis. If the scanning MDA exceeds the DCGL on a net basis, any scanning result exceeding the MDA will also be investigated.

In a building surface class 1 survey unit, measurements above the $DCGL_W$ are not unexpected, so a special derived concentration guideline for elevated measurements, $DCGL_{EMC}$, is used as the basis for investigation levels. The derived concentration

21 NUREG-1727, Appendix E
22 MARSSIM, §5.5.2.6.

guideline level for the EMC (elevated measurement comparison) is determined as described in Appendix C.

The area on which $DCGL_{EMC}$ is derived *a priori* will be the systematic grid cell area derived in section 4.4.5.

Table 4-5 summarizes the investigation levels.

Table 4-5
Summary of Investigation Levels[a]

Survey Unit Classification	Flag direct measurement or sample result when gross β is greater than the larger of:	Flag scanning measurement result when gross β is greater than the larger of:
Building Surface Class 1	$x_{ref} + 2 \cdot s_{ref} + DCGL_{EMC}$	$x_{ref} + 2 \cdot s_{ref} + DCGL_{EMC}$
Building Surface Class 2	$x_{ref} + 2 \cdot s_{ref} + DCGL_W$	MDA or $x_{ref} + 2 \cdot s_{ref} + DCGL_W$
Building Surface Class 3	MDA or $x_{ref} + 2 \cdot s_{ref}$	MDA or $x_{ref} + 2 \cdot s_{ref}$

[a] Where: x_{ref} and s_{ref} are the mean and standard deviation of the reference, or background, measurements, or indices calculated from measurements.

4.4.8.4 Conduct Statistical Analysis

Unless they passed the *Min-Max test*, stationary measurements at systematic grid locations will be examined statistically to determine whether release criteria have been satisfied. This stage will also include reassessment of the power of the hypothesis test, based on survey data. If the survey unit does not pass, consideration according to §4.4.9 of values of Δ/σ, α, and β more suitable than those used in the survey design may be appropriate.

Statistical analysis is required only on class 1 or 2 survey units where the mean survey value is above the mean reference value by an amount less than $DCGL_W$. The analyst will select the test that is most appropriate to the survey unit and the statistical test program will perform one of the following tests:

Statistical Tests
Sign Test[23]
Wilcoxon Rank Sum Test (WRS)[24]
Kruskall-Wallis Test
Sign Test for Paired Data[25]

Different forms of statistical analysis are required depending on whether the nuclides of interest are present in the background. These two forms are discussed separately below. The test result will be examined to decide whether it passes or fails the tested hypothesis. If criteria are not met, resurvey or remedial action may be required.

4.4.8.4.1 Nuclides Present in Background

For situations in which the nuclides of interest are present in the background, the following logic will be used to judge compliance with the DCGL:

Table 4-6
Summary of Statistical Tests
(when radionuclide(s) is (are) in background)

Survey Result	Conclusion
All survey unit measurements are less than $DCGL_W$ on a net basis	Survey unit meets release criteria
Difference between any survey unit measurement and any reference area measurement greater than $DCGL_W$ (not to be used for survey units with less than 5 measurements)	Conduct WRS test or Sign Test for Paired Data and elevated measurement comparison

If statistical analysis is necessary for a survey unit, either the two-sample Wilcoxon Rank Sum (WRS) test (also called the Mann-Whitney test) or the Sign Test for Paired Data will be conducted.

The WRS test assumes that the reference area and survey unit data distributions are similar except for a possible shift in the medians, and is applied as follows:[26]

[23] Also called the "One-sample Sign Test", the *Sign Test* evaluates whether the median of the data is above or below the $DCGL_W$. It should be used when there is no background present or if it is so low as to be insignificant.

[24] Also called the "Mann-Whitney Test", a *Wilcoxon Rank Sum Test* should be used when there is background radiation present and the background characteristics and radioactivity distribution are similar for the materials present in the survey unit.

[25] A *Sign Test for Paired Data* may be applied where there are multiple background materials within the same survey unit. This test may be the most commonly used test in the CT Project because of the wide variety of matrix materials present. The derivation of the σ term for multiple background materials will be based on the guidance of NUREG-1505.

[26] MARSSIM, §8.4.2.

Obtain the adjusted reference area measurements, Z_i, by adding the $DCGL_W$ to each reference area measurement, X_i as shown in equation 4-7.

$$Z_i = X_i + DCGL_W \qquad \text{[Equation 4-7]}$$

The m adjusted reference sample measurements, Z_i, from the reference area and the n sample measurements, Y_i, from the survey unit are pooled and ranked in order of increasing size from 1 to N, where $N = m+n$.

If several measurements are tied (have the same value), they are all assigned the average rank of that group of tied measurements.

Use of "less than" values in data reporting will be minimized to the extent practical. If more than 40 percent of the data from either the reference area or survey unit are "less than," the WRS test will not be used. If there are t "less than" values, they are all given the average of the ranks from 1 to t. Therefore, they are all assigned the rank $t(t+1)/(2t) = (t+1)/2$, which is the average of the first t integers.[27]

Sum the ranks of the adjusted measurements from the reference area, W_r.

Compare W_r with the critical value given in Table I.4 of the MARSSIM manual for the appropriate values of n, m, and α. If W_r is greater than the tabulated value, reject the hypothesis that the survey unit exceeds the release criterion.

4.4.8.4.2 Contaminant Not Present in Background

For situations in which the contaminant of interest is not present in the background, the following logic will be used to judge central tendency compliance with the release criteria:

Table 4-7
Summary of Statistical Tests
(when radionuclide(s) is (are) not in background)

Survey Result	Conclusion
All measurements less than $DCGL_W$	Survey unit meets release criterion
Any measurement greater than $DCGL_W$ and average is less than $DCGL_W$ (not to be used for survey units with less than 5 measurements)	Conduct Sign test and elevated measurement comparison

If statistical analysis is required for a survey unit, the one-sample Sign test will be used.

27 NUREG-1505, p. 2-19.

The Sign test evaluates whether the median of the data is above or below the DCGL, and is applied as follows:[28]

- List the survey unit measurements, X_i , i = 1, 2, 3..., N.
- Subtract each measurement, X_i, from the DCGLw to obtain the differences, D_i using equation 4-8. where i = 1, 2, 3..., N.
- $D_i = DCGL_W - X_i$ [Equation 4-8]
- If any difference is exactly zero, discard it from the analysis, and reduce the sample size, N, by the number of such zero measurements.
- Count the number of positive differences. The result is the test statistic S+. A positive difference corresponds to a measurement below the DCGLw and contributes evidence that the survey unit meets the release criterion.

Large values of S+ indicate that the null hypothesis (that the survey unit exceeds the release criterion) is false. The value of S+ is compared to the critical values in Table I.3 of the MARSSIM manual. If S+ is greater than the critical value, k, in that table, the null hypothesis is rejected. Otherwise, the null hypothesis is accepted.

4.4.8.5 Draw Conclusions and Document Survey

The last stage of data evaluation will examine whether final survey results met survey design objectives. An affirmative conclusion indicates the survey is complete. Otherwise, reclassification, resurvey, remedial action, or some other contingent action in section 4.4.9 would be appropriate.

A final status survey will provide a record of the radiological status of the survey unit, relative to the DCGL. To the extent practicable, this report will be a stand-alone document with minimum information incorporated by reference.

Each final radiation status survey report shall demonstrate compliance with the radiological criteria for release for unrestricted use. For each survey unit, a report shall include:

- description and a map or sketch outlining the location and boundaries of the survey unit;
- survey unit classification;
- if used, a description of background reference area(s) and or medium(media);

[28] MARSSIM, §8.3.2

- name and model of detector(s) and instrument(s) used to perform final status radiation measurements of record, and detection sensitivity for each survey unit and material surveyed therein.
- if the survey unit is Class 1, a summary of elevated measurements comparisons with $DCGL_{EMC}$;
- Types 1 (α) and 2 (β) decision error values and justification for any value >0.05;
- summary of statistical test(s) used to evaluate survey data; and
- justification of any statistical test method not included in the MARSSIM, in NUREG-1505, or in NUREG-1727 Appendix E;
- description and or explanation of deviation from FSS DQO and/or decommissioning plan;
- discussion of methods used to address survey units that failed and how they were addressed and;
- QA/QC compliance and supporting data.

The report will be reviewed and approved prior to release, publication, or external distribution.

4.4.9 Contingencies

In the event final survey unit measurement(s) appear not to satisfy a release criterion, some alternative actions may be taken to assess whether it does and or to enable the survey unit to pass criteria. Some acceptable alternatives to remediating the entire survey unit and performing another final status survey follow. One or more may enable demonstration of compliance:

- Objectives would be reviewed with respect to how to assess whether a survey unit meets survey criteria.[29,30]
- Ordinarily, survey data would first be reviewed to confirm its acceptability. One may also decide whether additional data are needed to determine whether the survey unit complies with release criteria.[31]
- Reassess the reference area or material to be compared with the survey unit

If DQO are inappropriate or if a survey unit is misclassified, Mallinckrodt may:

- Review the DQO. If warranted, adjust values of parameters such as Type 1 and Type 2 error criteria or the lower bound of the gray region (LBGR).
- Reclassify part of a survey unit that contains elevated measurements. Remediate if necessary. Measure at the density appropriate to the new classification. If the reclassified part were Class 1, the measurement density appropriate for Class 1, and the number of measurements in it were fewer than would be estimated for an

29 MARSSIM, p. 8-25.
30 NUREG-1505, p. 3-1.
31 MARSSIM, p. 8-24.

entire Class 1 survey unit, compliance would be accepted if every measurement[32] in the reclassified part were less than the $DCGL_W$.

- o In the event a Class 1 survey unit area is less than 15 m^2 and the number of measurements are specified and tested statistically for compliance with $DCGL_W$, the area factor shall not exceed that specified in Appendix C for the elevated measurement test. Alternatively, in the event a Class 1 survey unit area is less than about 15 m^2, the number measurements estimated to satisfy a WRS, Quantile, or Sign test might be unreasonably large in that survey unit. When both conditions exist, measurement density will be at least one measurement per square meter at locations based on judgment. In that circumstance, the criterion for release shall be that every measurement in the survey unit does not exceed the $DCGL_W$.[33]

- o In the event a Class 2 survey unit area is less than 100 m^2, the number measurements estimated to satisfy a WRS, Quantile, or Sign test might be unreasonably large in that survey unit. When both conditions exist, measurement density will be at least one measurement per 10 m^2 at locations based on judgment. The criterion for release in that circumstance, shall be that every measurement in the survey unit does not exceed the $DCGL_W$.[34]

- If the scanning method was not sensitive enough in a Class 2 unit, a portion containing measurements greater than the $DCGL_W$ may be reclassified as Class 1, measured at the measurement density for a Class 1 area, with the rest of the survey unit remaining Class 2.[36]

If an elevated measurements test is failed, Mallinckrodt may:

- If a survey unit passes statistical test(s) but radioactivity concentration in a local area exceeds the $DCGL_{EMC}$, *i.e.*, the product of $DCGL_W$ x area factor, for its actual size, remediate the local area. If a post-remediation survey of the local area demonstrates residual radioactivity is below the $DCGL_W$, compliance is acknowledged. Else, statistical tests for the survey unit are performed again.[35]
- If a survey unit passes. Compute the radiological dose associated with each measurement as if it represented the entire survey unit and calculate the arithmetic mean dose represented by all the measurements in the area of elevated radioactivity. If the mean dose does not exceed the product, area factor x

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[32] interpret to be measurement net of background
[33] MARSSIM, p. 4-15.
[34] MARSSIM, p. 4-15.
[35] MARSSIM, p. 8-24 & 25.

radiological dose criterion, *i.e.*, AF x $DCGL_W$, compliance would be demonstrated for the elevated measurements criterion for that local area.

If a non-parametric statistical test is failed, Mallinckrodt may:

- Construct a retrospective power curve[36] of the measurements. Evaluate whether the survey unit would have passed the release criterion using the WRS or Sign test. If not, it would be acceptable to make more measurements at random locations in the survey unit and perform statistical test(s) on the expanded data set.[37]
- Make more appropriate measurements to improve determination of background.
- Reverse the tested hypothesis and apply an alternate, appropriate statistical test, *e.g.*, from scenario A to scenario B.[38] Specific DQO would be developed for this approach and be submitted to the NRC for approval, or would be addressed in the FSS report for survey units that fail.
- In lieu of statistical testing, compute the radiological dose associated with the mean of measurements in the survey unit. Alternatively compute the radiological dose attributable to each measurement as if it represented the entire survey unit and calculate the arithmetic mean dose represented by all the measurements in the survey unit. If the mean dose does not exceed the radiological dose criterion, compliance would be demonstrated for the survey unit.

4.5 FINAL STATUS SURVEY QA/QC

Section 5.0 describes the Quality Assurance Program (QAP) which will ensure that the results of the Final Status Survey are accurate and the uncertainties have been considered.

4.6 FSS DESIGN CONSIDERATIONS

This document provides the guidance needed to design and create FSS designs in accordance with the Mallinckrodt CT Phase 1 Decommissioning Plan. All FSS plans used in the CT project should conform to these guidelines.

The CT Phase 1 Decommissioning Plan distinguishes between buildings and equipment.

4.6.1 Buildings

Buildings, structures, building components, and apparatus installed before CT operation ceased will be subject to the building surface release criteria [ref. DP section 2.2.3] in the survey unit in or on the building in or on which it is delineated.

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36 MARSSIM, Appx I.
37 MARSSIM, p. I-25 & I-27.
38 NUREG-1505, §2.5.

Some examples of parts of a *building* or structure, including installed apparatus that are subject to buildings release criteria are structural members, floors, walls, ceilings, doors, windows, sinks, hoods, lighting fixtures, built-in laboratory benches, built-in furniture, ventilation ducts, and process vessels.[39]

Buildings or structures and equipment or apparatus installed after C-T operation ceased are presumed clean and are not required to have a final status survey.

Otherwise, buildings slated for demolition and contaminated apparatus therein[40] will be disposed of in accordance with DP section 2.2.4, and are not subject to final status survey.[41]

4.6.2 Equipment

Equipment is defined as material that is not a structural component of the building, is not permanently attached to, nor is an integral part of a building or structure. Examples of items that are not part of a building or structure include furniture, office machines, instruments, and appliances that are not built-into nor attached to the structure, stocks of chemicals, reagents, metals, and other supplies, a motor vehicle, and any other item that would not normally be conveyed with a building when it is sold.[42, 43] Removable equipment which will be released for unrestricted use will be surveyed in accordance with the *Guidelines*[44] [ref. DP section 2.2.2].

Mallinckrodt staff relied on historical knowledge of C-T processing and material handling to identify process, support, and proximate facilities to be characterized. 116 areas were examined during an extensive radioactivity characterization.[45] This extensive characterization, along with process knowledge and judgment, will enable final status survey units to be identified and survey areas within them that may require special consideration to be recognized and addressed

As experience is gained by using the FSS design considerations, modifications may be made in accordance with Mallinckrodt's administrative control plan, provided fundamental criteria in Section 2.6 in the DP are not changed.

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39 DP §4.4.1.2.
40 Except for equipment identified in DP §4.4.1.1 or decontaminated to meet *Guidelines*.
41 DP §4.0, p. 4-1.
42 DP §4.4.1.1.
43 DG-4006, §B, p. 2.
44 NRC. "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material." Table 1. April 1993.
45 Not included in the 116 are those areas slated to be demolished.

4.6.3 Summary of FSS Design Process

The FSS design process relies on considerations in this section and on a designer's judgment to apply it. Survey design on accessible surfaces of a building, structure, or accessible surfaces of installed apparatus is addressed by the basic MARSSIM process . A survey designer should take advantage of knowledge of the history of the survey unit, especially of historical presence of regulated radioactive material in it. A survey designer should be aware of conditions that may require specific attention and deserve guidance for surveying. Conditions that may deserve such attention include:

- roofing;
- casework, *i.e.*, cabinetry, work-bench top, and shelving;
- interior of installed apparatus that could have become contaminated during CT operation, including;
- process apparatus, vessels, piping;
- ventilation exhaust hoods, filter housings, and exhaust duct;
- embedded piping, *e.g.*, clean-out trap, floor drain, or open sump; (Contaminated plumbing, piping, and sewerage below ground-level will be in the Phase 2 decommissioning plan.)
- a wall or other such surface that might have become contaminated during CT operation, now covered by and or access otherwise blocked by:
    - o apparatus installed after cessation of CT operation;
    - o or another wall, *e.g.,* a false wall as a result of remodeling
- paint and/or coated surface.

These would represent logical groups of survey design characteristics in addition to those expected to be processed routinely in accordance with MARSSIM methodology. For instance, survey of installed apparatus and or embedded piping might be included as judgment, or biased, measurements within a survey unit compromised mainly of building surfaces, *e.g.*, within a room, separate from the systematic survey.

Reminded by the final status survey design considerations to consider both general and specific design features, a final status survey designer will use his or her judgment to design each survey.

A large number of reference samples for a number of materials have been measured[46] already as part of the Characterization Survey process. It is possible, however, that additional measurements may be necessary for a given survey unit in order to satisfy the sampling criteria or to include new matrix materials for which no background data exist at this time. When new background data are required Mallinckrodt will specify the following information and requirements:

- matrix materials,

[46] Table 4.3 of the MI D Plan.

- number and location of data points
- and the type of equipment.

4.7 SURVEY METHOD AND SURVEY CONSIDERATIONS FOR BUILDING SURFACE AND INSTALLED APPARATUS

After the number of required samples has been established and the location of the statistical samples is determined systematically, the survey design must be completed. Both scans and stationary measurements at systematic and judgment (biased) locations must be included. Finalize the survey plan according to the following considerations: Items subject to survey are listed in DP Appendix A. Installed apparatus, equipment, roofing, walls, and other surfaces installed after cessation of C-T operation are considered unaffected and will be excluded from final status survey.

4.7.1 Systematic Measurements

Systematic, direct measurements will be obtained based upon locations determined in a survey design.

Systematic measurements will be made on an actual wall, floor, or ceiling surface where practical (instead of on intervening apparatus) and recorded.

Consider whether a wall or floor may have been contaminated before CT operation ceased and later covered by another wall or installed apparatus during renovation. In such circumstance, specify that the original, hidden wall or floor be accessed for systematic survey if practical. Else, specify biased survey of original surface as nearly representative nearby as practical.

Installed apparatus, such as process vessels, casework, cabinets, shelving, that will remain in use and are unlikely to be removed are considered part of the building structure surfaces and are subject to final status survey at the classification of the survey unit in which they are installed.

Locations of systematic survey and bias measurement locations on the exterior surfaces of installed apparatus are specified using the same grid system used for points which fall on the floor, ceiling or walls of the room. The interior of installed apparatus is not subject to systematic survey, only biased survey.

4.7.2 Inaccessible

In the event installed apparatus blocks a designated systematic survey point, alternatives are:

- Move the survey point as reasonably close as is accessible on the surface of interest; or

- Select a random location and measure it instead of the originally designated location; or

- Specify that the measurement point fall on the apparatus, if it was installed before cessation of CT operation.

4.7.3 Ventilation System

Process exhaust ventilation includes hoods, air ducts, and housings that might have become contaminated by exhausting air where regulated radioactive material was processed or handled in an open container.

Outside surfaces of process exhaust ventilation apparatus is subject to final status survey at the classification of the survey unit in which it is installed.

4.7.4 Roofing

Specify that systematic survey measurements be made on roofing locations that are prepared by removing loose gravel, by raking or combing, in order to detect potential radioactive contamination on the fixed surface.[47]

4.7.5 Biased Suveys

The survey designer will use process knowledge, existing data, and health physics "good practices" to design a survey in specific areas where contamination is likely to be found, especially on and around items of installed apparatus. The survey designer shall identify those areas that require bias surveys. The collection of additional bias survey measurements is left to the judgment of the survey designer and or surveyor and should focus on identifying areas of elevated radioactivity. New installed apparatus, equipment, roofing, walls and surfaces installed after cessation of use for CT are considered unaffected and will be excluded from biased surveys.

4.7.6 Internal Surveys

The following discussion of internal surveys is applicable to installed apparatus subject to the building release criteria.

A survey access point will be any accessible trap or other opening, such as open pipes, conduits, intakes, exhausts, inlets, outlets, sumps, supply line, product delivery chute, grate, or service opening. To the extent practical, access points will receive a survey by either direct measurement, scan, or smear to determine the potential for internal radioactivity within the equipment. The degree of effort to obtain internal access to

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[47] Weathering would seem likely to carry deposited particulate from loose gravel downward onto the fixed roof surface.

interior surfaces will rely on the FSS designer and surveyor insofar as applying a rational approach to the survey.

In the event internal surfaces of apparatus might be contaminated radioactively, access to survey the internals would be warranted if practical. Process knowledge and history of the apparatus or equipment will serve as one of the best indicators of potential for contamination within equipment. External surveys, especially at interior access points, may help decide the potential for existence of internal contamination. To the extent practical, survey points to assess the potential for contamination at internal access points will be decided by judgment of the surveyor on-site. Where multiple access points exist that, due to the functioning or history of the device, should give similar results (*e.g.*, tubes in a heat exchanger), measurements at a representative portion of the access points to provide an indication of the potential for internal radioactivity exceeding the release criteria will be decided by judgment of the surveyor on-site.

Internal surveys may be performed using a variety of methods, such as in the following examples. The specific methods to be used will rely on the professional judgment of the FSS designer and surveyor, feasibility of method, and potential for internal contamination. Some examples include direct surveys, scan surveys, or other surveys obtained by reaching inside access points; use of smaller diameter probes to obtain interior measurements; sample or survey sediment or residue in sumps and/or sediment accumulation areas; physically disassemble the device to obtain access to internals. Some examples include:

- direct surveys,
- scan surveys,
- surveys obtained by reaching inside access points,
- use of smaller diameter probes to obtain interior measurements,
- survey of sediment or residue samples from sumps and or sediment accumulation areas,
- physically disassembling the device to obtain access to internals, and internal surfaces of apparatus,
- equipment, or scrap which are likely to be contaminated in excess of $DCGL_W$, but are of such size, construction, or location as to make the surfaces inaccessible(i.e., no access points available) for purposes of measurement, shall be presumed to be contaminated in excess of building release limits.

4.7.7 Contingency

If direct or scan results in excess of the release criteria exist or are known to exist for countertops, cabinets, or other installed apparatus then, additional actions are required such as pulling out the associated casework in order to survey all areas.[48]

[48] If the survey unit is Class 2 or Class 3, then at least a portion of the survey unit must be reclassified and re-surveyed to Class 1.

4.7.8 Ventilation System

Process exhaust ventilation apparatus is subject to final status survey at the classification of the survey unit in which it is installed.

Specify a biased survey on accessible exterior surfaces in location(s) most likely to accumulate radioactive contamination. If measurements and scans on the accessible, external surface are less than $DCGL_W$, inaccessible, external areas are considered releasable without measurement.

Specify that accessible external areas of a hood be surveyed by a combination of biased direct measurements and scanning.

In the event process exhaust ventilation apparatus, including exhaust hoods and air ducting is proximate enough to a building surface to make surveying an external surface impractical, specify biased measurements on the apparatus and on the building surface as near as practical to the area between the apparatus and the building surface.

Apply knowledge of process, history of the ventilation system, scans on the exterior near air intake hoods and vent slots, and practicality of access to the inside to decide whether to specify survey of the inside of a process exhaust ventilation system.

If the survey does not find contamination in excess of the $DCGL_W$ in the hood, further biased survey inside the exhaust ventilation duct is not required.
Additional survey inside the ventilation system downstream should be specified if the survey does find contamination in excess of the $DCGL_W$ in the hood.

4.7.9 Roofing

On tar-gravel roofing, specify that biased measurements be made at locations that seem more likely to accumulate radioactive contamination and near rainwater drains and gutters. Specify that loose gravel be raked or combed away at each location to expose the solid surface to be measured.

4.7.10 Sump, Drain, and Plumbing

Potentially contaminated sumps, trenches, drains, and or piping in buildings that are candidates for unrestricted release are candidates for biased survey.

REFERENCES

U.S. NRC. Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM). Springfield, VA: National Technical Information Service; NUREG-1575/ EPA 402-R-97-016, December 1997.

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SECTION 5

QUALITY ASSURANCE

Mallinckrodt Inc.

Phase 1 Plan
For C-T Decommissioning

NRC Docket: 40-06563
NRC License : STB-401

January 9, 2002

5. QUALITY ASSURANCE

5. QUALITY ASSURANCE

Decommissioning activities will be performed in a manner to ensure the results are accurate and that uncertainties have been adequately considered. The quality assurance program will operate in all stages of decommissioning through the final survey, validation of the data, and the interpretation of the results to verify that this has occurred.

Persons or organizations responsible for ensuring that the quality assurance program has been established and verifying that activities affecting quality have been correctly performed will have sufficient authority, access to work areas, and organizational freedom to:

- identify quality problems;
- initiate, recommend, or provide solutions to quality problems through designated channels;
- verify implementation of solutions; and,
- ensure that further decommissioning activities are controlled until proper disposition of a nonconformance or deficiency has occurred.

Such persons or organizations will have direct access to responsible management at a level where appropriate action can be taken. Such persons or organizations will report to a management level such that required authority and organizational freedom are provided, including sufficient independence from cost and schedule considerations.

The major aspects of the quality assurance program for the decommissioning activities are discussed in the following sections.

5.1 QUALITY ASSURANCE PROJECT PLAN

For execution of decommissioning activities at the Mallinckrodt C-T Project, a Quality Assurance Project Plan (QAPP), consistent with applicable guidelines will be developed . The QAPP will be reviewed and approved by Mallinckrodt prior to implementation. The objective of the QAPP is to ensure confidence in the sampling, analysis, interpretation and use of radiological data generated during the decommissioning project.

The QAPP will ensure collection of reliable data by serving as the instrument of control for field and analytical activities associated with the project. Stated within the QAPP are the quality assurance policies, quality control criteria, and reporting requirements that must be followed by all site and contractor personnel when carrying out their assigned responsibilities on this project. The QAPP describes the functional activities and quality assurance/quality control (QA/QC) protocols necessary to collect data of adequate quality.

5.2 PROCEDURES

Supporting Quality Implementing Procedures (QIPs) will provide step-by-step details for complying with project QA requirements. The final radiological survey, including development of sampling plans, direct measurements, sample analysis, instrument calibration, daily functional checks of instruments, and sampling methods will be performed according to written procedures. These written procedures will be reviewed and approved by the Mallinckrodt project manager.

5.3 SUBCONTRACTORS

The activities to be conducted during decommissioning will require the services of a decommissioning contractor and various specialty subcontractors such as a qualified drilling contractor or a licensed surveyor. Contractor activities will be under the direct supervision of Mallinckrodt personnel in accordance with the QAPP. Subcontractor activities will be under the direct supervision of the decommissioning contractor personnel, also in accordance with the QAPP.

5.4 LABORATORY SERVICES

In the event off-site analysis of a sample is required, a qualified laboratory recommended by the decommissioning contractor and approved by Mallinckrodt will perform those radiological analytical laboratory services for the project. The laboratory will be responsible for all bench level QA/QC, data reduction, data reporting, and analytical performance monitoring. Laboratory accuracy will be evaluated by the analysis of blank and spiked samples. Sample handling protocols, analytical procedures, and reporting procedures employed by the analytical laboratory will be described in the laboratory's Quality Assurance Plan.

The off-site laboratory will be responsible for assuring that all appropriate laboratory personnel are thoroughly familiar with the Quality Assurance Project Plan and good laboratory practices, and that all appropriate laboratory personnel meet the requisite qualifications for their positions within the laboratory. The laboratory Director, or his equivalent/representative, will review and approve all reports. The Director will also be responsible for assuring laboratory personnel have appropriate training to perform assigned responsibilities, and for daily management of the laboratory and its staff.

The off-site laboratory will have a QA designee who will be responsible for assuring that the QA/QC requirements of the QAPP, the laboratory Quality Assurance Plan, and its associated operating procedures are strictly followed. The QA designee will be responsible for review of data, alerting the Mallinckrodt decommissioning Project Manager and the Contractor Project Manager of the need for corrective action (when necessary), performing internal audits as specified by the QAPP, and maintenance of the QC records. The QA designee will also be responsible for preparing project specific QA/QC plans, as necessary.

5.5 SURVEYS AND SAMPLING ACTIVITIES

Trained individuals following written procedures will perform surveys using properly calibrated instruments. The custody of samples will be tracked from collection to analysis. Final survey data will be retained until License STB-401 is terminated by the NRC.[1] The designated sampler or analytical laboratory will collect a split sample when desired by the NRC to obtain samples that are duplicates of those to be analyzed. When this operation is performed, the procedure for obtaining duplicate samples will be followed.

QC hold points will be utilized as necessary to ensure quality of surveys and sampling. Hold points will also be used to ensure that debris is moved only after QA has verified that the proper sampling and survey information for the debris in question has been obtained.

5.6 DOCUMENTATION

Data will be recorded and documented in a data management system. Entries will include the location of the survey or sampling point on the appropriate building grid. Data management personnel will also ensure that chain-of-custody and data management procedures are followed for decommissioning-related samples. The decommissioning contractor's procedures for proper handling, shipping and storage of samples will be used.

Both direct measurements and analytical results will be documented. The results for each survey measurement or sample and its grid block location, will be listed in tabular form (*i.e.*, result versus sample or survey location).

Data will be recorded in an orderly and verifiable way and reviewed for accuracy and consistency. Every step of the decommissioning process, from training personnel to calculating and interpreting the data, shall be documented in a way that lends itself to audit. Records of training to demonstrate qualification will also be maintained.

5.7 EQUIPMENT MAINTENANCE AND CALIBRATION

Measuring equipment will be maintained, calibrated, and tested according to Regulatory Guides 4.15 and 4.16 recommendations. Further, the procedures, responsibilities, and schedules for calibrating and testing equipment will be documented.

Proper maintenance of equipment varies, but maintenance information and use limitations are provided in the vendor documentation. Measuring and analyzing equipment will be tested and calibrated before initial use and will be recalibrated if maintenance or modifications could invalidate earlier calibrations. Field and laboratory equipment, specifically used for obtaining final radiological survey data, will be calibrated based on standards traceable to NIST. In those cases where NIST-traceable standards are not available, standards of an industry-recognized organization (for example, the New

[1] 10 CFR Part 40.36(f)

Brunswick Laboratory for various uranium standards) will be used. Minimum frequencies for calibrating equipment will be established and documented.

Measuring equipment will be tested at least once on each day the equipment is used. Test results will be recorded in tabular or graphic form and compared to predetermined, acceptable performance ranges. Equipment that does not conform to the performance criteria will be promptly removed from service until the deficiencies can be resolved.

5.8 DATA MANAGEMENT

5.8.1 Laboratory Data

Data reduction, QC review, and reporting will be the responsibility of the analytical laboratory. Data reduction includes all automated and manual processes for reducing or organizing raw data generated by the laboratory. The laboratory will provide a data package for each set of analyses that will include a copy of the raw data in electronic format, and any other information needed to check and recalculate the analytical results.

Once a data package is received from the laboratory, the analytical results and pertinent QA/QC data will be compiled onto standardized data formats. The data packages will serve as basic reference sheets for data validation, as well as for project data use.

5.8.2 Field Survey Data

The generation, handling, computations, evaluation and reporting of final radiological survey data will be as specified in the decommissioning contractor's procedures. Included in these procedures will be a system for data review and validation to ensure consistency, thoroughness and acceptability. Qualified health and safety, operations, and/or engineering personnel will review and evaluate survey data.

5.8.3 Data Evaluation

Prior to releasing data for use by project staff, selected data will undergo data evaluation based on intended end use of the data. Data points chosen for evaluation will be examined to determine compliance with QA requirements and other factors that determine the quality of the data. Data taken during a characterization survey will be subjected to quality verification before use as FSS data. Data taken during a prior survey, e.g., characterization survey, may be usable as FSS data provided the data are subjected to quality verification and satisfy data quality objectives.

If sample data are rejected or data omissions are identified during the data validation, this data will be evaluated to judge the impact on the project. Other corrective action may include re-sampling and analyzing, evaluating and amending sampling and analytical procedures and accepting data acknowledging the level of uncertainty.

In the event final status survey data are processed by computer, the application program[2] and each modification thereof will be verified to perform as intended before its initial use. A knowledgeable person will verify that the algorithms are as intended and will compare an instance of computer-generated result and an independently derived result of the same process. Mallinckrodt will document the application program, including its algorithms and a listing or copy of the program.

5.9 SAMPLE CHAIN-OF-CUSTODY

One of the most important aspects of sample management is to ensure that the integrity of the sample is maintained; that is, that there is an accurate record of sample collection, transport, analysis, and disposal. 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.

Sample custody will be assigned to one individual at a time. This will prevent confusion of responsibility. Custody is maintained when (1) the sample is under direct surveillance by the assigned individual, (2) the sample is maintained in a tamper-free container, or (3) the sample is within a controlled-access facility.

The individual responsible for sample collection will initiate a chain-of-custody record using a standard form provided by the decommissioning contractor. A copy of this form will accompany the samples throughout transportation and analyses; and any breach in custody or evidence of tampering will be documented.

5.10 AUDITS

Periodic audits will be performed to verify that decommissioning activities comply with established procedures and other aspects of the QAPP and to evaluate the overall effectiveness of the QA program. Mallinckrodt and Contractor Quality Assurance personnel will verify that qualified personnel are used to conduct audits to ensure that the applicable procedures are being properly implemented. The audits will be conducted on at least a quarterly basis, in accordance with written guidelines or checklists. Health and safety personnel will also conduct semiannual audits in their area of concern. External program audits may also be used at the discretion of either Mallinckrodt or contractor management. Audit results will be reported to both Mallinckrodt and contractor management in writing, and actions to resolve identified deficiencies will be tracked and appropriately documented.

[2] An *application program* consists of instructions and or algorithms created specifically for processing data for the CT decommissioning project. It does not pertain to generic software, including for example, a spreadsheet program such as Microsoft EXCEL™ or a database program such as Microsoft ACCESS.™

SECTION 6

FINANCIAL ASSURANCE

Mallinckrodt Inc.

Phase 1 Plan
For C-T Decommissioning

NRC Docket: 40-06563
NRC License : STB-401

January 9, 2002

6. FINANCIAL ASSURANCE

The work required to reduce the level of residual radioactive material within the area of the Mallinckrodt St. Louis Plant where C-T operations were performed has been divided into two phases. Phase I is the above-grade decommissioning of the buildings and equipment. Phase II completes the decommissioning of the C-T Project by addressing the grade and below-grade structures and soil.

The remainder of this section is being updated and will be submitted soon.

APPENDIX A
CLASSIFICATION FOR SURVEYS

Mallinckrodt Inc.

Phase 1 Plan
For C-T Decommissioning

NRC Docket: 40-06563
NRC License : STB-401

January 9, 2002

APPENDIX A

CLASSIFICATION FOR SURVEYS

These classifications are bases for designing final status surveys of buildings and release surveys of equipment.

Each item classified in Appendix A as either *Impacted Class 1, Impacted Class 2, or Impacted Class 3* is subject to a final status survey for conformance to building release criteria. Each item of equipment classified as either *Impacted Category 1 or Impacted Category 2* is subject to survey for equipment release criteria. A release survey is not required for other items not identified as either *Impacted Class 1, Impacted Class 2, Impacted Class 3, Impacted Category 1, or Impacted Category 2,* or that were designated for disposal in an approved facility, *e.g.*, N/A or Demolish. Structures erected and apparatus installed (except if installed in Building 238, 247, or 248) after cessation of C-T operation are not subject to final status survey nor removable equipment release survey.

Characterization survey data "Results" represent the range of beta radiation measurements on surfaces, *i.e.*, (low)/(high) areal density. Negative numbers in the "Result" column are statistical artifacts. Net beta reported is derived by subtracting open window ($\beta+\gamma$) minus closed window (γ). Since gross $\beta+\gamma$ or γ measurements may be a few thousand cpm, one standard deviation of a gross count may be near one hundred. Thus, the counting error may propagate to produce the negative results tabulated. To aid statistical analysis, the measured result, even if it is a negative number, is recorded.[1]

Characterization survey data and conservative "Criteria" were employed to classify the locations surveyed. Release limits are specified in DP section 2.2

Wherever a coating or paint is mentioned, the characterization survey was performed before paint or coating was applied. The characterization survey results represent measurements made before the paint or coating was applied.

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[1] Currie, L.A., *Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements,* NUREG/CR-4006, p. 76, 1984.

Appendix A

Area	Bldg.	Location / Surface	Result*	Evaluation	Classification
1	62	Tile Floors - Men's Area	77/106	Clean	Non Impacted
2	62	Tile Floors - Women's Area	-80/-40	Clean	Non Impacted
3	90	1st Floor Machinery	0/42	Clean	Non Impacted
4	90	1st Floor and Drains etc.: New 2nd floor drains tied into mains on 1st floor after CT operations ceased.	TBD[1]	Assumed Clean	Non Impacted
5	90	2nd Floor and Drains etc.: 8" masonry wall built approximately 8' from existing brick center main bearing wall with metal roof covering for hallway to 91-2 auditorium. West wall of hallway is studded and drywalled with drop ceiling. Brick above hallway to roof is exposed.	TBD	Assumed Clean	Non Impacted
6	91	1st Floor Machinery: several pieces of machinery, lathes, milling machines, drill presses, have been replaced since CT operations have ceased.	0/42	Clean	Non Impacted
7	91	1st Floor and Drains etc.: New 2nd floor drains tied into mains on 1st floor after CT operations ceased.	TBD	Assumed Clean	Non Impacted
8	91	2nd Floor and Drains: New sewers and drains installed for 2 new restrooms, utility room and kitchen to accommodate new auditorium. All drains tied into main on first floor. Modifications performed after CT operations ceased.	TBD	Assumed Clean	Non Impacted

* Mean/95 % Confidence (βpm/100 cm^2)

Appendix A

Area	Bldg.	Location / Surface	Result*	Evaluation	Classification
9	91	2nd Floor Crane Room: New offices, restrooms, kitchen, utility room, auditorium and lobby erected. All walls studded and drywalled. 8 new A/C- HVAC units installed above roof of auditorium and offices (inside) Crane is still in place. Gable of roof and approximately 8-12' of walls above new additions are exposed. Walls directly north of auditorium are exposed. All new additions after CT operations ceased.	343/389	Clean	Non Impacted
10	91	2nd Floor crane pad.	TBD	Assumed Clean	Impacted Class 3
11	101	Former Gas Heaters/Roof Equipment: Additional AC units installed after CT operations ceased.	93/148	Clean	Impacted Category 2
12	101	Incinerator Pad Area	553/868	Contaminated	Impacted Class 2 (Phase II)
13	101	Incinerator Unit/Equipment	935/1777	Contaminated	Impacted Category 1
14	101	External Walls: Painted once since CT operations ceased. One coat of paint over original paint.	TBD	Assumed Clean	Impacted Class 3
15	101	Roof and Decking: Roof membrane and portions of insulation replaced 10/94. Original metal roof decking.	318/368	Clean	Impacted Class 3
16	200E	Roof: Roof repaired in several spots where old equipment was removed. New roof installed on building annex east side after CT operations ceased. Original roof decking.	TBD	Assumed Clean	Impacted Class 3
17	200E	Roof Equipment: New FRP panels installed on roof and side of stairwell penthouse. All equipment, blowers, condensers, piping and piping supports removed and disposed of. New exhaust blower installed on building annex eastside.	211/409	Clean	Impacted Category 2
18	200W	Roof: Roof replaced in spring of 2000, original roof decking.	TBD	Assumed Clean	Impacted Class 3

* Mean/95 % Confidence (βpm/100 cm^2)

Appendix A

Area	Bldg.	Location / Surface	Result*	Evaluation	Classification
19	200W	Roof Equipment: New FRP scrubber system with FRP ductwork installed in center of building roof. 1 old HVAC system on northwest side of building roof was removed and disposed of after CT operations ceased.	8/136	Clean	Impacted Category 2
22	204	Roof: Repaired in areas where new equipment was installed and old calciner stacks were removed, all performed post CT.	TBD	Assumed Clean	Impacted Class 3
23	204	Roof Equipment: 2 new 3 ton A/C units installed on southwest corner of building roof. 1 new HVAC unit (gas fired) installed on northwest corner of building roof. One new building ventilation (air intake) stack installed at center of building roof. Two exhaust stacks for calciner removed and disposed of. All modifications performed in 2000. Three blowers and the HVAC system still remain on roof.	116/624	Clean	Impacted Category 2
24	213	Pump Room - Interior Walls	1/78	Clean	N/A - Demolish
25	213	Pump Room - Floor	310/509	Clean	N/A - Demolish
26	213	Pump Room - Ceiling	129/272	Clean	N/A - Demolish
27	213	Pump Room - Equipment	-126/18	Clean	N/A - Demolish
28	213	North Room - Interior Walls	386/562	Clean	N/A - Demolish
29	213	North Room - Floor	1748/3415	Contaminated	N/A - Demolish
30	213	North Room - Ceiling	63/183	Clean	N/A - Demolish
31	213	North Room - Equipment/Pipe	6500/13000	Contaminated	N/A - Demolish
32	213	North Foyer - Interior Walls	716/1017	Contaminated	N/A - Demolish
33	213	North Foyer - Floor	698/1852	Contaminated	N/A - Demolish
34	213	North Foyer - Ceiling	28/205	Clean	N/A - Demolish
35	213	South Foyer - Interior Walls	-153/-38	Clean	N/A - Demolish
36	213	South Foyer - Floor	182/287	Clean	N/A - Demolish
37	213	South Foyer - Equipment	TBD	Assumed Clean	N/A - Demolish

* Mean/95 % Confidence (βpm/100 cm^2)

Appendix A

Area	Bldg.	Location / Surface	Result*	Evaluation	Classification
38	213	South Foyer - Ceiling	79/299	Clean	N/A - Demolish
39	213	South Foyer - Ceiling Pipe	1500/5400	Contaminated	N/A - Demolish
40	213	Women's BR - Interior Walls	-34/103	Clean	N/A - Demolish
41	213	Women's BR - Floor	226/395	Clean	N/A - Demolish
42	213	Women's BR - Ceiling	-11/149	Clean	N/A - Demolish
43	213	Women's BR - Pipe/Equipment/Ledge	5723/11000	Contaminated	N/A - Demolish
44	213	Men's BR - Interior Walls	173/377	Clean	N/A - Demolish
45	213	Men's BR - Ledge/Pipe/Equipment	1845/2023	Contaminated	N/A - Demolish
46	213	Men's BR - Floor	336/724	Clean	N/A - Demolish
47	213	Men's BR - Ceiling	139/451	Clean	N/A - Demolish
48	213	Breakroom - Interior Walls	TBD	Assumed Clean	N/A - Demolish
49	213	Breakroom - Ledges/Pipe	TBD	Assumed Clean	N/A - Demolish
50	213	Breakroom - Floor	96/126	Clean	N/A - Demolish
51	213	Breakroom - Ceiling	TBD	Assumed Clean	N/A - Demolish
52	213	Roof	TBD	Assumed Contaminated	N/A - Demolish
53	213	Cooling Tower	TBD	Assumed Contaminated	N/A - Demolish
54	213	South Exterior Wall	111/185	Clean	N/A - Demolish
55	213	North Exterior Wall	2400/3300	Contaminated	N/A - Demolish
56	213	West Exterior Wall	155/256	Clean	N/A - Demolish
57	213	213/238 Pipe Rack	TBD	Assumed Contaminated	N/A - Demolish
58	213	213/238 Steam Line Expansion	TBD	Assumed Contaminated	N/A - Demolish
59	214	Interior Walls	232/528	Clean	N/A - Demolish
60	214	Floor	1245/1984	Contaminated	N/A - Demolish
61	214	Above Ceiling Tile Support	1344/4775	Contaminated	N/A - Demolish
62	214	Exterior Walls	496/1202	Assumed Contaminated	N/A - Demolish
63	214	Equipment	TBD	Assumed Clean	N/A - Demolish
64	214	East Side Pipe Rack	TBD	Assumed Contaminated	N/A - Demolish
65	214	Roof	TBD	Assumed Contaminated	N/A - Demolish
66	219	Foam/Metal: Building was dismantled and disposed of in 1998.	825/1063	FUSRAP	N/A - Demolished

* Mean/95 % Confidence (βpm/100 cm^2)

Appendix A

Area	Bldg.	Location / Surface	Result*	Evaluation	Classification
67	219	Concrete Supports: Cut off at grade with pavement.	136/484	FUSRAP	N/A - Demolished
68	219	Roof Steel and Material: Building was dismantled and disposed of in 1998.	309/654	FUSRAP	N/A - Demolished
69	222	Roof: repaired where new equipment was installed. New equipment installed consisted of 2 new HVAC units with ductwork down to 1st floor outside on west wall of building. Two new process exhaust blowers located on east side of building roof. Equipment installed 1997-1998.	TBD	Assumed Clean	Impacted Class 3
70	223	Roof: Repaired when 2 new building exhaust fans were installed. Repairs and equipment installed in 2000.	TBD	Assumed Clean	Impacted Class 3
71	235	East Room - Interior Walls: One new 36" mandoor located on southeast corner installed for accessing new hydrogenation room. Several new penetrations in wall to accommodate piping modifications. This work was performed in 1998.	22/81	Clean	Impacted Class 3
72	235	East Room - Floor: Epoxy coating applied in July 1998.	-41/9	Clean	Impacted Class 3
73	235	East Room - Ceiling	160/287	Clean	Impacted Class 3

* Mean/95 % Confidence (βpm/100 cm^2)

Appendix A

Area	Bldg.	Location / Surface	Result*	Evaluation	Classification
74	235	West Room - Interior Walls: New dryer room installed in 1998 at southwest corner of main room. Room is approximately 29' 6" long and 20' deep with 18' ceilings. New 8" masonry walls on east side and north side with 4' double doors for each room. South walls of each room have old brick exposed and have been coated with one coat of epoxy paint. West room drier is also old brick with epoxy coating. New vault installed in 1996-1997 approximately 44' long and 24' 9" deep with 16' ceiling. There is a 2' 8" chase between west wall of vault and old brick interior wall leaving brick exposed.	-163/113	Clean	Impacted Class 3
75	235	West Room - Floor: Epoxy coating installed 7-98	197/280	Clean	Impacted Class 3
76	235	West Room - Expansion Joint: Sealed with epoxy when new floor coating was installed.	341/571	Clean	Impacted Class 3
77	235	West Room - Ground Rod Holes	884/18000	Contaminated	Impacted Class 1
78	235	West Room - Ceiling/ I Beams	172/253	Clean	Impacted Class 3
79	235	North Exterior Wall: New drum pumping station locating on north east wall of building fabricated from steel and FRP siding - installed 6-00. Station is approximately 26' long and 17' wide with 10' sloped roof. New concrete foundation and floor. Old brick of north wall of building exposed inside south wall of filling station.	-46/13	Clean	Impacted Class 3
80	235	South Exterior Wall: New oxidizer installed on southwest corner of building in 1996. New concrete pad for oxidizer poured above grade, 10 to 12" thick. New FRP scrubber system installed on south wall center of building in 1998. Pad poured above grade for scrubber.	159/328	Clean	Impacted Class 3

* Mean/95 % Confidence (ßpm/100 cm^2)

Appendix A

Area	Bldg.	Location / Surface	Result*	Evaluation	Classification
81	235	East Exterior Wall: New 4' thick foundation and pad poured in 1998. New steel and FRP siding structure installed over pad for hydrogenation room in 1998. Several holes core drilled in bldg. 235 east wall for piping. Approximately 4' - 5' chase between hydrogenation bldg. and existing east wall. Bricks are exposed. Thirty-five feet of exterior brick to the height of 12 feet was removed starting at the south east corner of the building prior to the building of the hydrogenator. This work was performed in the fall of 1997.	1306/1720	Contaminated	Impacted Class 2
82	235	Roof Piping and Equipment: several new roof penetrations since CT operations. New piping and conduit installed across roof on supports with sleepers for hydrogenation room expansion.	TBD	Assumed Contaminated	Impacted Category 2
83	235	Roof	TBD	Assumed Contaminated	Impacted Class 2
84	236	Interior Walls/Ledges: New offices, locker rooms, showers and security office constructed inside of building within 2 floors. All walls are studded and drywalled covering inside brick surfaces. Areas where old brick is exposed are as follows: North wall approximately 35' long by 10' high. West wall approximately 36' long by 10' high. Men's shower area east wall south approximately 18' long by 7' 6" high. Upper level janitors room east wall north side, approximately 8' long by 10' high. All building modifications were performed in 2000. All exposed brick wall coated with one layer of epoxy paint.	1326/1935	Contaminated	Impacted Class 2

* Mean/95 % Confidence (ßpm/100 cm^2)

Appendix A

Area	Bldg.	Location / Surface	Result*	Evaluation	Classification
85	236	Floor: An area of 48' by 25' of concrete was removed in 2000 for construction of the locker room and offices. This area is 18' south of north wall. It consists of new sewer ties for locker room, showers and restrooms. Floor is tile except for showers which are epoxy coated or ceramic tile.	23/180	Clean	Impacted Class 3
86	236	Ceiling/Equipment: All new drop ceilings installed for new offices on 2nd level. Underside of roof accessible through small amount of space above drop ceiling.	107/192	Clean	Impacted Class 3 / Category 2
87	236	North Exterior Wall: All windows and sills removed. New 8" masonry block walls installed. Two new 36" mandoors installed, one on the upper level and one on the lower level. Both doors are located on the northeast corner of the building.	312/578	Clean	Impacted Class 3
88	236	South Exterior Wall/Ledges: Windows and sills and one mandoor removed. Openings on windows blocked up with 8" masonry blocks. Mandoor blocked up with 8" masonry block with brick outside.	1107/1939	Contaminated	Impacted Class 2
89	236	West Exterior Wall/Ledges: Windows and sills, sliding door removed, windows blocked with 8" masonry block. Sliding door replaced with two new 36" mandoors and new window above door headers for 2nd floor offices.	1246/1973	Contaminated	Impacted Class 2
90	236	Roof: Roof replaced 5-2000, original decking.		Removed	Non-Impacted
91	236	Roof Equipment: New HVAC units installed with piping and conduit in 2000.		Removed	Non-Impacted
92	238	Main Room - North Interior Wall (PRI)	5123/7113	Contaminated	N/A - Demolish
93	238	Main Room - South Interior Wall (PRI)	8461/12000	Contaminated	N/A - Demolish

* Mean/95 % Confidence (βpm/100 cm^2)

Appendix A

Area	Bldg.	Location / Surface	Result*	Evaluation	Classification
94	238	Main Room - East Interior Wall (PRI)	3520/5000	Contaminated	N/A - Demolish
95	238	Main Room - West Interior Wall (PRI)	8800/11500	Contaminated	N/A - Demolish
96	238	Main Room - Floor (PRI)	13000/14200	Contaminated	N/A - Demolish
97	238	Main Room - Overheads (PRI)	4000/5500	Contaminated	N/A - Demolish
98	238	Main Room - 1st Floor Equipment (PRI)	3240/5800	Contaminated	N/A - Demolish
99	238	Mezzanine - North Interior Wall (PRI)	109/322	Assumed Contaminated	N/A - Demolish
100	238	Mezzanine - South Interior Wall (PRI)	1969/2931	Contaminated	N/A - Demolish
101	238	Mezzanine - East Interior Wall (PRI)	1053/2082	Contaminated	N/A - Demolish
102	238	Mezzanine - West Interior Wall (PRI)	1334/2224	Contaminated	N/A - Demolish
103	238	Mezzanine - Floor (PRI)	5470/6359	Contaminated	N/A - Demolish
104	238	Mezzanine - Equipment (PRI)	4211/7000	Contaminated	N/A - Demolish
105	238	Mezzanine - Overheads (PRI)	2628/4000	Contaminated	N/A - Demolish
106	238	Mezzanine - Ceiling (PRI)	524/788	Assumed Contaminated	N/A - Demolish
107	238	S. Switch Gear Room - Interior Walls (PRI)	3627/5068	Contaminated	N/A - Demolish
108	238	S. Switch Gear Room - Floor (PRI)	4754/7614	Contaminated	N/A - Demolish
109	238	S. Switch Gear Room - Equipment	TBD	Assumed Clean	N/A - Demolish
110	238	S. Switch Gear Room - Roof	519/968	Contaminated	N/A - Demolish
111	238	S. Switch Gear Room - Exterior Walls, East	308/1031	Clean	N/A - Demolish
112	238	S. Switch Gear Room - Exterior Walls, West	63/230	Clean	N/A - Demolish
113	238	S. Switch Gear Room - Exterior Walls, South	128/128	Clean	N/A - Demolish
114	238	S. Switch Gear Room - Ceiling (PRI)	749/1620	Contaminated	N/A - Demolish
115	238	N. Switch Gear Room - Interior Walls (PRI)	2395/3979	Contaminated	N/A - Demolish
116	238	N. Switch Gear Room - Floor	13360/20800	Contaminated	N/A - Demolish
117	238	N. Switch Gear Room - Equipment	TBD	Assumed Clean	N/A - Demolish
118	238	N. Switch Gear Room - Roof	598/1404	Contaminated	N/A - Demolish
119	238	N. Switch Gear Room - Exterior Walls	2326/11400	Contaminated	N/A - Demolish
120	238	N. Switch Gear Room - Ceiling (PRI)	262/486	Clean	N/A - Demolish
121	238	W. Switch Gear Room - Exterior Walls	92/220	Clean	N/A - Demolish
122	238	Ball Mill Room, 1st Floor - Interior Walls (PRI)	3360/4726	Contaminated	N/A - Demolish
123	238	Ball Mill Room, 1st Floor - Floor (PRI)	11300/13500	Contaminated	N/A - Demolish

* Mean/95 % Confidence (βpm/100 cm^2)

Appendix A

Area	Bldg.	Location / Surface	Result*	Evaluation	Classification
124	238	Ball Mill Room, 1st Floor - Equipment (PRI)	11240/17078	Contaminated	N/A - Demolish
125	238	Ball Mill Room, Mezzanine - Interior Walls (PRI)	2130/3679	Contaminated	N/A - Demolish
126	238	Ball Mill Room, Mezzanine - Floor (PRI)	5756/8432	Contaminated	N/A - Demolish
127	238	Ball Mill Room, Mezzanine - Exterior Walls	866/2063	Contaminated	N/A - Demolish
128	238	Ball Mill Room, Mezzanine - Ceiling	TBD	Assumed Contaminated	N/A - Demolish
129	238	Ball Mill Room, Mezzanine - Roof/Equipment	TBD	Assumed Contaminated	N/A - Demolish
130	238	Ball Mill Room, Mezzanine - Equipment	11240/17100	Contaminated	N/A - Demolish
131[2]	238	Blender Room, 1st Flr - Interior Walls (PRI)	2368/3900	Contaminated	N/A - Demolish
132[2]	238	Blender Room, 1st Flr - Floor (PRI)	6587/8466	Contaminated	N/A - Demolish
133[2]	238	Blender Room, 1st Flr - Ceiling/Beams (PRI)	1491/2543	Contaminated	N/A - Demolish
134[2]	238	Blender Room, 1st Flr - Equipment-All (PRI)	1139/1534	Contaminated	N/A - Demolish
135[2]	238	Blender Room, 2nd Flr - West Wall (PRI)	320/496	Clean	N/A - Demolish
136[2]	238	Blender Room, 2nd Flr - Floor (PRI)	2979/9715	Contaminated	N/A - Demolish
137[2]	238	Blender Room, 3rd Flr - West Wall (PRI)	238/238	Clean	N/A - Demolish
138[2]	238	Blender Room, 3rd Flr - Floor/Landing (PRI)	1070/1070	Assumed Contaminated	N/A - Demolish
139[2]	238	Blender Room, 3rd Flr - Roof/Equipment	TBD	Assumed Contaminated	N/A - Demolish
140[2]	238	Blender Room, 3rd Flr - Exterior Walls	4870/8060	Contaminated	N/A - Demolish
141	238	Metal Disv. Room, 2nd Flr - S/E Walls (PRI)	2228/4323	Contaminated	N/A - Demolish
142	238	Metal Disv. Room, 2nd Flr - Floor (PRI)	1986/3551	Contaminated	N/A - Demolish
143	238	Metal Disv. Room, 2nd Flr - Interior Walls (PRI)	544/957	Assumed Contaminated	N/A - Demolish
144	238	Metal Disv. Room, 2nd Flr - Floor (PRI)	1193/2208	Contaminated	N/A - Demolish
145	238	Metal Disv. Room, 2nd Flr - Ceiling	TBD	Assumed Contaminated	N/A - Demolish
146	238	Metal Disv. Room, 2nd Flr - Roof	TBD	Assumed Contaminated	N/A - Demolish
147	238	Metal Disv. Room, 2nd Flr - Exterior Walls	359/1098	Assumed Contaminated	N/A - Demolish
148	238	Metal Disv. Room, 2nd Flr - Equipment (PRI)	551/828	Assumed Contaminated	N/A - Demolish
149	238	Calciner Room, 1st Flr - Interior Walls (PRI)	259/370	Clean	N/A - Demolish
150	238	Calciner Room, 1st Flr - Floor (PRI)	766/1366	Contaminated	N/A - Demolish
151	238	Calciner Room, 1st Flr - Equipment (PRI)	351/380	Assumed Contaminated	N/A - Demolish

* Mean/95 % Confidence (βpm/100 cm^2)

Appendix A

Area	Bldg.	Location / Surface	Result*	Evaluation	Classification
152	238	Calciner Room, 2nd Flr - Interior Walls (PRI)	48/157	Clean	N/A - Demolish
153	238	Calciner Room, 2nd Flr - Floor (PRI)	667/934	Assumed Contaminated	N/A - Demolish
154	238	Calciner Room, 2nd Flr - Ceiling (PRI)	84/84	Clean	N/A - Demolish
155	238	Calciner Room, 2nd Flr - Equipment (PRI)	348/374	Assumed Contaminated	N/A - Demolish
156	238	Calciner Room, 2nd Flr - Exterior Walls	194/323	Clean	N/A - Demolish
157	238	Calciner Room, 2nd Flr - Roof	TBD	Assumed Contaminated	N/A - Demolish
158	238	North Exterior Wall	4471/8411	Contaminated	N/A - Demolish
159	238	South Exterior Wall	4029/5865	Contaminated	N/A - Demolish
160	238	East Exterior Wall	5394/10169	Contaminated	N/A - Demolish
161	238	West Exterior Wall	1214/3418	Contaminated	N/A - Demolish
162	238	East Pipe Rack	367/643	Clean	N/A - Demolish
163	238	Roof (PRI)	TBD	Assumed Contaminated	N/A - Demolish
164	238	Roof Equipment	10586/16236	Contaminated	N/A - Demolish
165	238	Outside Equipment East Side	450/970	Assumed Contaminated	N/A - Demolish
166	238	Outside Equipment North Side	TBD	Assumed Contaminated	N/A - Demolish
167	238	Outside Equipment South Side (PRI)	10900/21500	Contaminated	N/A - Demolish
168	240	East Exterior Wall/Ledges	-143/-38	Clean	Impacted Class 3
169	240	North Exterior Wall/Ledges	-53/84	Clean	Impacted Class 3
170	240	Floor/Doorways	43/110	Clean	Impacted Class 3
171	240	Roof: Repairs in 2000, new A/C unit in 1999.	TBD	Assumed Clean	Impacted Class 3
172	245	Interior Walls: New concrete and steel vault on east side of building approximately 80' long and 37' deep with a 16' ceiling. Small chase above vault for maintenance. Packaging area west side of building is studded out and FRP siding installed over top. Packaging area center of building is studded and drywalled. Office area, restroom, janitors closet in center of building is studded and drywalled with epoxy paint.	40/159	Clean	Non Impacted

* Mean/95 % Confidence (βpm/100 cm^2)

Appendix A

Area	Bldg.	Location / Surface	Result*	Evaluation	Classification
173	245	Floor/Drains: All of building floor is a 10-14" concrete overlay poured over old floor. Sewers taken out of service. No sewers in packaging area. One new sewer for restroom and janitors closet installed on south wall of building. All construction done in 1997-1998.	72/253	Clean	Non Impacted
174	245	Ceiling: All new drop ceilings in all packaging area and office area. Ceiling of building exposed over vault.	TBD	Assumed Clean	Non Impacted
176	245	North Exterior Wall: Foundation and pad poured and steel with FRP siding erected at center of north wall for loading dock. Exterior painted with epoxy paint. 1997	180/247	Clean	Impacted Class 3
177	245	South Exterior Wall: New 4' double door installed on 1st level. New 36" mandoor installed on 2nd level. New stairs installed to 2nd level. Wall painted with epoxy paint	TBD	Assumed Clean	Impacted Class 3
178	245	Switch Gear, South Wall	TBD	Removed	N/A
179	245	East Exterior Wall: New USP water system room constructed 2000. Constructed in area vacated by spray drier. New room is approximately 24' X 33' with 18' ceiling. New 8" block walls on north and east side. Old east brick wall and south 8" masonry block wall from original spray drier room are exposed and painted with epoxy coating.	217/328	Clean	Impacted Class 3
180	245	West Exterior Wall/Equipment: Equipment removed 1996. Exterior painted with epoxy paint.	53/688	Assumed Clean	Impacted Category 2
181	245	West Exterior Pad: Removed 1997	983/2058	Removed	N/A - Removed
182	245	Exterior Pipe, NW Corner: Most of exterior pipe from CT era has been removed.	173/229	Clean	Impacted Category 2
191	246A	Tile Floor	197/230	Assumed Contaminated	N/A - Demolish

* Mean/95 % Confidence (βpm/100 cm^2)

Appendix A

Area	Bldg.	Location / Surface	Result*	Evaluation	Classification
192	246A	Interior Walls & Ceiling	-289/251	Clean	N/A - Demolish
193	246A	Roof, Equipment	TBD	Assumed Contaminated	N/A - Demolish
194	246A	Roof Metal Shed, Equipment	TBD	Assumed Contaminated	N/A - Demolish
195	246A	East Wall Pipe Rack	TBD	Assumed Contaminated	N/A - Demolish
196	246A	East Exterior Wall	-106/102	Clean	N/A - Demolish
197	246A	North Exterior Wall	252/404	Clean	N/A - Demolish
198	246A	South Exterior Wall	233/964	Clean	N/A - Demolish
199	246A	246A/238 Pipe Rack	TBD	Assumed Contaminated	N/A - Demolish
200	246A	246A/238 Steam Line Expansion	TBD	Assumed Contaminated	N/A - Demolish
201	246B	1st Floor - Interior Walls (PRI)	614/844	Assumed Contaminated	N/A - Demolish
202	246B	1st Floor - Floor (PRI)	334/427	Assumed Contaminated	N/A - Demolish
203	246B	1st Floor - Ceiling	TBD	Assumed Contaminated	N/A - Demolish
204	246B	1st Floor - Equipment (PRI)	619/881	Assumed Contaminated	N/A - Demolish
205	246B	2nd Floor - Interior Walls/Ceiling (PRI)	1370/2634	Contaminated	N/A - Demolish
206	246B	2nd Floor - Floor (PRI)	1289/1802	Contaminated	N/A - Demolish
207	246B	2nd Floor - Equipment (PRI)	1428/2737	Contaminated	N/A - Demolish
208	246B	North Exterior Wall	35/228	Clean	N/A - Demolish
209	246B	South Exterior Wall	165/298	Assumed Contaminated	N/A - Demolish
210	246B	Roof	TBD	Assumed Contaminated	N/A - Demolish
211	246B	Roof Equipment	283/537	Clean	N/A - Demolish
212	246B	246B/238 Elevated Walkway	TBD	Assumed Contaminated	N/A - Demolish
213	246B	Elevated Walkway Roof	282/716	Clean	N/A - Demolish
214	247A	1st Floor - Interior Walls (PRI)	328/595	Assumed Contaminated	N/A - Demolish
215	247A	1st Floor - Floor (PRI)	627/776	Assumed Contaminated	N/A - Demolish
216	247A	1st Floor - Ceiling (PRI)	TBD	Assumed Contaminated	N/A - Demolish
217	247A	1st Floor - Equipment (PRI)	542/922	Assumed Contaminated	N/A - Demolish
218	247A	2nd Floor - Interior Walls (PRI)	214/407	Assumed Contaminated	N/A - Demolish
219	247A	2nd Floor - Ceiling	TBD	Assumed Contaminated	N/A - Demolish
220	247A	2nd Floor - Floor (PRI)	1915/2676	Contaminated	N/A - Demolish
221	247A	2nd Floor - Equipment (PRI)	1207/1900	Contaminated	N/A - Demolish

* Mean/95 % Confidence (βpm/100 cm^2)

Appendix A

Area	Bldg.	Location / Surface	Result*	Evaluation	Classification
222	247A	North Exterior Wall	-55/65	Clean	N/A - Demolish
223	247A	South Exterior Wall	88/170	Clean	N/A - Demolish
224	247A	247 A Roof	TBD	Assumed Contaminated	N/A - Demolish
225	247A	247 A Roof Equipment	82/147	Clean	N/A - Demolish
226	247B	1st Floor - Interior Walls (PRI)	131/204	Assumed Contaminated	N/A - Demolish
227	247B	1st Floor - Floor (PRI)	322/387	Assumed Contaminated	N/A - Demolish
228	247B	1st Floor - Ceiling (PRI)	TBD	Assumed Contaminated	N/A - Demolish
229	247B	1st Floor - Equipment (PRI)	264/363	Contaminated	N/A - Demolish
230	247B	2nd Floor - Interior Walls (PRI)	49/110	Assumed Contaminated	N/A - Demolish
231	247B	2nd Floor - Floor (PRI)	209/271	Assumed Contaminated	N/A - Demolish
232	247B	2nd Floor - Equipment (PRI)	244/340	Contaminated	N/A - Demolish
233	247B	2nd Floor - Ceiling (PRI)	374/457	Contaminated	N/A - Demolish
234	247B	North Exterior Wall	-57/102	Assumed Contaminated	N/A - Demolish
235	247B	South Exterior Wall	54/189	Assumed Contaminated	N/A - Demolish
236	247B	247 B Roof	TBD	Assumed Contaminated	N/A - Demolish
237	247B	247 B Roof Equipment	125/437	Assumed Contaminated	N/A - Demolish
238	246/7 N. Pad	N. Switch Gear Room - Interior Walls	TBD	Assumed Contaminated	N/A - Demolish
239	246/7 N. Pad	N. Switch Gear Room - Floor	TBD	Assumed Contaminated	N/A - Demolish
240	246/7 N. Pad	N. Switch Gear Room - Equipment	TBD	Assumed Contaminated	N/A - Demolish
241	246/7 N. Pad	N. Switch Gear Room - Ceiling	TBD	Assumed Contaminated	N/A - Demolish
242	246/7 N. Pad	N. Switch Gear Room - Exterior Walls	TBD	Assumed Contaminated	N/A - Demolish
243	246/7 N. Pad	N. Switch Gear Room - Roof	TBD	Assumed Contaminated	N/A - Demolish
244	248A	1st Floor - Interior Walls (PRI)	-285/-259	Clean	N/A - Demolish
245	248A	1st Floor - Floor (PRI)	210/242	Clean	N/A - Demolish
246	248A	1st Floor - Ceiling (PRI)	57/63	Clean	N/A - Demolish
247	248A	1st Floor - Equipment (PRI)	345/466	Clean	N/A - Demolish
248	248A	2nd Floor - Interior Walls (PRI)	-317/-297	Clean	N/A - Demolish
249	248A	2nd Floor - Floor (PRI)	119/150	Clean	N/A - Demolish
250	248A	2nd Floor - Ceiling (PRI)	119/204	Clean	N/A - Demolish
251	248A	2nd Floor - Equipment (PRI)	735/1354	Assume Contaminated	N/A - Demolish

* Mean/95 % Confidence (βpm/100 cm^2)

Appendix A

Area	Bldg.	Location / Surface	Result*	Evaluation	Classification
252	248B	1st Floor - Interior Walls (PRI)	-159/-106	Clean	N/A - Demolish
253	248B	1st Floor - Floor (PRI)	100/147	Clean	N/A - Demolish
254	248B	1st Floor - Ceiling	54/54	Clean	N/A - Demolish
255	248B	1st Floor - Equipment	193/210	Clean	N/A - Demolish
256	248B	2nd Floor - Interior Walls (PRI)	-132/-82	Clean	N/A - Demolish
257	248B	2nd Floor - Floor (PRI)	100/130	Clean	N/A - Demolish
258	248B	2nd Floor - Ceiling (PRI)	121/200	Clean	N/A - Demolish
259	248B	2nd Floor - Equipment (PRI)	202/238	Clean	N/A - Demolish
260	248E. Pad	Switch Gear Room - Interior Walls/Floor	TBD	Assumed Clean	N/A - Demolish
261	248E. Pad	Switch Gear Room - Equipment	TBD	Assumed Clean	N/A - Demolish
262	248E. Pad	Switch Gear Room - Exterior Walls/Roof	TBD	Assumed Clean	N/A - Demolish
263	248E. Pad	West Exterior Wall	131/244	Assumed Clean	N/A - Demolish
264	248E. Pad	South Exterior Wall	18/80	Clean	N/A - Demolish
265	248E. Pad	East Exterior Wall	-33/26	Clean	N/A - Demolish
266	248A&B	North Exterior Wall	24/116	Clean	N/A - Demolish
267	248A&B	Roof	TBD	Assumed Clean	N/A - Demolish
268	248A&B	North Wall Pipe Rack	TBD	Assumed Clean	N/A - Demolish
269	248A&B	Outside Equipment North Side	TBD	Assumed Clean	N/A - Demolish
270	248A&B	Outside Equipment South Side	TBD	Assumed Clean	N/A - Demolish
271	248A&B	248B/250 Pipe Line	TBD	Assumed Clean	N/A - Demolish
272	248A&B	Roof Equipment	27/61	Clean	N/A - Demolish
273	250	1st Floor - Rm 101 Closet: Area remodeled for entrance to lobby area. Walls are studded and drywalled on north side. South side masonry painted with epoxy coating.	17/103	Clean	Impacted Class 2
274	250	1st Floor - Rm 103/A/B Doors: Removed and area remodeled for new conference room. Walls are studded and drywalled.	53/197	Clean	Impacted Class 2
275	250	1st Floor - Rm 110 Desk Drawer: Removed and disposed of.	-38/-38	Clean	Impacted Category 2

* Mean/95 % Confidence (βpm/100 cm^2)

Appendix A

Area	Bldg.	Location / Surface	Result*	Evaluation	Classification
276	250	1st Floor - Rm 115 Doorways: Removed and disposed of. New walls installed. Area is studded and drywalled for labs and office cubicles on north side. New drop ceilings.	78/253	Clean	Impacted Class 2
277	250	1st Floor - Rm 119 Lockers: Men's locker room and rest room remains the same.	88/384	Clean	Impacted Class 2
278	250	1st Floor - Rm 120 Sink-original	626/1462	Contaminated	Impacted Category 1
279	250	1st Floor - Rm 122 Sink-original	1060/2173	Contaminated	Impacted Category 1
280	250	1st Floor - East Stairwell-original	1349/2185	Contaminated	Impacted Class 2
281	250	1st Floor - West Stairwell-original	716/1262	Contaminated	Impacted Class 2
282	250	1st Floor - CT Lead Well	-42/130	Clean	Impacted Category 2
283	250	1st Floor CT Lab Office - Interior Walls-mostly drywall covered post CT operations, block walls original and covered with one coat of epoxy paint since CT.	-138/-112	Clean	Impacted Class 2
284	250	1st Floor CT Lab Office - Floor: removed and replaced with linoleum post CT operations.	25/80	Clean	Impacted Class 2
285	250	1st Floor CT Lab Office - Ceiling: all new ceiling tiles post CT, drop ceiling structural components accessible.	-128/-12	Clean	Impacted Class 2
286	250	1st Floor CT Lab Hall - Interior Walls: drywall post CT except areas of exposed block. Exposed block painted with one coat epoxy paint.	26/50	Clean	Impacted Class 2
287	250	1st Floor CT Lab Hall - Floor: replaced with new tile post CT operations.	30/90	Clean	Impacted Class 2
288	250	1st Floor CT Lab Hall - Ceiling: all new ceiling tiles post CT, drop ceiling structural components accessible.	-143/91	Clean	Impacted Class 2
289	250	1st Floor CT Lab Count - Interior Walls: all drywall covered post CT operations.	-21/2	Clean	Impacted Class 2

* Mean/95 % Confidence (βpm/100 cm^2)

Appendix A

Area	Bldg.	Location / Surface	Result*	Evaluation	Classification
290	250	1st Floor CT Lab Count - Floor tile replaced post CT operations.	40/58	Clean	Impacted Class 2
291	250	1st Floor CT Lab Count - Ceiling: all new ceiling tiles post CT, drop ceiling structural components accessible.	69/109	Clean	Impacted Class 2
292	250	1st Floor CT Lab Count - Ceiling Drain Pipes-original	783/1506	Contaminated	Impacted Class 1
293	250	2nd Floor Rm 201 - Countertops/Sinks: original	800/1749	Contaminated	Impacted Category 2
294	250	2nd Floor Rm 201 - East Hood: original	131/383	Clean	Impacted Category 2
295	250	2nd Floor Rm 201 - NE Hood: Removed, disposed of and replaced.	24322	Contaminated	N/A - Demolish
296	250	2nd Floor Rm 201 - West Hood: original	425/1123	Contaminated	N/A - Demolish
297	250	2nd Floor Rm 201 - North Wall: Painted with epoxy coating.	166	Clean	Impacted Class 2
298	250	2nd Floor Rm 202 - Floors: New inlaid linoleum installed.	338/664	Remediated	Impacted Class 2
299	250	2nd Floor Rm 202 - Interior Walls: Repainted with epoxy coating.	49/96	Remediated	Impacted Class 2
300	250	2nd Floor Rm 202 - East Hood: Removed and new hood installed.	610/6157	Remediated	N/A
301	250	2nd Floor Rm 202 - West Hood: removed and new hood installed.	2661/6157	Remediated	N/A
302	250	2nd Floor Rm 202 - Countertops: New countertops, sinks, cabinets above and below installed, new inlaid linoleum flooring and ceiling tile installed.	44/202	Remediated	N/A
303	250	2nd Floor Rm 205 - East Wall: Repainted with epoxy coating.	78/875	Assumed Clean	Impacted Class 2
304	250	2nd Floor Rm 207 - Sinks: Removed and disposed of.	698/1778	Contaminated	N/A - Demolish

* Mean/95 % Confidence (βpm/100 cm^2)

Appendix A

Area	Bldg.	Location / Surface	Result*	Evaluation	Classification
305	250	2nd Floor Rm 207 - Staging Area: floor under sink remediated post CT operations	43/118	Clean	Impacted Class 2
306	250	2nd Floor Rm 209 - West Hood: original	1994/6464	Contaminated	N/A - Demolish
307	250	2nd Floor Rm 209 - East Hood: original	760/3452	Contaminated	N/A - Demolish
308	250	2nd Floor Rm 209 - Countertops: original	-117/3	Clean	Impacted Category 2
309	250	2nd Floor Rm 209 - Drains: original	-2/468	Clean	Impacted Category 2
310	250	2nd Floor Rm 209 - Floors: original	11	Clean	Impacted Class 2
311	250	2nd Floor Rm 210 - Floors: original	-59/-16	Clean	Impacted Class 2
312	250	2nd Floor Rm 210 - North and West Walls: original	1/2	Clean	Impacted Class 2
313	250	2nd Floor Rm 210 - Mechanical Room Equipment: Vacuum hold tank, air compressor, air handler, and chiller unit - original.	447/945	Assumed Clean	Impacted Category 2
314	250	2nd Floor Rm 211 - West Hood: original	109/158	Clean	Impacted Category 2
315	250	2nd Floor Rm 211 - Countertops: original	-71/9	Clean	Impacted Category 2
316	250	2nd Floor Rm 211 - Drain: original	-92	Clean	Impacted Category 2
317	250	2nd Floor Rm 211 - Cabinets: original	69/269	Clean	Impacted Category 2
318	250	East Exterior Wall	52/88	Clean	Impacted Class 3
319	250	South Exterior Wall	TBD	Assumed Clean	Impacted Class 3
320	250	Roof Decking: Roof replaced post CT operations, some decking repairs and replacement in process area.	52/62	Clean	Impacted Class 3
321	250	East Pad	TBD	Assumed Contaminated	Impacted Class 2 (Phase II)
322	260	Constructed in 2000	NA	Clean	Non Impacted
323	503	Constructed in 1980. Open air manufacturing building. Control room 2nd level is enclosed. Building equipment dismantled and disposed of 1994-1995. Steel structure with control room, MCC room and rest room remain.	NA	Clean	Non Impacted

* Mean/95 % Confidence (βpm/100 cm^2)

Appendix A

Area	Bldg.	Location / Surface	Result*	Evaluation	Classification
324	510	Constructed 1992 - 1993 after CT operations ceased.	NA	Clean	Non Impacted
325	501	Roof: FUSRAP impacted roof.	NA	FUSRAP	Non CT Impacted

1 Data will be collected during decommissioning.
2 The results include Potassium 40.

* Mean/95 % Confidence (βpm/100 cm^2)

APPENDIX B
EXAMPLE SURFACE MAPS

Appendix B was removed and is no longer relevant.

Mallinckrodt Inc.

Phase 1 Plan
For C-T Decommissioning

NRC Docket: 40-06563
NRC License : STB-401

January 9, 2002

APPENDIX B

EXAMPLE SURFACE MAPS

Appendix B was removed and is no longer relevant.

APPENDIX C

DERIVATION OF NUCLIDE LIMITS FOR BUILDING SURFACES

Mallinckrodt Inc.

Phase 1 Plan
For C-T Decommissioning

NRC Docket: 40-06563
NRC License : STB-401

January 9, 2002

APPENDIX C

DERIVATION OF NUCLIDE LIMITS FOR BUILDING SURFACES

1. OVERVIEW

The purpose of this appendix is to describe the derivation of nuclide limits for building surfaces. Under NRC regulations, 10 CFR 20, Subpart E, limits for radioactive material remaining after license termination are expressed in terms of radiation dose (effective dose equivalent) that might reasonably be anticipated from exposure to the residual radionuclides. However, the anticipated dose cannot be measured directly. Therefore, to implement the regulation, the dose limit must be translated into a corresponding limit of some related quantity that can be measured directly. In this decommissioning plan, that is done in two major steps. First, the dose limit is translated into nuclide limits in terms of nuclear transformations per unit time per unit area of building surface. That translation is described in this appendix. The second major step is the use of nuclide limits, derived as described in this appendix, in conjunction with building-specific nuclide mixes to derive limits expressed in terms of a composite limit that may be compared with gross radiation measurements. The derivation in the second major step is described in Appendix D.

One objective of this decommissioning plan is that buildings remaining on the site after license termination may be used without restrictions on use. The dose limit specified in 10 CFR Part 20, Subpart E, for license termination without restrictions on use is 25 millirem per year. Therefore, the dose basis for the nuclide limits derived in this appendix is 25 millirem per year.

Derivation of the nuclide limits described in this appendix occurs in a series of steps. The first step is the identification of an exposure scenario to serve as the basis for the nuclide limits to be derived. The exposure scenario specification describes how persons might be exposed to residual radioactive materials and includes values for parameters important in determining dose, such as exposure times. The ideal basis exposure scenario would be a reasonably anticipated exposure scenario that would result in the maximum dose from exposure to residual radioactive material after termination of the license. The exposure scenario definition process is described in Section 2.0 of this appendix.

Radiation dose assessment models are usually used to calculate radiation doses from input nuclide concentrations in some medium. However, because radiation dose from a nuclide scales directly to the nuclide concentration, other factors remaining constant, results can be used to compute the nuclide concentration that corresponds to any particular dose limit. The radiation dose assessment model input and output and the computation of nuclide limits from the output are described in Section 3.0 of this appendix.

Nuclide concentration limits derived in Section 3.0 of this appendix account for radiation dose from progeny nuclides, but are expressed in terms of the concentration of the first nuclide in the decay chain. The product of this work is a table of nuclide limits that serve as input to the second major step in the derivation of a composite areal density limit that may be compared with measurements.

2. EXPOSURE SCENARIO

This section specifies the exposure scenario to serve as the basis for derivation of nuclide limits on building surfaces. The exposure scenario specification describes how persons might be exposed to residual radioactive materials and would include values for parameters important in determining dose, such as exposure times.

As noted in the overview, the ideal exposure scenario would be a reasonably anticipated exposure scenario that would result in the maximum dose from exposure to residual radioactive material after termination of the license. For purposes of this decommissioning plan, the scenario assumed for purposes of deriving nuclide limits for building surfaces is the based on the building occupancy scenario from NUREG\CR-5512[1]. As explained in NUREG\CR-5512, this scenario was developed for generic pathway modeling. Consequently, the scenario specification incorporates some exposure assumptions that tend to be somewhat unrealistic and tend to cause the dose associated with this scenario to be substantially higher than might be expected for more likely scenarios specifically tailored to the C-T situation after license termination. A large fraction of the exposure scenarios that might be reasonably anticipated after termination of the license for the C-T areas would lead to substantially lower doses. It is considered that exposure scenarios that would lead to substantially higher radiation dose would be highly unlikely.

The building occupancy scenario incorporates conservative assumptions regarding potential exposure characteristics. The scenario involves chronic exposure to an individual for a full work year in a building that is part of a typical commercial facility. With regard to use as part of a commercial facility, the scenario fits the C-T situation well. The C-T buildings to remain after license termination have been used only for commercial purposes and most likely will continue to be for the remainder of their useful lives. The building occupancy scenario exposure pathways considered sufficiently important for analysis include external exposure to penetrating radiation from surface sources, inhalation of radionuclides eroded from surfaces and suspended in room air, inhalation of radionuclides resuspended from surfaces, and inadvertent ingestion of radionuclides directly and indirectly from surfaces. The sources of exposure were assumed to be on floor and walls. Other exposure pathways may exist, but are considered to be much less important in terms of potential dose (NUREG\CR-5512, Section 3.1.2).

3. MATHEMATICAL MODEL

As noted in the overview, radiation dose assessment models are usually used to calculate radiation doses from input nuclide concentrations in some medium. However, because

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[1] Kennedy, W.E., and Strenge, D.L., "Residual Radioactive Contamination from Decommissioning: Technical Basis for Translating Contamination Levels to Annual Total Effective Dose Equivalent," NUREG\CR-5512, vol.1, Pacific Northwest Laboratory, Richland, Washington, September, 1992.

radiation dose from a nuclide scales directly to the nuclide concentration, other factors remaining constant, results can be used to compute the nuclide concentration that corresponds to any particular dose limit. The RESRAD-BUILD (Version 3.0) computer code[2], developed by ANL for the United States Department of Energy, and commonly used in radiation dose assessments of this type, was used for this analysis. The radiation dose assessment model input and output and the computation of nuclide limits from the output are described in this section.

The radioactive material source is modeled as a planar, area source on the interior surfaces of a building. This approach reflects C-T building conditions—concrete floors and brick walls. Common experience in cleaning such somewhat porous surfaces shows that scabbling away as much as 1/8 inch of the surficial material typically removes "surficial" radioactive contamination.

The following are bases of selected values of parameters in RESRAD-BUILD that are used to derive $DCGL_W$ for Mallinckrodt building surfaces.

3.1 Areal Source

The radioactive sources entered into RESRAD-BUILD are the thorium series, the uranium series, and the actinium (U^{235}) series. Each decay series is assumed to be in radioactive equilibrium.

3.2 Inhalation Exposure Pathway

To evaluate dose from inhalation, it was assumed that the worker inhaled eroded, radionuclide-containing surface material suspended in air. Erosion of surface material can be expected to be minimal for likely operations in the C-T buildings. Eroded material would typically remain deposited on the surface or would be removed in cleaning. For that reason, it was assumed that only a fraction, 0.1, of the eroded material is suspended in air and mixed uniformly in the room ventilation flow. The fraction of contaminated non-combustibles made airborne and the respirable fraction of airborne dust are estimated to be 0.1 and 0.7, respectively.[3] Their product, 0.1 x 0.7 = 0.07, was entered into RESRAD-BUILD as the air release fraction of respirable dust, consistent with the recommended default value.

Deposition and resuspension of originally suspended dust was also modeled. The ANL-recommended maximum default values[4, 5] of deposition velocity = 2.7×10^{-3} m/s and

[2] Yu, 1994. Yu, C., LePoire, D.J., Jones, L.G., and Chen, S.Y., "RESRAD-BUILD: A Computer Model for Analyzing the Radiological Doses Resulting from the Remediation and Occupancy of Buildings Contaminated with Radioactive Material," ANL/EAD/LD-3, Argonne National Laboratory, Argonne, Illinois, November, 1994

[3] Biwer, B.M., *et al.*, "Parameter Distributions for Use in RESRAD and RESRAD-BUILD Computer Codes." in Yu, C., *Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes.* NUREG/CR-6697, Attachment C, p. 8-14 & 8-15. Dec. 2000.

[4] *Op. cit.,* Attach. C, pp. 7-18 thru 7-22.

[5] *Op. cit.*, Attach C, pp. 7-5 thru 7-7.

resuspension factor = 1.4 x 10^{-5}/s were entered into RESRAD-BUILD. The computed radiological dose was not very sensitive to variation of either deposition velocity or resuspension factor.

3.3 Direct Ingestion Rate of Originally Removable Material

Direct Ingestion refers to the incidental ingestion of contaminated material directly from the source.[6] Recently, Biwer, *et. al.*, recommended the direct ingestion rate of radioactive source material be set = 0 in RESRAD-BUILD.[7] They stated:

> The direct ingestion rate is included in the RESRAD-BUILD code for unlikely events when a receptor could directly ingest source material. Such a receptor could be conducting a maintenance or renovation activity that involved physical contact with the source. The direct ingestion rate is normally set to 0 for most calculations.[8]

It is ANL's opinion that direct ingestion of source material is unlikely, is normally estimated to be zero. In recognition that contamination in buildings remaining for final status survey is likely to be well below the DCGL since the CT process buildings will be demolished, direct ingestion rate will be set to zero in RESRAD-BUILD derivation of the DCGL for remaining building surfaces that are subject to final status survey.

3.4 Indirect Ingestion Rate of Settled Dust

Biwer, *et. al.*, recently reviewed data on ingestion rate of soil,[9] including the data also reviewed by Beyler, *et. al.* With an interest in adult ingestion of dust deposited in a building, so-called *indirect ingestion*, Biwer, *et. al.*, recommend 0.5 mg/da indirect ingestion rate as a default value. They interpret this to be equivalent to ingesting dust deposited on 1.1 x 10^{-4} m^2/hr. This is practically the same as Sandia's estimate.[10] In view of ANL's recommendation, a dust ingestion rate from 1.1 x 10^{-4} m^2/hr contacted is entered into RESRAD-BUILD to simulate an indirect ingestion rate of settled dust for the purpose of deriving CT buildings areal DCGL.

3.5 External Irradiation Exposure Pathway

To evaluate dose from external radiation, the exposure at a point in the center of the room at a height of 1 meter above the floor was calculated. Exposure at that location is likely to be the maximum a worker would receive, based on geometry considerations. An exposure time of 2,000 hours per year was assumed.

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[6] Biwer, B.M., *et al.*, NUREG/CR-6697, Attachment C. p. 5-26.Dec. 2000.

[7] *Ibid.*

[8] Biwer, B.M., *et al.*§5.7, p. 5-26. Dec. 2000.

[9] Biwer, *et. al.*, §5.8, p. 5-27, Dec. 2000.

[10] Beyler, W.E., *et. al.*, Jan. 1998.

3.6 Occupancy Time

The occupation time is assumed to be the full working year, 2,000 hours per year, equivalent to 50 weeks at 40 hours per week.

3.7 Breathing Rate

A breathing rate of 1.4 m^3/h (33.6 m^3/d) was assumed for the worker. This value is typical for light activity[11] and is considered appropriate for long-term average exposure in the NUREG/CR-5512 building occupancy exposure scenario.

3.8 Room and Ventilation

The typical C-T building is modeled as a single room 36 m^2 in area and 2.8 m high. The product of a corresponding wall length = 6 m and room height = 2.8 m corresponds to a wall area = 16.8 m^2. 36 m^2 is the default floor area of a room for deterministic calculations in RESRAD-BUILD[12] The $DCGL_w$ was derived on the basis of a floor and 4 walls whose total area = 103 m^2.

The ventilation rate is assumed to be 10 volume changes per hour, typical for buildings currently in use at the site. Mallinckrodt has historically designed chemical manufacturing areas for about 10 air changes per hour, with the capability of increasing the ventilation rate in the event of a chemical spill. That would include, for example, Buildings 200W, 222S, 235, and 250. That is consistent with ASHRAE recommendations of 8 to 12 air changes per hour for a variety of building uses. Office areas, not expected to have been contaminated, will be ventilated in the range of 4 to 10 air changes per hour according to the ASHRAE Handbook.[13]

3.9 Air Fraction.

The fraction of loose material, eroded from the surface, that becomes airborne in the respirable particulate range is called the *air release fraction*, or *fraction released to air*, or the *air fraction*. For contaminated non-combustibles, Biwer, *et. al.,*[14] estimate the bounding value of the airborne release fraction and respirable fraction to be 0.1 and 0.07 respectively. Their product, 0.1 x 0.7 = 0.07, is entered into RESRAD-BUILD as the air release fraction of respirable dust.

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[11] Beyeler, W.A., Hareland, W.A., Duran, F.A., Kalinina, E., Gallegos, D.P., and Davis, P.A., "Review of Parameter data for the NUREG/CR-5512 Building Occupancy Scenario and Probability Distributions for the DandD Parameter Analysis, Draft Letter Report, Sandia National Laboratories, 1998.

[12] Biwer, B.M., *et.al.,p. 7-29.*

[13] Amer. Soc. Heat., Refrig., & Air-cond Engr. *1999 ASHRAE Handbook*, Heating, Ventilating, and Air-Conditioning Applications. Inch-pound ed. Table 1. 1999.

[14] Biwer, B.M., et.al., p. 8-14 & 8-15.

3.10 Removable Fraction

The combination of a *removable fraction* and the *time for source removal, or source lifetime,* estimate an erosion rate that is constant until the source is depleted. Some fraction of areal U series and Th series residue is estimated to be removed at a linear rate between time zero and the *time for source removal*. Biwer, *et. al.*,[15] recommend that the most likely value of the removable fraction of regulated radioactive residue on a surface be estimated as:

$$\frac{\text{maximum acceptable removable areal density}}{\text{maximum acceptable average areal density}} = 0.2$$

This recommended default value is entered into RESRAD-BUILD to derive DCGL for radioactive contamination on a building surface.

3.11 Time During Which Surface Contamination is Removed

The *removable fraction* is used together with the *time during which surface contamination is removed* to estimate the rate of original erosion of surface contamination into dust that is suspendable into indoor air. The most likely value of time during which surface contamination is removed, recommended by Biwer, *et. al.*,[16] = 10,000 days = 27.4 yr, is derived from the bounding value of an estimated rate of aerodynamic entrainment of dust from among debris[17, 18] Biwer, *et. al.*'s estimate is derived from an estimated release rate from constrained solid powder[19,20] This value, 10,000 days, is entered into RESRAD-BUILD to estimate DCGL for building surfaces.

The combination of *removable fraction* = 0.2 and *time to remove* = 10,000 days are thus entered into RESRAD-BUILD to estimate the removal rate from building surfaces.

Removable Fraction	Time to Remove (da)	Time to Remove (yr)
0.2	10,000	27.4

[15] Biwer, B.M., et.al., p. 8-6.

[16] Biwer, B.M., et.al., p. 8-20 & 8-21.

[17] USDOE. *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities.* 1, §5.3, pp. 5-21 thru 5-29.

[18] ANS. *Airborne Release Fractions at Non-reactor Nuclear Facilities.* ANSI std. ANSI/ANS-5.10-1998. p. 14. May 1998.

[19] *Ibid.*

[20] *Ibid.*

3.12 Other Parameters

Values of some other parameters of significance are:

Parameter	Quantity
Source	Natural uranium + progeny in radioactive equilibrium Natural thorium + progeny in radioactive equilibrium
Source geometry type	Area
Receptor indoors	0.23 of 365 da/yr
Receptor inhales	33.6 m^3/da
Building room floor area	36 m^2
Building room wall height	2.8 m
Building room air exchange rate	10 /hr

4. COMPUTER MODEL RESULTS

In the computer analysis, a separate computer run was performed for each major nuclide: U^{238}, U^{234}, Th^{230}, Ra^{226}, Pb^{210}, Th^{232}, Ra^{228}, and Th^{228}. The RESRAD-BUILD code included dose contributions from short-lived daughters of each of these nuclides.

The input and output for the RESRAD-BUILD runs used in this analysis are listed in Attachment 7. The RESRAD-BUILD output for each radionuclide can be interpreted as a dose factor (mrem/y per pCi/m^2), which in turn may be interpreted as a maximum acceptable areal density of the radionuclide on a surface, also called the $DCGL_W$, corresponding to a potential radiological dose equivalent of 25 mrem/yr.

The input and output for the RESRAD-BUILD runs used in this analysis are listed in Attachment C1. The RESRAD-BUILD output for each radionuclide can be interpreted as a dose factor (mrem/y per pCi/m^2), which in turn may be interpreted as a maximum acceptable areal density of the radionuclide on a surface, also called the $DCGL_W$, corresponding to a potential radiological dose equivalent of 25 mrem/yr.

Table C1. Uranium Series and Thorium Series Limits on Building Surfaces

Radionuclide	Dose Factor	Areal Density Equal to 25 mrem/yr	
	mrem/yr per pCi/m^2	pCi/100 cm^2	dpm/100 cm^2
U-238	2.30E-6	1.09E+05	2.41E+05
U-235 +DI[c]	5.89E-5	4.24E+03	9.42E+03
U-234	2.24E-6	1.12E+05	2.48E+05
Th-230	5.58E-6	4.48E+04	9.95E+04
Ra-226	1.80E-5	1.39E+04	3.08E+04
Pb-210	4.53E-7	5.25E+05	1.23E+06
Th-232	2.84E-5	8.80E+03	1.95E+04
Ra-228	1.26E-5	1.98E+04	4.40E+04
Th-228	1.72E-5	1.45E+04	3.23E+04
U-238 +DI[b]	2.98E-5	8.39E+03	1.86E+04
Th-232 +DI[a]	5.82E-5	4.30E+03	9.54E+03

[a] Th-232 +DI is the limit for Th-232 in the situation in which all progeny nuclides are present in equilibrium concentration (*i.e.*, concentration of each equal to the Th-232 concentration). Because Th-232 progeny grows in to equilibrium within about 30 years, and because the C-T facilities have existed for nearly that long, Th-232 progeny can be expected to be near equilibrium.

[b] U-238 +DI is the limit for U-238 in the situation in which all progeny nuclides are present in equilibrium and the U^{235} series is present in equilibrium as in natural uranium

[c] Radioactivity ratio of U^{235} -to- U^{238} = 0.0455 in natural uranium.

Since U-238+DI and Th-232+DI represent the uranium and thorium series in radioactive equilibrium, each of their maximum dose factors is derived during the 0 to1 year time increment. The contribution from each key, long-lived radionuclide, with short-lived progeny grown-in is also tabulated in Table C1. Short-lived progeny of key radionuclides in the thorium series grow to equilibrium within about one year. The maximum dose factor occurs during the first 10 years for some key radionuclides in the uranium series that have progeny whose half-life is greater than one year. Table C1 lists the maximum dose factor for those key uranium series radionuclides.

$$\text{Area Factor} = \frac{\text{composite dose factor for survey unit area}}{\text{composite dose factor for local area of contamination}}$$

Figure C1 is the area factor as a function of a localized area of radioactive contamination consisting of U series to Th series in a 2 -to-1 parent ratio. Mallinckrodt intends to specify that the maximum tolerable area factor employed shall be no greater than 10.

The maximum tolerable areal density of residual radioactive contamination, above background, within a small area of elevated radioactivity is derived by the relation

$$DCGL_{EMC} = \text{Area Factor} \times DCGL_W$$

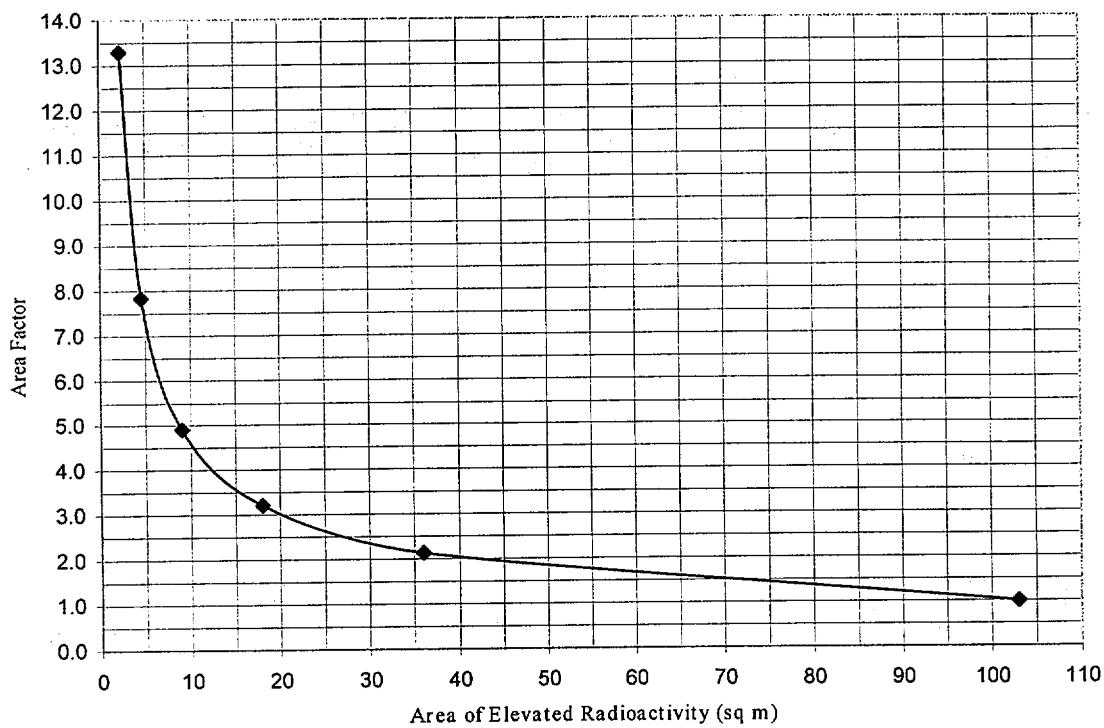

Figure C1. Area Factor for Elevated Measurement Criterion

Removed for Proprietary Reasons

APPENDIX D

METHOD OF INTERPRETING SURFACE RADIOACTIVITY LIMIT AND GROSS BETA MEASUREMENT IN COMPARABLE UNITS

Mallinckrodt Inc.

Phase 1 Plan
For C-T Decommissioning

NRC Docket: 40-06563
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January 9, 2002

APPENDIX E

DERIVATION OF CONCENTRATION LIMIT FOR UNRESTRICTED RELEASEOF BUILDING RUBBLE

Appendix E was removed and is no longer relevant.

Mallinckrodt Inc.

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For C-T Decommissioning

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January 9, 2002

APPENDIX F

PROJECT SCHEDULE

Mallinckrodt Inc.

Phase 1 Plan
For C-T Decommissioning

NRC Docket: 40-06563
NRC License : STB-401

January 9, 2002

APPENDIX F
PROJECT SCHEDULE

1. OVERVIEW

This appendix contains the proposed schedule for the C-T Decommissioning Project. The schedule, Figure F-1 is a Microsoft Project version of the expected general task schedule for the Phase I project. It shows a project completion 23 months after Phase 1 Plan approval.

The following assumptions and constraints were used in generating the schedule:

- Job site mobilization should commence 3 months after approval, unless there are extenuating circumstances.
- The schedule includes contractor and Mallinckrodt activities.
- The schedule is based on a 5 day work week, 8 hours per day.
- Labor overtime is not scheduled (considered contingency).
- 11 Holidays per year are factored into the schedule.
- Parallel activities can be performed in all the affected areas.
- Equipment will be removed as required before decontamination or demolition activities.
- Buildings 246, 246A, 247, 247A, and 248 will be demolished together.

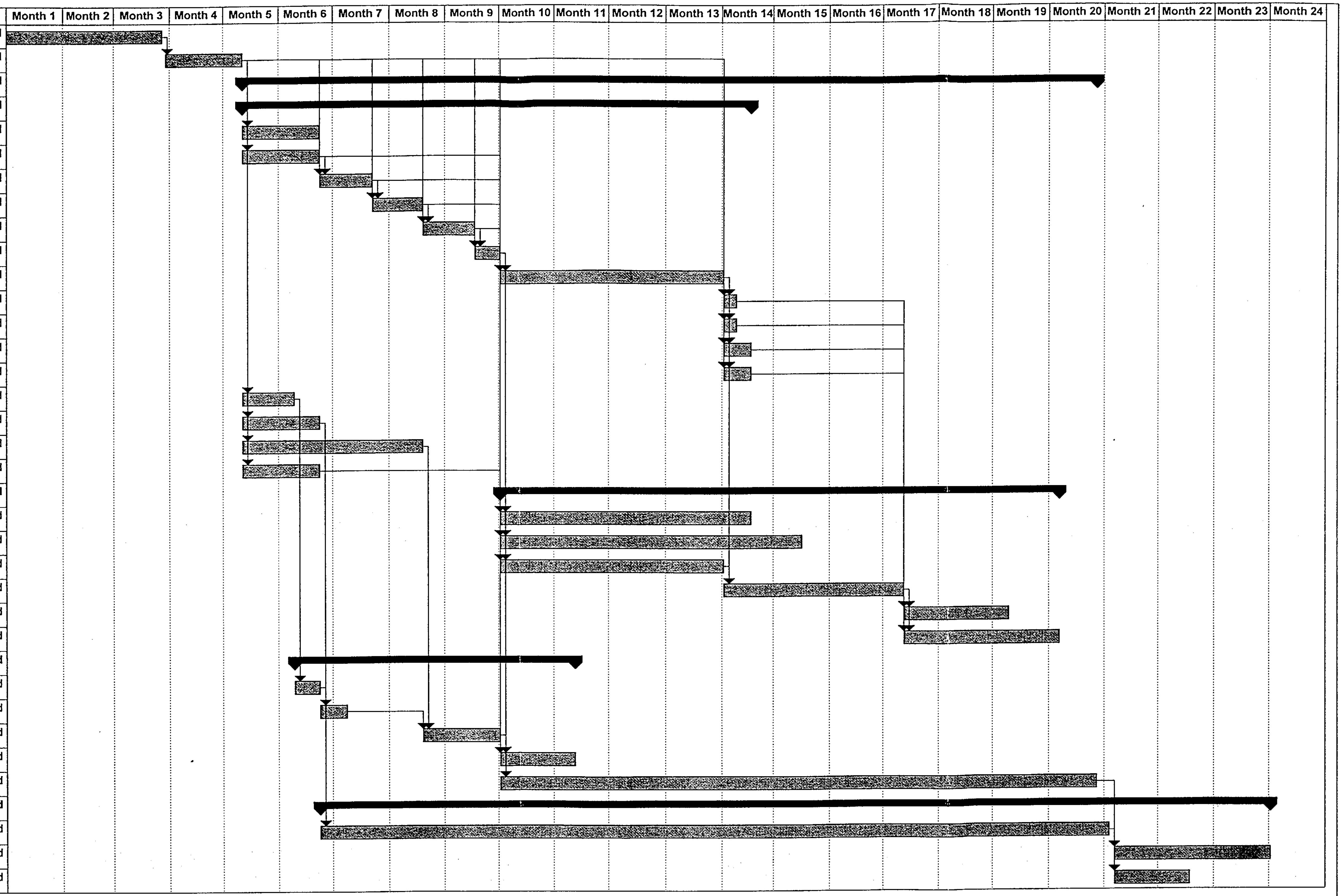
The remediation contractor's project manager will create the actual schedule and will maintain the detailed schedule, control resource changes, and coordinate with Mallinckrodt to meet all reporting requirements. The sequence of site activities may change from the attached schedule to minimize on-site time and interference with ongoing production activities in nearby buildings.

Building groupings have been made for ease of schedule presentation. The buildings have been grouped according to their decommissioning paths, i.e. buildings with similar decommissioning activities have been grouped together.

- Building Groups A and B consist of buildings scheduled to be demolished (razed) and include Bldgs 213, 214, 238, 246, 247, and 248.
- Building Group C consists of Buildings 90 and 91. Some equipment in these buildings needs to be surveyed prior to final release.
- Building Group D consists of Buildings 62 and 240. These buildings will be surveyed for final release only.
- Building Group E consists of Buildings 235 and 236. Some surface decontamination, equipment removal, and roof replacement and surveys are necessary for release.
- Building Group F consists of Buildings 101, CT Incinerator, 200E, 200W, 222, and 223. These roofs will have the equipment surveyed, decontaminated if needed and then the roofs will be subjected to a final survey.

In the event mixed waste is identified during remediation activities, Mallinckrodt will assess the effect on the schedule and will notify the NRC.

ID	Task Name	Duration
1	**Site Preparation**	60d
2	**Contractor Mobilization**	30d
3	**Site Activities**	**325d**
	Equipment Removal	**190d**
5	Non Buildings	30d
6	Equipment: Building 248	30d
7	Equipment: Building 247	20d
8	Equipment: Building 247A	20d
9	Equipment: Building 246A	20d
10	Equipment: Building 246	10d
11	Equipment: Building 238	80d
12	Equipment: Building 214	5d
13	Equipment: Building 213B	5d
14	Equipment: Building 213	10d
15	Equipment: Building 213A	10d
16	Equipment: Group C Buildings	20d
17	Equipment: Group D Buildings	30d
18	Equipment: Group E Buildings	70d
19	Equipment: Group F Buildings	30d
20	**Demolition**	**210d**
21	Demo: Building 248	90d
	Demo: Building 247	110d
23	Demo: Building 246	80d
24	Demo: Building 238	70d
25	Demo: Building 214	40d
26	Demo: Building 213	60d
27	**Decontamination**	**110d**
28	Decon: Group C Buildings	10d
29	Decon: Group D Buildings	10d
30	Decon: Group E Buildings	30d
31	Decon: Group F Buildings	30d
32	**Waste Shipping**	225d
33	**Final Survey**	**360d**
34	Final Surveys	300d
35	Final Survey Report	60d
36	**Demobilization**	30d



Project: C-T D&D Phase I Schedule 1
Date: Wed 11/19/97

F-1

Task | Progress | Milestone | Summary | Rolled Up Task | Rolled Up Milestone | Rolled Up Progress

APPENDIX G

ACRONYM LIST

Mallinckrodt Inc.

Phase 1 Plan
For C-T Decommissioning

NRC Docket: 40-06563
NRC License : STB-401

January 9, 2002

ACRONYM LIST

Acronym	Definition
α	alpha
ACM	asbestos containing material
AEC	Atomic Energy Commission
ALARA	As Low As Reasonably Achievable
β	beta
BKG	Background
BZ	Breathing zone
CERCLA	
CFR	Code of Federal Regulations
Ci	Curie(s)
Cpm	counts per minute
CPR	Cardio Pulmonary Resuscitation
C-T	Columbium-Tantalum
Δ	delta
DCGL	Derived Concentration Guideline Level
DEA	Drug Enforcement Agency
DG	draft regulatory guide
DOE	Department of Energy
DP	Decommissioning Plan
D Plan	Decommissioning Plan
dpm	disintegrations per minute
DQO	Data Quality Objectives
EPA	Environmental Protection Agency
ERDA	Energy Research and Development Agency
ES&H	Environmental Safety and Health
FC	Fuel Cycle and Material Safety Division
FFA	Federal Facilities Agreement
FI	Field Instruction
FSSR	Final Status Survey Report
FUSRAP	Formerly Utilized Site Remedial Action Program
GSA	General Services Administration
HAZWOPER	Hazardous Wasted Operations and Emergency Response
K	potassium
LBGR	lower bound of gray region
LLRW	low level radioactive waste
LLD	lower limit of detection
MAAD	maximum acceptable areal density
MAAAD	maximum acceptable average areal density
MARSSIM	Multi-agency radiological site survey and investigation manual
MCW	Mallinckrodt Chemical Works

ACRONYM LIST

MDA	minimum detectable activity
MDAD	minimum detectable areal density
MDC	minimum detectable concentration
MDCR	minimum detectable count rate
MDNR	Missouri Department of Natural Resources
MED	Manhattan Engineer District
MIBK	methyl isobutylketone
Mrem	millirem
MSD	St. Louis Metropolitan Sewer District
NIST	National Institute of Science and Technology
NRC	Nuclear Regulatory Commission
NUREG	Nuclear Regulations
ORNL	Oak Ridge National Laboratory
OSHA	Occupational Safety and Health Administration
p	pico
pCi	picocurie(s)
P	probability
PM	Project Manager
PRI	Preliminary Radiological Investigation
QA	Quality Assurance
QAPP	Quality Assurance Project Plan
QC	quality control
QIP	Quality Implementing Procedure
R	Roentgen
Ra	Radium
RCRA	Resource, Conservation, and Recovery Act
RESRAD	USDOE residual radioactivity simulation computer program
ROD	Record of Decision
RWP	Radiation Work Permit
RSO	Radiation Safety Officer
σ	sigma
SAIC	Science Applications International Corporation
SRSO	Site Radiation Safety Officer
SWP	Safety Work Permit
TBD	to be determined
Th	thorium
TLD	thermoluminescent dosimeter
THR	threshold
TSCA	Toxic Substances Control Act
U	uranium
UO_2	uranium dioxide
URO	unreacted ore
USACE	U. S. Army Corps of Engineers

ACRONYM LIST

WRS	Wilcoxon Rank Sum test

APPENDIX H

ASSESSMENT OF MAXIMUM ACCEPTABLE AREAL DENSITY BY ALPHA RADIATION MEASUREMENT

Mallinckrodt Inc.

Phase 1 Plan
For C-T Decommissioning

NRC Docket: 40-06563
NRC License : STB-401

January 9, 2002

APPENDIX H

ASSESSMENT OF MAXIMUM ACCEPTABLE AREAL DENSITY BY ALPHA RADIATION MEASUREMENT

1. APPROACH

To implement radiological criteria for decommissioning buildings that will remain, limits on acceptable areal density of radionuclides are derived by mathematical modeling. For equipment, NRC *Guidelines*[1] specify areal density limits on alpha radiation emitted by uranium and thorium decay series on a surface. Both need to be expressed as a single measurable quantity that can verified by an instrument that can measure gross alpha radiation emitted from a surface. A method to express a single-valued, acceptable average areal alpha radiation density and a method to interpret gross alpha radiation measurements in the same units are described herein.

Appendix H was originally derived for interpretation of alpha measurements. Surfaces, absent any coating covering the α-emitter, may be surveyed by alpha-ray detection under conditions specified herein. Recognizing the greater range of beta rays than of alpha rays, Mallinckrodt intends to rely mainly on survey by beta-ray detection. An equivalent beta limit has been derived and is included herein.

2. RADIONUCLIDES

The radioactive materials of interest on the C-T project site are the uranium series and the thorium series. The key radionuclides are the long-lived radionuclides: U^{238}, Th^{230}, and Ra^{226} in the uranium series and Th^{232}, Ra^{228}, and Th^{228} in the thorium series. The thorium series will have grown near to radioactive equilibrium after any disruption that may have occurred during chemical processing, and thus may be assumed to be in equilibrium.

Relative radioactivity fractions of key uranium series radionuclides were measured in 74 material samples taken from surfaces in and on buildings. The relative activity fractions are similar enough for both the parent and progeny nuclides that the uranium series and thorium series may be treated as being in equilibrium. When deriving a composite limit, tolerance of some disequilibrium in a decay series is consistent with NRC guidance.[2] Consistent with this, the U series to Th series ratio in each sample may be represented by the average of $U^{238} + Th^{230} + Ra^{226} \div$ the average of $Th^{232} + Ra^{228} + Th^{228}$. Absent aberrant ratios in single samples attributable to near zero or net negative radioactivity reported, derived limits are consistent with an average U series -to- Th series ratio of 2 -to- 1 among the remaining 69 samples.

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1 USNRC. "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Restricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material," in FC 83-23.

2 NRC. "Health Physics Surveys in Uranium Mills," Reg. Guide 8.30, §§ 1.5, 1.6 & 1.7.

3. BASIC LIMITS

3.1 Equipment. Equipment release criterion is based on NRC *Guidelines*[3] and referenced in Mallinckrodt's radioactive material license. The **maximum** acceptable areal density on equipment is three times the maximum acceptable **average** areal density (MAAAD) and the removable limit is 1/5 the MAAAD, following the convention of the *Guidelines*.

The composite $MAAAD_\alpha$ has been computed for 69 of the 74 samples; the average of the results is 2430 αdpm/100 cm^2. That agrees with the $MAAAD_\alpha$ derived by computing the average U series to Th series ratio among all samples, 2 U -to- 1 Th, and then deriving a composite limit; the result is

$$\text{composite MAAAD}\,\alpha = \frac{2\times 8 + 1\times 6}{\frac{2\times 8}{5000} + \frac{1\times 6}{1000}} = 2400\ \alpha\text{dpm}\ /100\ \text{cm}^2$$

A higher U -to- Th series ratio would yield a higher $MAAAD_\alpha$ than 2400 αdpm/100 cm^2.

A composite $MAAAD_\alpha$ = 2400 αdpm/100 cm^2 would be reasonable to apply to surfaces of all equipment surveyed for unrestricted release.

3.2 Buildings. Release criteria for surfaces of buildings subject to this plan and that will remain in plant area 5 are derived on the basis of 25 mrem/yr dose limit for unrestricted industrial use. The computer code, RESRAD-BUILD, is used to derive average areal radionuclide density ($DCGL_W$) that potentially delivers 25 mrem/yr. Further information on this derivation is in Appendix C. On building surfaces, elevated measurements areal density limit may be derived as the product of the $DCGL_W$ and an area factor,[4] derived as described in the MARSSIM.[5]

Alternatively, in the event a building is to be surveyed for release subject to the NRC "Guidelines" (license conditions 16), then the basic limits in the previous paragraph, 3.1 Equipment, apply.

4. DERIVATION OF MAXIMUM ACCEPTABLE AVERAGE AREAL ALPHA DENSITY

The method described herein is intended to accommodate the uranium series and thorium series ratio observed on site and translate the residual radioactivity limit to a single-valued alpha radiation limit. This section describes the methodology used to derive an alpha limit from the *Guidelines* or from derived limits corresponding to 25 mrem/yr. This derived surface release limit is the maximum acceptable **average** areal density (MAAAD on equipment or $DCGL_W$ on a building).

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3 *Ibidem.*

4 An area factor is the magnitude by which the areal density in a small area of elevated radioactivity can exceed the MAAAD while maintaining compliance with the release criterion.

5 MARSSIM Committee. *Multi-agency Radiation Survey and Site Investigation Manual.* NUREG-1575. P. 5-36. Dec. 1997.

4.1 Relative Radionuclide Spectrum

Surface areas were scanned and the locations of elevated radioactivity were identified for sampling. Surface material samples were collected to represent the radionuclide spectrum, *i.e*, concentration relative to each other, to enable a composite limit to be derived. The concentration of key radionuclides of interest was measured by alpha or gamma spectrometry. To allow for variability, the reported concentration of each key radionuclide in each sample and the history of each area may be considered.

4.2 Derivation of Alpha Limit

To accommodate compliance testing by gross α measurement, limits expressed as a function of radionuclide or of decay series need to be interpreted in the form of a single-valued, or composite, limit.

Equipment. *Guideline* values are specified in units of $\alpha/(\text{min}\cdot 100\ \text{cm}^2)$ for natural uranium and its associated decay products and are interpreted the same way for natural thorium and its decay products.

The composite maximum acceptable areal alpha radioactivity density on equipment may be derived with the equation:

$$\text{composite}\ \ \text{MAAAD}_{\alpha} = \frac{\sum_i C_i \cdot \alpha_i}{\sum_i \frac{C_i \cdot \alpha_i}{L_i}}$$

in which weighting must be proportional to alpha emission since the limit, L_i, is expressed as α activity, where

composite MAAAD_α = maximum acceptable average areal density of alpha emitted ($\alpha/(\text{min}\cdot 100\ \text{cm}^2)$)

- C_i = is the relative fraction of the areal radioactivity of U series or Th series (*e.g.*, pCi or dis/min), referenced to parent radioactivity
- α_i = number of alpha particles emitted in decay series per unit radioactivity of parent radionuclide i (α/parent dis). $\alpha_i = 8\ \alpha/U^{238}$ dis. $\alpha_i = 6\ \alpha/Th^{232}$ dis.
- L_i = maximum acceptable average areal density of U series or Th series ($\alpha/(\text{min}\cdot 100\ \text{cm}^2)$)

Whenever measurements are not available to derive it, a U series -to- Th series ratio = 2 -to-1, referenced to parent radionuclide, may be assumed.

Buildings. A maximum acceptable average areal density ($DCGL_W$) on a building surface is derived in terms of parent radioactivity/area in units dis/(min·100 cm^2). A composite $DCGL_W$ would be derived in proportion to the relative radioactivity fractions of each chain according to the equation:

$$\text{composite}\ \ \text{DCGL}_{W_{ry}} = \frac{\sum_i C_i}{\sum_i \frac{C_i}{Lm_i}}$$

where

composite $DCGL_{W\,ry}$ = maximum acceptable average areal radioactivity density (dis/(min·100 cm^2)

C_i = is the relative fraction of the radioactivity of U series or Th series (*e.g.*, pCi or dis/min), referenced to parent radioactivity

Lm_i = maximum acceptable average areal density of U series or Th series (dis/(min·100 cm^2) referenced to parent or key indicator radionuclide.

Since the limits, Lm_i, are expressed in radioactivity, the weighting factor should also be in radioactivity units.

An equivalent composite limit expressed in α activity/area may be derived as a measurable quantity for an alpha survey instrument. Essentially, it is the product of *composite radioactivity limit* times *composite alpha emission/dis*. An exact expression is:

$$DCGL_{W\alpha} = \frac{\sum_i C_i \cdot \alpha_i}{\sum_i \frac{C_i}{Lm_i}}$$

where

$DCGL_{W\alpha}$ = composite alpha limit, *i.e.*, maximum acceptable average areal alpha ray density for a building surface α/(min·100 cm^2)

Whenever measurements are not available to derive it, a U series -to- Th series ratio = 2 -to- 1, referenced to parent radionuclide, may be assumed.

Alternatively, in the event a building is to be surveyed for release subject to the NRC "Guidelines" (license conditions 16), then the basic limits in the previous paragraph (Equpment) applies.

5. MEASUREMENTS

5.1 Interpretation

The composite $MAAAD_\alpha$ is derived to be tested by a portable instrument that detects gross alpha radiation emitted from a surface. Each measurement will be interpreted to represent alpha radiation emitted from regulated radioactive material on the surface per unit time and area, *e.g.*, in units α/(min·100 cm^2). Measurements will be interpreted according to the relation

$$\alpha \text{ emitted} = \frac{N}{t_s \cdot \frac{A}{100} \cdot E_t}$$

where N = number of counts registered from α rays in the instrument in time, t_s. (ct)

t_s = measurement integration time, or count time (min)

E_t = detection efficiency (ct/min)/(dis/min)

$$= \frac{\text{net count rate}}{\text{total radioactivity under physical area of detector window}}$$

$= \varepsilon_i \cdot \varepsilon_s$ where ε_I = instrument detection efficiency and ε_s = source efficiency

A = detector area (cm^2)

100 = reference area (cm^2)

5.2 Detection Efficiency.

Instrument efficiency will be estimated by measurements made on a plated or deposited standard source with an instrument and detector of the type used to perform the survey. Radioactive sources used to determine detection efficiency will be selected to have alpha energy approximating that emitted by U series and Th series as closely as practical. Detector efficiency, E_t, has units (ct/min)/α emitted/min) to interpret in units comparable with the $MAAAD_\alpha$.

In the event an alpha survey instrument is used on a bare surface other than a smooth[6] one, the source efficiency would be estimated to be:[7]

Surface Material	Source Efficiency
Concrete	0.4
Brick	0.4
Steel, bare	0.2
Wood	0.5
Gypsum Board	0.5

An alpha survey detector will not be used to survey a potentially contaminated surface, subsequently coated, because the coating would absorb the alpha particles.

5.3 Scanning

In the event static measurements, *i.e.*, systematic and judgment-biased measurements at fixed locations, are made by alpha survey, any scanning specified by the same final status survey plan would be performed by beta survey. Since the source efficiency for β radiation is greater than it is for α radiation, scanning by β survey will provide increased assurance of detecting any contamination that might be attenuated by paint or similar coating.

6. BETA EQUIVALENT

A composite beta equivalent of the composite alpha limit, $MAAAD_\alpha$, of a U series and Th series mixture may be derived. A composite $MAAAD_\alpha$ is first derived with the equation in section 4.2 where the

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[6] *e.g.*, glass, formica, linoleum, or a painted surface on which radioactive contamination might deposit are examples of surfaces that would be considered smooth.

[7] NUREG-1507, Table 5-5

$$\text{composite MAAAD}_{\alpha} = \frac{\sum_i C_i \cdot \alpha_i}{\sum_i \frac{C_i \cdot \alpha_i}{L_i}}$$

The beta-to-alpha ray ratio of a U series and Th series mixture is derived by the equation

$$\left(\frac{\beta}{\alpha}\right)_{mix} = \frac{\sum_i C_i \alpha_i \left(\beta/\alpha\right)_i}{\sum_i C_i \alpha_i}$$

where $(\beta/\alpha)_i$ = β-to-α ratio in the decay series.

Then the composite beta limit, or $MAAAD_{\beta}$, equivalent to $MAAAD_{\alpha}$ in the mixture is derived with the equation $MAAAD_{\beta} = MAAAD_{\alpha} \times \left(\beta/\alpha\right)_{mix}$.

When the U series -to- Th series ratio is 2-to-1, the composite $MAAAD_{\alpha}$ = 2400 $\alpha/(\text{min}\cdot 100\ \text{cm}^2)$; $\beta/\alpha_{mix} = 0.66$; and the $MAAAD_{\beta}$ = 1600 $\beta/(\text{min}\cdot 100\ \text{cm}^2)$. Application of either would be sufficient.

ATTACHMENT 1
REVISION KEY

Mallinckrodt Inc.

Phase 1 Plan
For C-T Decommissioning

NRC Docket: 40-06563
NRC License : STB-401

January 9, 2002

Item	Location	Request for Information	Resolution
NRC letter August 7, 2000			
I.1.a.i	Appendix D	Basis of source term in accident analysis.	Attachment 6, §Accident Analysis, and Attachment 5, §ALARA Analysis, revised
I.1.a.ii	Appendix D	Source term sample description.	Attachment 6, §Accident Analysis, and Attachment 5, §ALARA Analysis, revised
I.1.a.iii	Appendix D	Detectable beta	Appendix D, §Composite U Series …, last ¶ inserted to explain. Will benchmark instrument calibration to Lucas & Colyott report with Sr^{90} calibration. Equation in Appendix D.
I.2.a	Admin. Ctrl. Plan	Safety Committee role.	Attachment 2, Administrative Control Plan revised to omit Safety Committee
I.2.b	Admin. Ctrl. Plan	HSES position	Attachment 2, Administrative Control Plan revised to omit HSES.
I.3.a			Resolved earlier.
I.4.a			Resolved earlier.
I.4.b	Admin. Ctrl. Plan	HSES position.	Attachment 2, Administrative Control Plan revised to omit HSES.
I.5.a	3.3, 3.3.1, & 3.3.3	ALARA	§3.3, 3.3.1, & 3.3.2 revised to address ALARA.
I.5.b	2.6	Change to decommissioning process.	§2.6 revised. Statement inserted in §2.6 after criterion d to identify and report any issue that might violate §2.6 criteria a through d.
I.6.a			Resolved earlier.
I.6.b	Table 3-2	"TBD"	Table 3-2 footnote revised to delete the use of the term of TBD which stood for To Be Determined.
I.7.a	Table D2. 3.6.3	Radionuclide characterization and waste classification	Characterization data are in Appendix D, Table D2. §3.6.3 revised to interpret data. Data histogram and answer to classification §3.6.3.
I.7.b	DFP	Decommissioning Funding Plan	Submitted separately.
I.7.c	3.6.4.2	Mixed waste	3.6.4.2 revised to refer reader to 2.2.4 which shows how we expect to handle mixed waste if found.
I.8.a			Resolved earlier.

I.8.b	2.2.4 3.6.4.2	"… an NRC-authorized …" transfer	2.2.4, item 2 revised and 3.6.4.2, item 2 revised with wording matching in both paragraphs.
I.9.a	2.2.4 3.6.4.2 Appendix F	Mixed waste	3.6.4.2, item 4 revised, saying that Mallinckrodt will notify the NRC…. Appendix F, p. F3 revised.
I.10			No response required.
I.11			No response required.
I.12.a		Building rubble release criteria & final status survey	§2.2.1, 2.2.4, 4.3, 4.4.1, 4.4.2.1, 4.4.6, 4.4.6.1, 4.4.8.1, and 4.4.8.3 were revised to delete rubble release criteria and final status survey.
I.12.b	Table 2-2 Table 2-3	Reference to Tables 2-2 and 2-3.	Table 2-2 and Table 2-3 deleted. Text revised.
I.13.a			Resolved earlier.
I.14.b	DFP	Decommissioning Funding Plan	Submitted separately.
I.15.1.a	FSS Design Guide	Survey unit characteristics. Specific survey design requirements. DQO, data evaluation	Info requested is addressed in Final Status Survey Design Guide.
I.15.1.b	4.4.1.1 4.4.1.2	Distinguish between equipment and building survey. Provide additional description of equipment survey.	Removable equipment is defined in §4.4.1.1. Equipment release criteria are in §2.2.2. Table 2-1 provides both average, maximum, and removable limits. Equipment survey specifications are in revised §4.4.7.1. Installed apparatus and other parts of a building are defined in §4.4.1.2. Release criteria for installed apparatus and other parts of a building are in §2.2.3. Survey specifications for installed apparatus and other parts of a building are in §4.4.1.2 and the FSS Design Guide.
I.15.1.c	Appendix A	Negative results of surface radioactivity measurements	Explanation is in Appendix A, p. 1, new introduction
I.15.1.d	4.4.7.2	10% scan on a class 2 survey unit.	4.4.7.2 §Class 2 Areas, revised.
I.15.1.e	Table 4-3	Background count rates are reported as ambient + material. "… for these and other…"	Footnote added to Table 4-3. ¶ following Table 4-3 revised.

I.15.1.f	Table 4-8	Investigation levels	Table 4-5 revised.
I.15.1.g	4.4.8.4	Include WRS test criterion α	Now §4.4.8.4, item 6, revised to include α
I.15.1.h	Table 2-1 Appendix D Appendix H	U -to- Th ratio 2 -to-1 and number of samples	Table 2-1 footnote revised. Appendix D, pp. D6 & D7 revised. Appendix H, pp. H1, H2, H3, H4, & H6 revised.
I.15.1.i	Appendix D	Clarify Figure D1 legend and supporting text.	Appendix D, p. D3 figure and text revised.
I.15.2.a	Appendix C	Independent derivation of DCGL on surface	Areal model adopted. DCGL revised in Appendix C.
I.15.3.a	Appendix D	AB-100 calibration method	Lucas & Colyott report on AB-100 calibration. Appendix D, §Detection Sensitivity Derivation, p. D10 Appendix D, §Composite Detection Sensitivity, p. D10
I.15.3.b	Appendix D	Effect of ambient gamma radiation on LLD?	Appendix D, §Ambient Gamma Radiation, pp. D15 & D16.
I.15.3.c	Appendix D	List numbering	Appendix D, p. D12, list numbering was revised.
I.15.4.a	FSS Design Guide	Specific survey designs and sampling density by survey unit.	Detailed guidance for final status survey design specifications is in the Final Status Survey Design Guide.
I.15.5.a	4.4.5.1		Resolved earlier.
I.15.6.a	4.4.5.1	Reference NUREG-1505, equation 9-6	§4.4.5.1, p. 4-11; reference was revised.
I.15.7.a	4.4		See response to item 15.1.a.
II.1	§1	Literature references	References added at end of §1.
II.2	2.2.2	Table 2-1, footnote a	Table 2-1 and footnote were revised.
II.3	3.3	ALARA emphasis	§3.3 & 3.3.1 were revised. §3.3.3 was added.
II.4	3.4.1	Effluent air monitoring	§3.4.1 was revised.
II.5	3.4.3	Liquid effluent monitoring	§3.4.3 was revised.
September 21, 2000 Meeting			
I.1	4.4.1.1 4.4.1.2 Appendix H Appendix A	Why does equipment release only consider α radiation? Describe type of equipment in survey units	Appendix H §4.2 and §6 explain. Appendix A lists equipment.
I.2	2.2.2	Equipment surface release limits	§2.2.2 states criteria for removable equipment. §2.2.3 states criteria for installed apparatus, *i.e.* installed equipment. §4.4.1.1 and 4.4.1.2 distinguish between kinds of apparatus or equipment.

I.3	2.2.3 Appendix D Appendix H	α or β release limits for buildings	§2.2.3 was revised to clarify. Appendix D explains interpretation of beta measurements to compare with building release limits Appendix H explains interpretation of alpha measurements to compare with removable equipment release limits.
I.4	2.3 & 2.2.3	Clarify p.2-2, final ¶ and footnote	§2, p.2-3 footnote 6 revised
I.5	2.3.1	Purpose for removing and fixing contamination on floor slab. Administrative controls on sealed floor slab?	§2.3.1 last bullet item revised
I.6.a	3.6.3 Figure Appx D Table D2.	How representative are 74 material samples?	74 sample data in Table D2 were added. Attachment 6, §Accident Analysis and Attachment 5 §ALARA Analysis, revised.§3.6.3 histogram of radioactivity concentration inserted. Appendix D, histogram of derived $DCGL_W$ revised.
I.6.b	Appendix C	Contamination depth in brick and concrete	Appendix C mathematical model changed to area model, assuming planar source. Appendix C revised. §2.2.3 $DCGL_W$ revised. $DCGL_W$ in Appendix D revised.
I.6.c	Appendix C	Apparent depletion of source term	Is a discrepant artifact in RESRAD-BUILD code.
I.6.d	Appendix C	Reconsider basis of DCGL derivation	Appendix C mathematical model changed to area model, assuming planar source. Appendix C revised. §2.2.3 $DCGL_W$ revised. $DCGL_W$ in Appendix D revised.
I.6.e	Figure D1	Explain data in Figure D1.	Figure D1 revised. Appendix D text added at Figure D1 to explain data. Data tabulated in Table D2.
I.6.f	Appendix D	Need discussion about equilibrium. Address variability among the 74 samples.	Revised Appendix D, §Approximate Equilibrium, to explain. Provided Figure D1 histogram to illustrate.
I.6.g	§2.2.2 Appendix D Appendix H	Apply one U -to- Th ratio consistently.	Revised DP §2.2 and Appendices D & H to apply U -to- Th = 2 -to- 1 consistently.
I.6.h	Appendix D	Describe beta and alpha shields in measurements	Appendix D, §Beta Radiation Measurement, p. D9, footnotes 4 & 5. added
I.6.i	Appendix D	Provide MDC for static and scan measurements.	Appendix D, §Lower Limit of Detection added on pp.

		Identify instruments used to survey internal surfaces of equipment.	D14 thru D16. Instrumentation and best estimates of *a priori* LLD (or MDC) are in revised Tables 4-1 & 4-2.
I.6.j	Appendix D	Potassium-40	Appendix D: - p. D12 - §Beta Radiation Measurement, pp. D7, D8, & D9 - §Survey Method, p. D9 - §Accounting for K^{40}, pp. D12 & D13 - §Composite U Series and Th Series …, p. D13 - Table D2, columns: K^{40}, K^{40} fraction of total radio-activity, and fraction of β emitted by K^{40}
I.6.k	AB-100 calibration report	Include final AB-100 calibration report by Lucas & Colyott. Explain how data in report will be used to benchmark instrument(s) used on-site.	Final AB-100 calibration report by Lucas and Colyott submitted. Explanation of use is in Appendix D, §Composite Detection Sensitivity, pp. D10 & D11.
I.6.l	Table 4-1 Table 4-2	Differentiate purpose of survey instruments.	Instrument usage column added to Tables 4-1 and 4-2. Instruments identified as FSS or Internal Surface are currently expected to be used in final status survey. Radiation protection instrumentation in Tables 3-2 and 3-3
I.6.m	Appendix D	Clarify terms E_c, G_{K40}, AD_c, & n_{K40}	Closed. Revised Appendix D, p. D12 – 14..
October 13 Meeting			
1.	§1	Update drawings, tables, and Appendix A	§1, text, tables, and drawings updated. §2.3 & 2.3.1 updated. Appendix A updated.
2.	drawings	Revise features of buildings on drawings to make current	Updated drawings in Figures 1-2, 1-3, 1-5, 1-7, 1-8, 1-9, 2-1, and 2-2.
3.	Table 1-5	Make footnote to Table 1-5 refer to a specific list in Appendix A.	Table 1-5 footnote revised
4.	Appendix A	Review Appendix A to include inventory of buildings subject to final status survey	Appendix A revised.
5.	Appendix A	Provide additional detail in Appendix A, such as	Appendix A has been revised to provide greater detail

		- "This room includes dead space between wall." - " This roof …"	than before.
6.	Appendix A	Say that structures erected after CT operations are not subject to FSS	Appendix A lists structures and apparatus subject to final status survey. If not in Appendix A, a building or structure is not subject to final status survey.
7.	DP §5.8.3 Data Evaluation	Verify that characterization survey data will be subject to quality verification before use as final status survey data.	DP §5.8.3 has been revised.
8.	Table 1-5	Add a footnote to Table 1-5 to refer to a specific list in Appendix A.	Footnote added to §1.6.4 Table 1-5, Ref. §1.6.3. Ref. §4.0. Ref §4.4.1.1, §4.4.1.2, §4.4.7.1
9.	Appendix A	Review Appendix A and include inventory of buildings that will be included and or excluded from final status survey.	Appendix A revised. Buildings and installed apparatus subject to final status survey are classified as class 1, class 2, or class 3.
10.	Appendix A	Say that structures erected after CT operations ceased are not subject to final status survey.	Done in Appendix A introductory page
11.	Appendix A	Consider each "TBD" i.e., " to be done" in Appendix A. Example responses could be: TBD assumed clean pending survey NA (not applicable) history non-impacted TBD for H&S assumed clean demolish	Additional definition provided in Column *Location/Surface* Additional consideration during survey design.
12.	Appendix A	Add notes to Appendix A to identify false wall covering potentially contaminated, original wall	Additional definition provided in Column *Location/Surface*
13.	Tables 4-4, 4-5, & 4-6.	Consider removing Tables 4-4 and 4-5 from the decommissioning plan. Instead, refer the reader to Appendix A.	Tables 4-4, 4-5, and 4-6 were deleted from DP §4. Remaining tables were renumbered.
14.	§1, Tables 1-1 thru 1-5	Revise DP Tables 1 thru 5 as needed to update.	Table 1-1 = no change Table 1-2 = has been updated Table 1-3 = no change Table 1-4 = no change Table 1-5 = has been updated
15.	DP §5.8.3 Data	What quality assurance qualification must characterization survey data pass to be accepted	DP §5.8.3 has been revised.

	Evaluation	as final status survey data?	
16.	Appendix A & FSS Design	In final status survey design, identify apparatus installed after cessation of CT operation.	Apparatus installed after CT ceased operation noted in Appendix A.
17.	FSS Design	Write a section in DP, §4 FSS Plan, dealing with circumstances having individualistic design requirements. For instance, what would be taken into account in FSS design of casework having internal and external surfaces, against a wall? How would systematic measurements, biased measurements, and unique access issues be specified in such circumstance? Address: 1. building surfaces, indoor and outdoor, including installed apparatus; 2. roofing, with and without installed apparatus such as process equipment, ventilation units, blowers, chillers, storage tanks, etc., on a roof 3. ventilation exhaust hoods, exhaust duct, and filter housings 4. casework, including shelves, benchtops, cabinets, safety cabinets 5. open sumps and trenches 6. false walls and other remodeled areas covering CT-affected areas 7. coated surfaces	FSS Design Considerations §4.6 specifies individualistic survey requirements. Painted surface is addressed in Appendix D.
18.	DP §2.2.1	Consider areas contaminated by radioactive material that is not covered by a former license, e.g., potassium-40 and or MED/AEC material.	Revised DP §1.5.1 and Table 1-5 to exclude MED-AEC material as well as K^{40}.
19.	FSS Design	Write a summary of the FSS design process into FSS Design §2.	Done in new §4.6 of FSS Design Considerations

20.	FSS Design	Add text to FSS Design saying that design will rely on designer's judgment. When addressing specific considerations, the designer will consider available knowledge of history and process.	Done in FSS Design Considerations § 4.6
21.	FSS Design Guide	Add text in FSS Design talking about professional judgment of designer will influence: - instrument selection - type of grid - number of biased measurements - requirement for access to survey points - how a surveyor will access it, and - coating on a surface, e.g. effect of paint on surveying	Ref. FSS Design Considerations §4.6
22.	FSS Design	Say the FSS Design Guide is a living document that may be revised by Mallinckrodt, subject to the Administrative Control Process	The FSS Design is subject to management of revision in accordance with the Administrative Controls Plan.
23.	FSS Design	Put removable equipment definition and installed apparatus definition in FSS Design [ref. DP §4 FSS Plan]	Ref. DP §4.4.1.1 for equipment and §4.4.1.2 for installed apparatus.
24.	Tables 4-1 & 4-2.	Revise Tables 4-1 and 4-2 to include MDC (minimum detectable concentration)	Best effort with currently available data. Some instruments deleted. *A priori* minimum detectable areal density estimated for key instruments in Tables 4-1, esp. for AB-100 and 43-89, detectors for which reasonable background data are available.
25.	Tables 4-1 & 4-2. Appendix D §LLD	Say specific MDC will be developed. Quote NUREG-1507 for method.	Table 4-1 & 4-2 footnotes specify that minimum detectable radioactivity will be determined before initial use for final status survey. Appendix D §LLD revised to specify LLD (MDC) method.
26.	Appendix C	Derive area factor for FSS Design Guide, p. 15	Area factor derived in Appendix C. Survey specification in FSS Design Consideration §4.6

27.	§5.5 & §5.7	Mention instrument calibration, daily check, trained technicians.	Ref DP §5.7 Ref. DP §5.5
28.	§5.7	Verify instrument operation before survey.	Ref. DP §5.7
29.	DP §5.8.3	Add a ¶ mentioning: - data analysis software package will do. - will be the acceptance test before use. - validation and verification for the software package	Added a quality specification for FSS data evaluation application program in DP §5.8.3 Data Evaluation.
		Appendix C	
30.	Appendix C	Generate a new dose factor table	Done in Appendix C
31.	Appendix C	Convert to the RESRAD-BUILD area model	Done in Appendix C
32.	Appendix C	Include bases of parameter values.	Done in Appendix C
33.	Appendix C	Include new references	Done in Appendix C footnotes
34.	Appendix C	When selecting dose factor, indicate time (yr) for which dose factor is selected.	DP Appendix C, §4 has been revised to explain this
35.	Appendix C	Explain why Th^{232} +DI = Th^{232} + Th^{228}	DP Appendix C, §4 has been revised to explain this Revised Table C1 tabulates Th^{232}, Ra^{228}, and Th^{228} dose factors in 0 to 1 yr time. The sum of the 3 equals Th^{232} + progeny in radioactive equilibrium.
36.	Appendix C	Include a table of major surface model parameters	Done in Appendix C
37.	Appendix C	Submit revised DCGL and computer runs	$DCGL_W$ revised in Appendix C. Computer runs will be submitted to NRC in addendum.
38.			
		Appendix D	
39.	Appendix D	Revise Table D1	Done
40.	Appendix D	Insert Table D2	Done
41.	Appendix D	Provide more insight about Figure D1	Done
42.	Appendix D	Normalize all U-to-Th ratios to 2-to-1	Done
43.	Appendix D	On p.5, revise areal limits	Areal density limits revised in Appendices C and D. Appendix D, Fig. D1 and table of statistics were revised.
44.	§2 &	Eliminate MAAAD when referring to DCGL.	Done

	Appendices D & H		
45.	Appendix D	Write a subsection to account for paint.	Appendix D §Accounting for Paint p. D-11 added.
46.	Appendix D	Include benchmarking calibration $= \text{Lucas} \times \frac{Sr^{90}\text{new}}{Sr^{90}\text{Lucas}}$	Done in Appx D §Composite Detection Sensitivity.
47.	Appendix D, p. D-7	Check bottom p. D-7 with respect to arithmetic average.	Done. Attachment 6 is the Accident Analysis. Note added that accident analysis source term was arithmetic average of the 6 material samples in Building 238. Ref. original accident analysis in Attachment 1 of March 24, 2000 submittal.
48.	Appendix D	Walk through the process describing how beta surveys will be made.	Added to Appendix D §Beta Radiation Measurement method, p. D-7 - D-10.
49.	Appendix D	Some areas may not need to consider K^{40}. If survey fails test, consider K^{40}.	Appendix D p. D-12 revised.
50.	Appendix D	Provide more information on the aluminum shield for the AB-100 detector.	Appendix D, footnotes 4 & 7 added
51.	Appendix D	Concerning paint covering potentially contaminated surface, mention that if paint is too thick to measure, i.e. if detection efficiency is too low, then strip paint to survey.	Appendix D §Accounting for Paint added on p. D-11
Final Status Survey Design Guide			
52.	§4.2 Appendix D	Put MDC in DP §4	Done. Specification inserted in DP §4.3, p. 4-5. Detailed equations stated in Appendix D.
53.	Table 4-1	Derive *a priori* MDC	DP Table 4-1 revised. Ref. Appendix D, §Lower Limit of Detection.
54.	4.4.7.1	For removable equipment, consider - operational history - contamination potential - whether dismantling is practical	Done. DP §4.4.7.1 was revised to include these items.

		- whether there is an accessibility issue - whether it may contain sediment, requiring volumetric source consideration - identify measurements to be made - RSO review and approval	
55.		Remove forms and flow diagrams. Refer reader to procedures.	Data forms and diagrams removed from FSS Design Guide.
		ATTACHMENT A (FSS DATA EVALUATION)	
56.	Admin. Ctrl Plan	§1. Review and approve survey design is already done and gone by the time we get to evaluation of survey results.	Review and approval process for FSS Design is in Administrative Control Plan
57.	QAPP	§2.2. Admin: Who would screen incoming data?	Data requirements are specified in FSS design. An analyst checks conformance of survey results with survey design specifications for the QA Manager.
58.	DP §4.4.8 & 4.4.8.1.	§2.3. Data requirements are specified in FSS design. Reviewer questions conformance of survey results with survey design specifications.	Administrative Control Plan revised to include review and approval of FSS Design Guide and final status survey designs.
59.	QAPP	If organization is needed for FSS Design and review, put in the DP §2.4 Put 2 or 3 sentences in DP §2 responsibilities section.	Administrative Control Plan revised to include FSS Design approval.
60.	Appendix D	Talk about screening out K^{40}. Discuss the contribution of K^{40} and other sources.	Appendix D revised to describe accounting for K^{40}. Ref. Appendix D, pp. D7, D8, D9, D10, *-D11, D12, and D13. DP §4.4.8.2
61.	4.4.8.2 4.4.8.3 4.4.8.4 & 4.4.9 4.4.8	What flags will be in the statistics package? What action if a flag trips?	Individual measurement tests include: - Min/Max - Background - $DCGL_W$, and - Elevated measurement Investigation levels for individual measurements are specified in §4.4.8.3. Table 4-5. Collective measurements statistical test includes one of: - Sign test

			- Wilcoxon Rank Sum test, or - Sign Test for Paired Data. Action if a test is failed is described for various tests and circumstances in DP §4.4.8.2, 4.4.8.3, 4.4.8.4, & 4.4.9
62.	§4.4.8 & 4.4.9	Discuss survey reclassification. Put in DP §4.4.8.4.	DP §4.4.8.3 ¶1, 4.4.8.4 ¶1, & 4.4.9 address reclassification.
63.	DP §4.4.8	Remove Attachment A, Data Evaluation, from FSS design Guide and put in DP §4.	Attachment A removed from FSS Design Guide. DP §4.4.8 revised in include Data Evaluation specifications
Meeting with NRC on November 13, 2000			
I.5.b	2.6	Responsibility to identify and report any issue that might violate §2.6 criteria a through d. Put a statement in §2.6 after criteria a through d.	Statement inserted in §2.6 after criterion d.
I.6.b	Table 3-2	Fix in Table 3-2	Done.
I.7.a		Supply data histogram and answer classification	Done.
I.14.b		Decommissioning funding plan	Is being handled separately at the same time.
I.15.b	4.4.1.1 & 4.4.1.2	Equipment versus installed apparatus	Definitions expanded and clarified with examples.
I.15.b	4.4.1.1 and Table 2-1.	Equipment will satisfy the average and maximum of *Guidelines*. Discuss what is equipment.	Equipment definition expanded and clarified with examples in §4.4.1.1. Table 2-1 provides both average, maximum, and removable limits
I.15.b	4.4.1.1	Provide for survey and unrestricted release of high value items, e.g. inconel furnace tube or filter press	Done in §4.4.1.1
I.15.b	4.4.1.1, 4.4.1.2, & FSS Design	Differentiate between removable equipment, installed apparatus, process waste, high value items	Differentiate between removable equipment in §4.4.1.1 and installed apparatus in §4.4.1.2. Deal with different survey circumstances in §4.4.7.1, §4.4.7.2 and the FSS Design
	Appendix D	Provide more text on derivation of values in Figure D1 histogram	Additional text provided at Figure D1.

Additional Request for Information			
65.	Appendix A	Identify which roofs will be surveyed.	Ref. Appendix A.
66.	Appendix A	Identify which roofs are either new construction or have been replaced after cessation of CT operation. Indicate that roofs installed after cessation of CT are not subject to final status survey.	Ref. Appendix A. Roofs installed after cessation of CT are not subject to final status survey. Refer to DP § 4.6.1
67.	Appendix A	Identify which parts of Building 250 are subject to final status survey, for instance the counting room downstairs, the laboratory(ies) upstairs.	Appendix A was updated and is explained on page A-0.
68.	Appendix A	Identify or specify that only exterior walls of building 245 are intended for final status survey.	Exterior walls only of building 245 are intended for final status survey. Explained in Appendix A, page A-0.
69.	Appendix A	Be more exact about describing location and description of each location subject to final status survey. This might be done by adding a column to Table 1-5 or by some kind of reference to Appendix A or added indicator to Appendix A. That is, crisply define areas to be final status surveyed.	Appendix A column: *Location/Surface* includes expanded descriptions.
70.	Appendix A Table 1-5	Remove Buildings that no longer exist, including 201 and 221, from lists.	Deleted from Table 1-5 and Appendix A
71.	Appendix A	Indicate that Building 240 exterior and roof only will be final status surveyed	Explained in Appendix A, page A-0.
72.	Appendix A	Identify building 250 roof as one that has been replaced after cessation of CT operation and therefore will not be surveyed. Revise Tables to exclude B250 roof from FSS.	Appendix A has been revised. Refer to §4.7 Survey Method and Considerations
73.	DP § 4	Describe how you will survey behind Building 250 walk-in hood panels	Refer to §4.73 & 4.7.8 Survey Method and Considerations
74.	DP § 4	Describe how you will move survey points in the event a designated survey point is inaccessible	Refer to §4.7.2 Survey Method and Considerations
75.	DP § 5.8.3	Describe how the quality of previous survey data, e.g., "FSS equivalent survey" or	DP § 5.8.3 revised.

		characterization data would be verified if used as part of a final status survey.	
76.	Appendix A	Indicate whether new construction or roof replaced since CT shutdown in 1987.	Described in Appendix A, column: *Location/Surface.*
77.	Appendix A	Identify parts of Building 250 subject to FSS	Ref. Appendix A.
78.	Appendix A	Note in Table 1-5 that Building 245 exterior only will be surveyed.	Ref. Appendix A
79.	Appendix D	Painted areas: address as part of calibration problem.	Appendix D, §Accounting for Paint, p. D-10 has been added to address paint layer.
80.	DP § 4	Where survey points are inaccessible	Refer to §4.7.2, 4.7.6, and 4.7.8 for survey design methods and consideration
Additional Request for Information (November 15, 2001 Letter from NRC & meetings December, 2001)			
1.		Use of FC 83-23 with reguard to release of material and equipment	§ 2.2.4 has been revised to satisfy this request for information.
2.		Justification of use of Parameters for RESRAD Build Modeling	Appendix C and Appendix D Table D1 have been revised to satisfy this request for information.

ATTACHMENT 2
ADMINISTRATIVE CONTROLS PLAN

Mallinckrodt Inc.

Phase 1 Plan
For C-T Decommissioning

NRC Docket: 40-06563
NRC License : STB-401

January 9, 2002

Administrative Controls Plan

The managers of Mallinckrodt Chemical intend to decommission the C-T facilities in a safe and controlled manner. The aim of this plan is to establish guidelines for creation, use, and control of administrative controls for C-T decommissioning activities involving radioactive material on the Mallinckrodt St. Louis downtown site. The objectives of these administrative controls are to ensure that C-T decommissioning is performed safely and in compliance with governing regulations, the NRC license, and the NRC-approved decommissioning plan. Clear, concise and technically correct procedures, field instructions, and or safety work permits are to be used to control activities which have the potential for adversely affecting employee safety or health, public safety or health, or adversely affecting the environment.

The main administrative controls to ensure safety and regulatory compliance during C-T decommissioning are:

1. plan (industrial safety, radiation protection,)
2. permanent procedure
3. field instruction,
4. hot work permit, and
5. safety work permit (may combine radiation work permit and *hot* work permit.)

Procedures, field instructions, and or safety work permits provide management with means of asserting control over an activity necessary to achieve the requisite outcome. Well written procedures, field instructions, and safety work permits should provide management adequate assurance that workers are provided necessary guidance to conform to laws and regulations, to be able to work safely, to achieve the objective, and to react to contingent conditions. They can also provide the user some assurance that they have been provided efficient guidance to perform their job safely and in a satisfactory manner.

Each procedure or field instruction shall contain a statement of its objective(s) and scope.

A procedure, field instruction, and or safety work permit shall state required actions its user must perform, in a logical order, that are necessary to produce the required results, safely and lawfully. A procedure, field instruction, or safety work permit shall inform its user of any limit, prohibition, or constraint they must observe while performing the activities to achieve the objective.

PROCEDURES

1. Preparation
   1.1. Proponent writes procedure.
   1.2. General format is specified.
2. Review
   2.1. Proponent submits the draft procedure to each executor:
      2.1.1. Contractor Project Manager (PM)
      2.1.2. Mallinckrodt Project Manager (PM)
      2.1.3. Mallinckrodt Radiation Safety Officer (RSO)
   2.2. Each executor reviews the procedure.
   2.3. Proponent resolves any issue that arises during review.
   2.4. If necessary, the proponent rewrites the draft
   2.5. If rewritten, the executors review the draft procedure again.
   2.6. Each approved procedure is re-examined within 2 years of its effective date.
3. Approval
   3.1. Executors approve each new or revised procedure by signature.
   3.2. Each executor may designate, in writing, an alternate who is authorized to approve a procedure when the executor is absent.
4. Revision
   4.1. Proponent writes proposed revision to procedure.
   4.2. Executors review and approve the revisions by signature.
   4.3. If a procedure or FI cannot be performed as specified, the Mallinckrodt PM and or the Contractor PM, or in absence of either, his designated alternate, may authorize a temporary revision of a procedure by signature.
   4.4. A minor revision unrelated to safety or regulatory compliance may be approved by the Mallinckrodt PM or the Contractor PM, or in absence of either, his designated alternate.
5. Control
   5.1. The QA Representative[1] or the document control clerk distributes a controlled copy of each approved new or revised procedure to each Controlled Copy recipient.

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[1] The remediation contractor's Quality Assurance Representative may be the contractor's Quality Assurance Manager.

5.2. Mallinckrodt maintains the Controlled Copy of procedures until the NRC license is terminated.

6. Training

6.1. The Contractor Environmental, Safety and Health Representative, with Mallinckrodt RSO concurrence, specifies training required to *qualify* users to perform each procedure.

6.2. Each worker shall receive specified training before becoming *qualified* to use the procedure.

7. Use

7.1. Users are instructed to perform a task in accordance with applicable permanent procedure, temporary procedure, field instruction, and or safety work permit.

7.2. If a step cannot be performed as specified, stop and contact the supervisor.

FIELD INSTRUCTIONS

A Field Instruction (FI) may be implemented for a task that is performed once or seldom and or that is not otherwise covered by written procedure and that deserves specified control(s) to assure safety and or regulatory compliance. Most C-T decommissioning and decontamination tasks are directed by Field Instructions. A Field Instruction is a written instruction administered in the following ways.

8. Preparation

8.1. An FI is prepared by a remediation contractor proponent.

8.2. An FI format differs from the procedure format.

8.3. An FI includes a specified expiration date.

9. Review and Approval

9.1. Proponent submits the draft FI to each executor.

9.2. Each executor reviews the draft FI.

9.3. Proponent resolves any issue that arises during review.

9.4. If necessary, the proponent rewrites the draft FI.

9.5. If rewritten, the executors review it again.

9.6. FI is Reviewed and approved by signature of each executor:

9.6.1. Contractor PM

9.6.2. Mallinckrodt PM

9.6.3. Mallinckrodt RSO

9.7. Each person may designate, in writing, an alternate who is authorized to approve an FI when the member is absent.

10. Revision

10.1. Proponent writes proposed revision to FI.

10.2. Revision reviewed and approved by each executor:

10.2.1. Contractor PM

10.2.2. Mallinckrodt PM

10.2.3. Mallinckrodt RSO

10.3. Each executor may designate, in writing, an alternate who is authorized to approve an FI revision.

10.4. If an FI cannot be performed as specified, the Mallinckrodt PM or the Contractor PM, or in absence of either, his designated alternate, may authorize a temporary revision by signature.

10.5. A minor revision unrelated to safety or regulatory compliance may be approved by the Mallinckrodt PM or the Contractor PM, or in absence of either, his designated alternate.

11. Control

11.1. Contractor PM sends approved FI to its Controlled Copy recipients.

11.2. Contractor PM sends approved FI to the QA Representative for retention.

11.3. Mallinckrodt maintains a file of the Controlled Copy of FI.

12. Training

12.1. An FI specifies training required to *qualify* users to perform each FI.

12.2. Each worker shall receive specified training before becoming *qualified* to work subject to the FI.

13. Use

13.1. Users are instructed to perform a task in accordance with approved, applicable procedure, FI, and or Safety Work Permit.

13.2. If a step cannot be performed as specified, stop and contact the supervisor.

SAFETY WORK PERMIT

A Safety Work Permit (SWP) specifies controls to ensure control of occupational safety, safety of members of the public, and environmental protection, including protection against unacceptable exposure to radiation and radioactive material. In the event a CT decommissioning activity needs control, but is not controlled by a written procedure, the Mallinckrodt RSO shall issue an SWP before the work begins. An SWP is administered in the following way.

14. Preparation

14.1. An SWP is prepared by a Contractor proponent or the Mallinckrodt RSO.

14.2. Each SWP includes a specified expiration date.

15. Review and Approval
    15.1. Reviewed and approved by
        15.1.1. Contractor Environmental, Safety and Health Representative
        15.1.2. Mallinckrodt RSO
    15.2. Each person may designate, in writing, an alternate who is authorized to approve an SWP .
    15.3. An SWP is issued by the Mallinckrodt RSO.

16. Control
    16.1. Mallinckrodt RSO sends approved SWP to the Document Control Officer.
    16.2. Document Control Officer sends approved SWP to Controlled Copy recipients.
    16.3. Mallinckrodt maintains a file of the Controlled Copy of SWP.

FINAL STATUS SURVEY

FSS Design

A Final Status Survey Design shall be in accordance with Section 4 of The Plan.

17. Review and Approval
    17.1. Survey designer submits the draft final status survey design (FSS Design) to each of two executors:
        17.1.1. Contractor Environmental, Safety and Health Representative, and
        17.1.2. Mallinckrodt RSO
    17.2. Each of these executor reviews the draft survey design.
    17.3. Survey designer resolves any issue that arises during review.
    17.4. Each FSS Design is reviewed and approved by signature of each executor:
        17.4.1. The final status survey designer
        17.4.2. Contractor Environmental, Safety and Health Representative.
        17.4.3. Mallinckrodt RSO.
        17.4.4. Mallinckrodt PM
        17.4.5. 17.4.5 Contractor PM
    17.5. Each executor may designate, in writing, an alternate who is authorized to approve a survey design or revision thereto when the member is absent.

18. Revision
    18.1. Survey designer writes proposed revision to FSS Design.

18.2. Revision reviewed and approved by each executor:

18.2.1. final status survey designer

18.2.2. Contractor Environmental, Safety and Health Representative

18.2.3. Mallinckrodt RSO

18.2.4. Mallinckrodt PM

18.2.5. Contractor PM

18.3. If a survey design cannot be performed as specified, the Mallinckrodt RSO or the Contractor Environmental, Safety and Health Representative, or in absence of either, his designated alternate, may authorize by signature a revision that does not diminish conformance with a DQO.

19. Control

19.1. Contractor QA Manager sends approved Final Status Survey Design to its Controlled Copy recipients.

19.2. Contractor QA Manager retains a controlled copy of each approved Final Status Survey Design.

19.3. Mallinckrodt maintains a file of the Controlled Copy of each FSS Design.

20. Use

20.1. Users are instructed to perform each survey task in accordance with the approved, applicable FSS Design and Safety Work Permit, if an SWP is issued.

20.2. If a final status survey task cannot be performed as specified, stop and contact the supervisor.

ADJUSTMENTS TO THE DECOMMISSIONING PROCESS

Mallinckrodt may make justified changes related to the decommissioning process in accordance with Section 2.6.

21. Evaluation

21.1. Either a proponent of a new administrative control document or a proposed change to one, or an executor is expected to recognize a proposed change to the decommissioning process.

21.2. The proponent of an adjustment to the decommissioning process, an executor, or other person charged by an executor shall examine the proposed change for conformance with each of the specified in DP §2.6 and shall document each determination.

21.3. The Contractor's and Mallinckrodt's Project Managers are responsible for ensuring that the project is in accordance with applicable health, safety, quality, and technical requirements. Mallincrodt PM shall be responsible for approval of operational and engineering changes. The contractor's ES&H representative and Mallinckrodt's RSO are responsible for assuring that each change conforms to health and safety program requirements.

22. Review and Approval

22.1. Determination of whether the conditions are met will be made by and each change shall require approval by

22.1.1. Mallinckrodt's PM,

22.1.2. Mallinckrodt's RSO,

22.1.3. Contractor's PM, and

22.1.4. Contractor's ES&H representative.

23. Control

23.1. Mallinckrodt shall retain records of the evaluation and approval until license termination.

ATTACHMENT 3

ENERGY DEPENDENT CALIBRATIONS FOR THE BICRON MODEL AB-100 BETA RAY SURVEY PROBE

Mallinckrodt Inc.

Phase 1 Plan
For C-T Decommissioning

NRC Docket: 40-06563
NRC License : STB-401

January 9, 2002

Energy Dependent Calibrations for the Bicron Model AB-100 Beta Ray Survey Probe

by A.C. Lucas and L.E. Colyott

1.0 Introduction. The energy dependent calibration measurements reported here have been performed in support of the calculation of nuclide specific release limits. The probe used for these measurements had been modified by the addition of a 7 mg/cm^2 mylar layer, sufficiently thick to stop alpha particles.

Accurate measurement of beta ray emitting radioactive material on a surface is complicated by the fact that the measured counting rate is inflated over the emission rate by backscattering of beta rays initially directed inward. Further, some fast electrons resulting from Compton interactions of photons with the surface layers of the contaminated surface can be interpreted incorrectly in the backgrounding process if the backgrounding absorber does not reasonably match the atomic number of the contaminated surface.

A final interpretation of counting rate data requires the separate efficiencies detailed below along with a knowledge of specific nuclide mix and release limits. This report provides the efficiencies necessary for that interpretation in a generic form suitable for interfacing with various forms of the mix calculation.

2.0 Instrumentation. The probe was powered by an HPI Model 7010 Multi-Channel Analyzer (MCA). The electrical connection and readout circuitry is identical to that employed in field readout instruments. Advantage is gained in this method by permitting energy dependent optimization with fewer measurements.

Typical pulse height spectra are shown in Figure 1. The spectrum shape is characterized by a low energy maximum representing the energy deposited by beta rays in the zinc-sulfide layer normally used to detect alpha particles and a higher energy continuum representing the energy deposited in the thin plastic scintillator layer.

The carbon-14 beta ray has sufficiently low energy that it is stopped in the plastic scintillator which is employed as the primary beta ray detector. The spectrum endpoint was determined to be at channel 428. The absolute kinetic energy of those beta rays is:

Residual energy = Initial energy - stopping power * thickness

$$135 = 156 - 3 * 7 \quad \text{kev}$$

where:

stopping power = 3 kev/mg/cm^2
thickness = 7 mg/cm^2

Channel 0 in the MCA represents 70 kev, the energy of the electron that just penetrates a 7 mg/cm^2 layer. This yields a scale factor of 0.15 kev/channel. The strontium-90 spectrum, also shown in Figure 1, represents an energy loss spectrum inasmuch as the beta rays of strontium-90 are sufficiently energetic to pass completely through the thin layer, losing only a small amount of their energy.

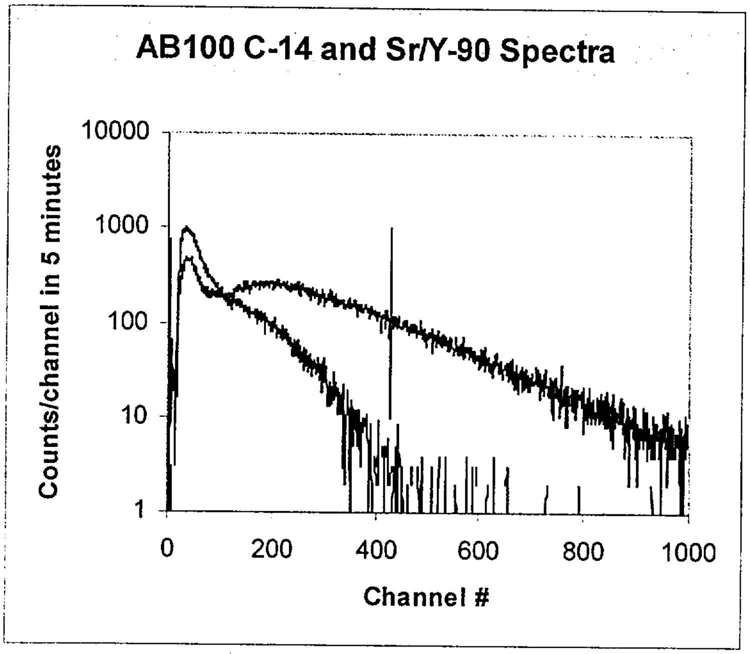

Figure 1. Pulse height spectra typical of those accumulated in these tests. The lower curve is for carbon-14 and the upper curve is for Sr/Y-90. The mark is at channel 428, the extrapolated end point for the carbon-14 spectrum.

In normal operation with the Ludlum Model 2221 scaler, the high voltage and threshold (THR) discriminator are set for optimum signal/background ratio. This condition was found to pertain in this instrument with the scale factor stated above with the THR set to 100 channels. Except where specifically stated otherwise, this is the value used in the tests below.

The NIST-traceable source set used for these calibrations was a Model BF-200 Calibration Source Set, recently purchased from Isotope Products Laboratories. The source certifications state the source activity in kBq and the number of beta rays exiting the useful face in 2-pi steradians per minute. The active material is deposited on stainless steel.

3.0 Detector Area Uniformity. The uniformity of the AB100 probe detector area was determined using a strontium-90 source. The source was supported by a thin mylar sheet and positioned on the AB100 probe surface in 1cm increments along the lateral and longitudinal axes and in 1 mm increments at the edges. Figure 2 shows the lateral profile of the AB100 probe surface, while Figure 3 shows the longitudinal profile. From these profiles, the effective area of uniformity of the AB100 probe is found to be 110 cm^2.

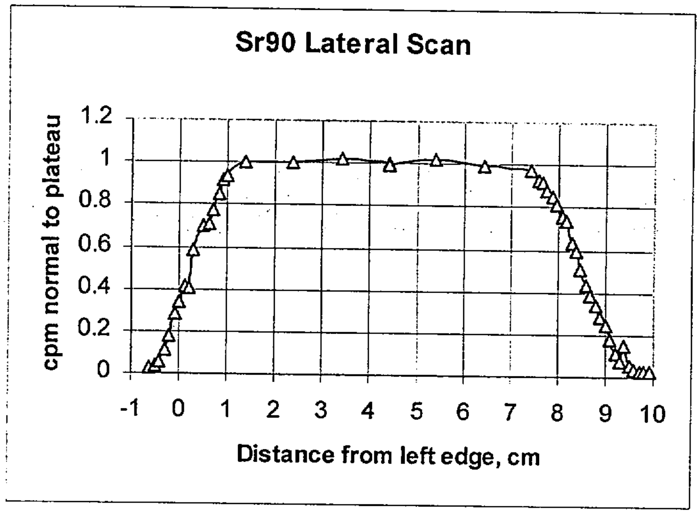

Figure 2. Lateral uniformity test using a Sr-90 source.

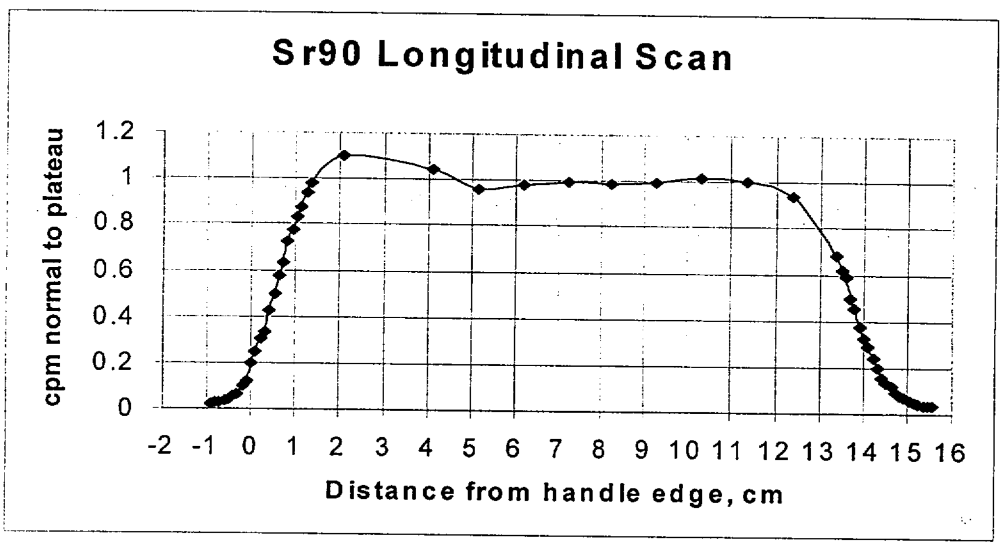

Figure 3. Longitudinal uniformity test using a Sr-90 source.

4.0 Energy Dependence. The energy dependence of the AB100 probe was determined using the six beta sources (C-14, Cl-36, Sr-90, Tc-99, Pm-147 and Pb-210) individually centered on the detector surface. The beta ray energies for the sources used were found in the computer database, RADDECAY, V2, March 1997. Figure 4 displays the resulting AB100 energy dependence using a threshold (THR) setting of channel 100 representing an electron energy of 85 kev. Data in Figure 4 are normalized to a source area of 100 cm^2.

The data of Figure 4 demonstrate the low energy cutoff generated by the 7 mg/cm^2 window in combination with the threshold setting. Noting that the graph is plotted in terms of the end point energy as the independent variable, the mean energy for any of the nuclides used in the calibration would be about one-third of the end point. The dependence of the low energy cutoff on the threshold setting would be approximately 15 kev per 100 channels. That is to say, increasing the threshold setting from 100 to 200 would increase the cutoff from 330 kev (end point) to 345 kev (end point). In other words, the shape of the energy dependence curve is not strongly dependent on the threshold setting.

Inasmuch as the calibration datum was, in each case, stated in terms of electrons/minute into 2-pi steradians, these data are representative of the absolute efficiency without backscattering. A backscattering factor greater than 1.00 would cause these response data to be increased. The derivation of the backscattering factor is discussed in section 6.

Using the energy dependence data of Figure 4, the sensitivity of the probe to any other beta ray emission can be found by interpolation and extrapolation. In particular, the sensitivity of the probe to the beta rays of uranium series, thorium series, and Potassium-40 can be found by tabulating their beta ray emissions and branching ratios along with the interpolated values of sensitivity.

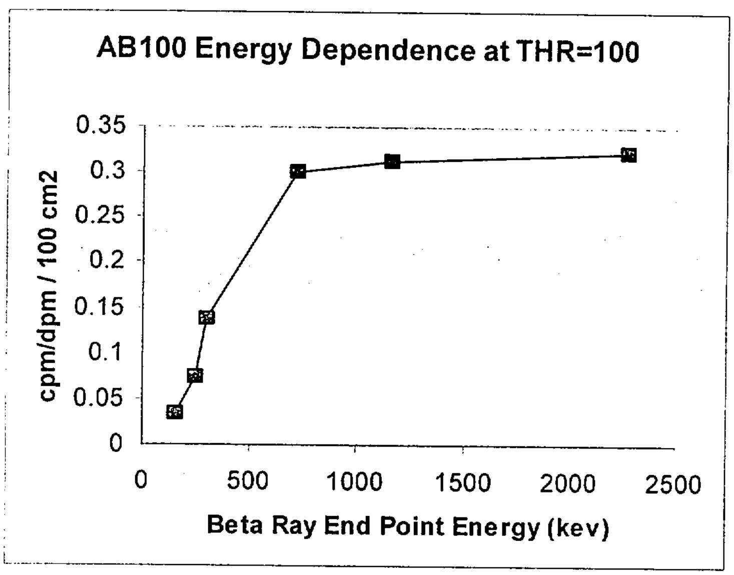

Figure 4. Energy dependence of the AB100 probe for the case of no backscattering in the source. The THR was set at channel 100 for these data. This setting combined with the window thickness establishes the minimum energy detected at 85 kev.

Appendix 1 shows the tabular data for Uranium-238 arranged in order by beta ray end point energy. The probability of emission, located in the third column, multiplied by the sensitivity for detection, located in the fourth column, yields the counting rate sensitivity for detection of β + *i.c.e.* from that particular nuclide. The summation of the sensitivities represents the net counting rate sensitivity per original uranium-238 decay. The counting rate sensitivity for β + *i.c.e.* emitted by uranium series radionuclides in radioactive equilibrium is 1.37 ct/dis of parent U^{238}. The sensitivity-weighted mean end point energy is 1.36 Mev.

Table 1. Net sensitivity of the AB100 probe to uranium-238 series beta rays and electrons.

	cpm/dpm ^{238}U
Internal Conversion Electrons	0.086
Beta Ray Sensitivity	1.281
Total ^{238}U Sensitivity	**1.367**

In a similar manner, the sensitivity of the probe for thorium-232 beta rays is developed in Table 2. The backscattering factor is developed in §6.0 below. Detection sensitivity will be increased by the backscattering factor. The counting rate sensitivity for β + *i.c.e.* emitted by thorium series radionuclides in equilibrium is 0.802 ct/dis of parent Th^{232}. The sensitivity weighted mean end point energy is 1.22 MeV.

Table 2. Net sensitivity of the AB100 probe to thorium-232 series beta rays and electrons.

	cpm/dpm ^{232}Th
Internal Conversion Electrons	0.123
Beta Ray Sensitivity	0.679
Total ^{232}Th Sensitivity	**0.802**

Potassium-40 decays with an 89.5% probability of emitting a 1.314 MeV beta ray. The probe sensitivity for potassium-40 without backscattering is, then:

S(K-40) = 0.313 x 0.895 = 0.285 cpm/dpm per 100 cm^2

5.0 Detectability. The sensitivity of the AB100 probe was determined by counting the six sources in turn for one minute each, subtracting the background from each and dividing by the source emission. This yields the sensitivity of the AB100 probe to each source. The minimum detectable activity (MDA) is then defined as

$$\textbf{MDA (counts/disintegration)} = 4.66 * (\text{bkg counts})^{1/2} / (\text{sensitivity})$$

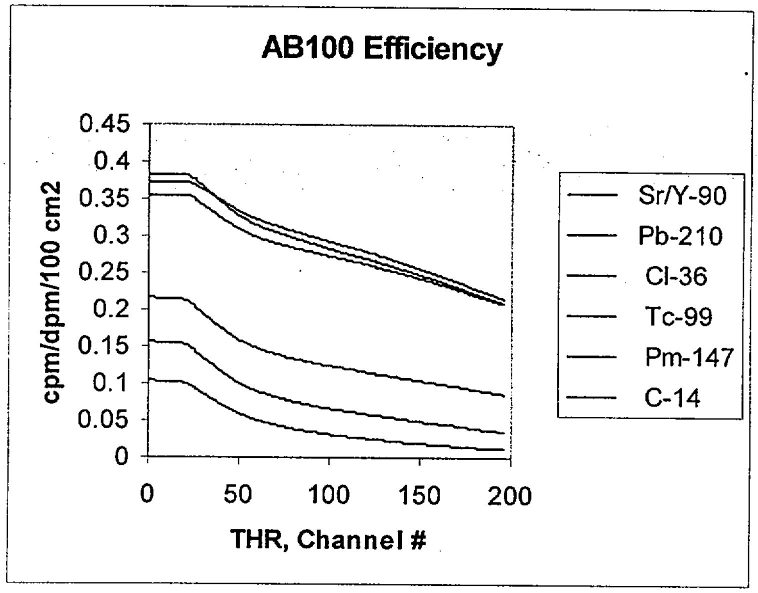

Figure 5. Efficiency as a function of THR setting. These data, along with similar data for background, permit optimization of the detectability below

Figure 5 shows the AB100 probe efficiency as a function of threshold (THR), while Figure 6 shows the AB100 probe MDA for one minute counting as a function of THR.

As described above, the data do not take backscattering into consideration. They represent the counting rates that would be observed if no backscattering were present. Finite backscattering will cause the MDA to be reduced in inverse proportion to the backscatter factor.

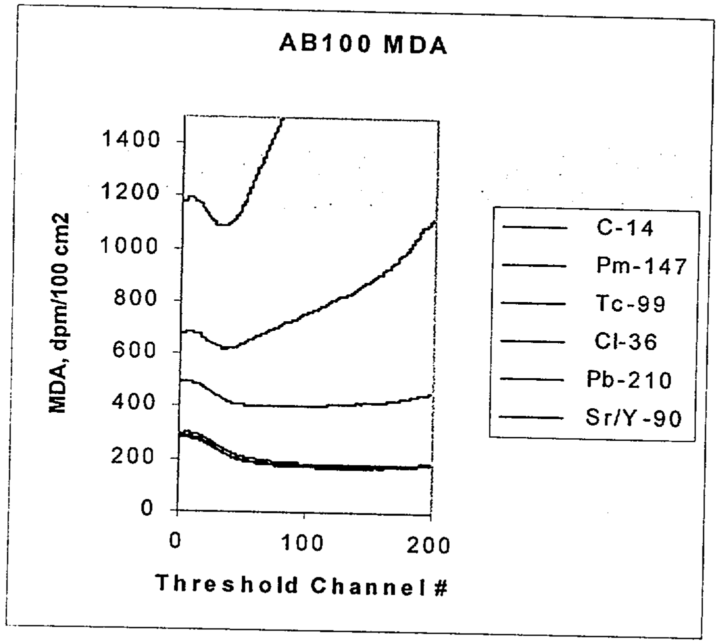

Figure 6. Minimum Detectable Activity (MDA) calculated as a function of lower level discriminator setting for the several calibrating nuclides. A counting time of 1 minute is assumed. No backscattering is assumed. Finite backscattering will cause the MDA to be reduced in inverse proportion to the backscatter factor.

6.0 Backscatter Corrections. Accurate measurement of the value of the surface deposited radioactivity is complicated by backscattering which might be different for the contaminated surface and the calibration surface. The calibration sources were deposited on stainless steel and calibrated in terms of beta rays exiting the thin window.

In the future the method described below in this report may be used as required to determine backscattering factors for additional materials such as:

Concrete
Brick
Metals
Wood
Gypsum Board

To exemplify the determination and use of the backscattering information, several experimental determinations were made. The beta ray backscatter was measured using thin layer sources of uranium and thorium deposited on thin mylar sheets. The uranium and thorium thin layer sources were coated with clear Krylon® to protect the sources and maintain secular equilibrium.

The values of the backscatter factor, i.e., the ratio of counts with backscatter to counts without backscatter, is tabulated in Table 3. These backscatter factors are to be taken as examples only as the final values must be subjective to the measuring situation. The uranium data are statistically more certain than the thorium data inasmuch as the source strength was somewhat greater.

Scatterer	Backscatter Factor	
	Uranium	Thorium
Dry Clay Soil	1.16	1.11
Lucite	1.10	1.09
Concrete	1.19	1.19
Aluminum	1.18	1.16
Brass	1.27	1.14

Table 3. Example values of the backscatter determined using uranium and thorium alternately as source material and using several materials of sufficient thickness to offer infinite backscatter.

Backscatter factors are generally smaller at lower energies and are larger with increasing atomic number.

Scatterer	Counting Rate Sensitivity cpm/dpm/100 cm^2 parent nuclide		
	Uranium	Thorium	Potassium*
Clay Soil	1.58	0.90	0.33
Lucite	1.50	0.88	0.38
Concrete	1.63	0.96	0.41
Aluminum	1.61	0.94	0.41
Brass	1.74	0.92	0.44

Table 4. Example values of the counting rate sensitivity for uranium-238, thorium-232 and their daughters along with potassium-40 in a backscattering geometry. The backscatter factor for uranium was used for the potassium column inasmuch as no thin potassium-40 source was available.

The final sensitivities are then determined by multiplying the values in Tables 1 and 2 by the backscattering factor. These values are shown in Table 4. Since the detection sensitivity is expressed per parent disintegration of U series and Th series and normalized to 100 cm^2 area, so are values in Table 4.

Additional types of material will be tested for backscattering as required.

7.0 Composite Sensitivities. The several sensitivities tabulated above form the basis for determining counting rate action levels or release limits. Some determination of the nuclide mix must be available for that calculation. In the case of the Mallinckrodt C-T building mix, data were determined using scraped samples and alpha or gamma spectrometry. This weighting is described in Appendix D to the Decommissioning plan.

8.0 Summary. A generic calibration method has been described for the Bicron AB100 beta ray detection probe. The method permits allocation of the sensitivities of individual components of a mixture to the net sensitivity and determination of the backscattering correction for subject surfaces.

The high energy calibrations are consistent with values currently employed by Mallinckrodt for use of this probe.

Appendix 1. Derivation of the net counting sensitivity of the AB100 probe for the beta rays of uranium-238 and its daughters using the measured energy dependence data of Figure 4 and neglecting backscattering. Internal conversion electron energies and branching ratios are shown in Appendix 1a and beta ray energies and branching ratios are shown in Appendix 1b. **The total uranium series sensitivity considering internal conversion electrons and beta rays is 1.367 cpm/dpm ^{238}U.**

Appendix 1a. Internal Conversion Electrons from the Uranium series

	i.c.e. Mev	Branching fraction	Efficiency cpm/dpm	U238 cpm/dpm
Th234	0.087	0.028	0.110	0.003
Ra226	0.088	0.006	0.110	0.001
U238	0.090	0.001	0.110	0.000
U234	0.100	0.001	0.130	0.000
Pb214	0.151	0.053	0.210	0.011
Ra226	0.168	0.012	0.230	0.003
Pb214	0.168	0.002	0.230	0.000
Ra226	0.182	0.003	0.230	0.001
Ra226	0.185	0.001	0.230	0.000
Pb214	0.205	0.075	0.260	0.019
Pb214	0.226	0.009	0.280	0.003
Pb214	0.238	0.003	0.302	0.001
Pb214	0.261	0.091	0.302	0.027
Pb214	0.279	0.013	0.304	0.004
Pb214	0.291	0.003	0.304	0.001
Pb214	0.295	0.001	0.304	0.000
Pb214	0.336	0.016	0.304	0.005
Bi214	0.516	0.007	0.315	0.002
Bi214	0.592	0.002	0.320	0.001
Pa234m	0.694	0.004	0.322	0.001
Bi214	1.027	0.002	0.326	0.001
Bi214	1.323	0.004	0.326	0.001
Uranium Series Sensitivity, cpm/dpm U238				**0.086**

Appendix 1b. Uranuim beta rays.

	E_{beta} Mev	Branching Fraction	Efficiency counts/electron	U238 cpm/dpm
Pb210	0.016	0.802	0.000	0.000
Pb210	0.063	0.198	0.000	0.000
Th234	0.096	0.068	0.020	0.001
Th234	0.096	0.185	0.020	0.004
Pb214	0.185	0.026	0.050	0.001
Th234	0.189	0.725	0.050	0.036
Pb214	0.490	0.145	0.220	0.032
Bi214	0.526	0.362	0.220	0.080
Bi214	0.541	0.004	0.225	0.001
Bi214	0.551	0.002	0.225	0.000
Bi214	0.575	0.001	0.240	0.000
Pb214	0.672	0.207	0.280	0.058
Pb214	0.729	0.227	0.300	0.068
Pa234m	0.747	0.002	0.300	0.001
Bi214	0.762	0.001	0.300	0.000
Bi214	0.764	0.002	0.300	0.001
Bi214	0.788	0.011	0.305	0.003
Bi214	0.822	0.028	0.306	0.009
Bi214	0.977	0.006	0.310	0.002
Bi214	1.003	0.001	0.310	0.000
Pb214	1.024	0.337	0.308	0.104
Bi214	1.061	0.003	0.310	0.001
Bi214	1.066	0.056	0.310	0.017
Bi214	1.077	0.009	0.310	0.003
Bi214	1.122	0.004	0.310	0.001
Bi214	1.151	0.044	0.310	0.014
Bi210	1.161	1.000	0.310	0.310
Bi214	1.182	0.001	0.310	0.000
Pa234m	1.236	0.007	0.312	0.002
Bi214	1.253	0.025	0.312	0.008
Bi214	1.259	0.015	0.312	0.005
Bi214	1.275	0.012	0.314	0.004
Bi214	1.380	0.016	0.314	0.005
Bi214	1.423	0.083	0.314	0.026
Pa234m	1.471	0.006	0.315	0.002
Bi214	1.506	0.177	0.315	0.056
Bi214	1.527	0.003	0.315	0.001
Bi214	1.540	0.179	0.315	0.056
Bi214	1.609	0.009	0.316	0.003
Bi214	1.727	0.034	0.317	0.011
Bi214	1.855	0.010	0.319	0.003
Bi214	1.892	0.079	0.320	0.025
Bi214	1.995	0.002	0.322	0.001
Pa234m	2.281	0.986	0.323	0.318
Bi214	2.661	0.006	0.324	0.002
Bi214	3.270	0.017	0.325	0.006

Uranium series beta ray sensitivity, cpm/dpm U238 = 1.281

Appendix 2. Derivation of the net counting sensitivity of the AB100 probe for the beta rays and conversion electrons of thorium-232 and its daughters using the measured energy dependence data of Figure 4 and neglecting backscattering. Internal conversion electron energies and branching ratios are shown in Appendix 2a and beta ray energies and branching ratios are shown in Appendix 2b. **The total thorium series sensitivity considering internal conversion electrons and beta rays is 0.809 cpm/dpm ^{232}Th.**

Appendix 2a. Int. Conversion Electron decays in Thorium series

	ice Mev	Branching fraction	Efficiency cpm/dpm	Th232 cpm/dpm
Th228	0.083	0.019	0.110	0.0021
Ac228	0.094	0.010	0.120	0.0012
Ac228	0.098	0.004	0.120	0.0004
Pb212	0.099	0.006	0.130	0.0008
Ac228	0.100	0.003	0.130	0.0004
Th232	0.106	0.001	0.140	0.0001
Ac228	0.109	0.007	0.150	0.0011
Pb212	0.111	0.002	0.150	0.0003
Th208	0.123	0.001	0.170	0.0001
Ac228	0.124	0.020	0.170	0.0033
Ac228	0.128	0.007	0.180	0.0013
Ra224	0.143	0.004	0.200	0.0008
Th208	0.145	0.000	0.200	0.0001
Pb212	0.148	0.331	0.205	0.0678
Ac228	0.161	0.001	0.210	0.0003
Ac228	0.164	0.010	0.210	0.0020
Th208	0.165	0.001	0.210	0.0002
Ac228	0.169	0.002	0.210	0.0004
Ac228	0.179	0.003	0.230	0.0007
Th208	0.189	0.011	0.240	0.0026
Bi212	0.203	0.000	0.255	0.0001
Pb212	0.210	0.013	0.280	0.0038
Pb212	0.222	0.057	0.280	0.0160
Ra224	0.223	0.005	0.280	0.0014
Ac228	0.229	0.003	0.280	0.0007
Ac228	0.231	0.001	0.290	0.0003
Pb212	0.235	0.013	0.290	0.0039
Ra224	0.237	0.002	0.290	0.0005
Pb212	0.238	0.005	0.290	0.0013
Th208	0.261	0.002	0.305	0.0006
Th208	0.274	0.001	0.305	0.0002
Pb212	0.284	0.002	0.305	0.0007
Ac228	0.353	0.001	0.308	0.0004
Th208	0.423	0.007	0.311	0.0021
Th208	0.495	0.001	0.312	0.0004
Th208	0.495	0.005	0.312	0.0014
Th208	0.507	0.000	0.312	0.0001
Th208	0.567	0.001	0.316	0.0004
Th208	0.579	0.000	0.318	0.0001
Bi212	0.634	0.000	0.319	0.0001
Ac228	0.685	0.001	0.320	0.0005
Th208	0.773	0.001	0.322	0.0003
Ac228	0.801	0.003	0.322	0.0008
Ac228	0.859	0.001	0.325	0.0004
Th208	2.527	0.001	0.325	0.0002

Thorium Series Sensitivity, cpm/dpm Th232 = 0.1230

Appendix 2b. Thorium-232 series data for beta rays

	E_{beta} Mev	Branching Fraction	Efficiency cpm/dpm	Th232 cpm/dpm
Ac228.	0.127	0.002	0.027	0.0001
Pb212	0.158	0.052	0.035	0.0018
Ac228.	0.193	0.003	0.050	0.0001
Ac228.	0.237	0.002	0.080	0.0001
Ac228.	0.244	0.002	0.080	0.0002
Ac228.	0.283	0.001	0.120	0.0001
Pb212	0.334	0.851	0.150	0.1277
Ac228.	0.377	0.002	0.170	0.0004
Ac228.	0.393	0.004	0.180	0.0007
Ac228.	0.401	0.002	0.175	0.0003
Ac228.	0.413	0.016	0.176	0.0028
Bi212	0.440	0.007	0.200	0.0015
Bi212	0.445	0.000	0.200	0.0001
Ac228.	0.449	0.024	0.200	0.0048
Ac228.	0.454	0.015	0.205	0.0032
Ac228.	0.491	0.049	0.220	0.0108
Ac228.	0.494	0.008	0.220	0.0017
Ac228.	0.499	0.013	0.220	0.0029
Bi212	0.567	0.003	0.240	0.0007
Pb212	0.573	0.099	0.245	0.0243
Ac228.	0.598	0.003	0.255	0.0007
Ac228.	0.606	0.080	0.255	0.0204
Bi212	0.625	0.022	0.270	0.0059
Ac228.	0.687	0.002	0.300	0.0007
Th208	0.712	0.001	0.300	0.0002
Bi212	0.733	0.017	0.300	0.0050
Ac228.	0.793	0.001	0.303	0.0004
Th208	0.812	0.001	0.303	0.0002
Th208	0.867	0.001	0.304	0.0002
Ac228.	0.910	0.008	0.304	0.0025
Ac228.	0.962	0.002	0.305	0.0006
Ac228.	0.969	0.033	0.305	0.0101
Ac228.	0.983	0.070	0.305	0.0214
Ac228.	1.014	0.066	0.308	0.0203
Th208	1.031	0.011	0.308	0.0032
Ac228.	1.046	0.002	0.309	0.0007
Th208	1.072	0.002	0.309	0.0006
Ac228.	1.115	0.034	0.310	0.0105
Ac228.	1.121	0.005	0.310	0.0014
Ac228.	1.158	0.002	0.311	0.0007
Ac228.	1.168	0.320	0.311	0.0995
Ac228.	1.193	0.002	0.311	0.0005
Th208	1.284	0.084	0.312	0.0261
Th208	1.517	0.082	0.316	0.0258
Bi212	1.519	0.051	0.316	0.0162
Ac228.	1.618	0.001	0.317	0.0003
Ac228.	1.741	0.120	0.318	0.0382
Th208	1.794	0.177	0.320	0.0568
Ac228.	2.079	0.080	0.322	0.0258
Bi212	2.246	0.310	0.324	0.1004
	Thorium series sensitivity, cpm/dpm Th232			**0.6793**

ATTACHMENT 4

ENERGY DEPENDENT CALIBRATIONS FOR THE LUDLUM 43-89 BETA RAY SURVEY PROBE

Mallinckrodt Inc.

Phase 1 Plan
For C-T Decommissioning

NRC Docket: 40-06563
NRC License : STB-401

January 9, 2002

ENERGY DEPENDENT CALIBRATION FOR THE LUDLUM 43-89 BETA RAY SURVEY PROBE

By A.C. Lucas and L.E. Colyott

1.0 Introduction. The energy dependent calibration measurements reported here have been performed in support of the calculation of nuclide specific release limits. The probe used for these measurements had been modified by the addition of a 7 mg/cm^2 mylar layer, sufficiently thick to stop alpha particles.

Accurate measurement of beta ray-emitting radioactive material on a surface is complicated by the fact that the measured counting rate is inflated over the emission rate by backscattering of beta rays initially directed inward. Further, some fast electrons resulting from Compton interactions of photons with the surface layers of the contaminated surface can be interpreted incorrectly in the backgrounding process if the backgrounding absorber does not reasonably match the atomic number of the contaminated surface.

A final interpretation of counting rate data requires the separate efficiencies detailed below along with a knowledge of specific nuclide mix and release limits. This report provides the efficiencies necessary for that interpretation in a generic form suitable for interfacing with various forms of the mix calculation.

2.0 Instrumentation. The probe was powered by a Ludlum 2360 alpha/beta survey meter. For the measurement of spectra, the probe was power by an HPI Model 7010 Multi-Channel Analyzer (MCA). The electrical connection and readout circuitry in the MCA is identical to that employed in field readout instruments.

Typical pulse height spectra are shown in Figure 1. The spectrum shape is characterized by low energy noise, a counting rate maximum, and a counting rate decreasing with increasing energy. The energies of the C-14 beta rays are such as to be stopped in the detector, while the energies of the Sr-90 and Y-90 beta rays are such that only part of the energy of the beta ray is left in the scintillator.

The carbon-14 beta ray has sufficiently low energy that it is stopped in the plastic scintillator which is employed as the primary beta ray detector. The strontium-90 spectrum, also shown in Figure 1, represents an energy loss spectrum inasmuch as the beta rays of strontium-90 are sufficiently energetic to pass completely through the thin layer, losing only a small amount of their energy.

In normal operation with the Ludlum Model 2360 scaler, the high voltage and threshold (THR) discriminator are set for optimum signal/background ratio.

The NIST-traceable source set used for these calibrations was a Model BF-200 Calibration Source Set, recently purchased from Isotope Products Laboratories. The source certifications state the source activity in kBq and the number of beta rays exiting the useful face in 2-pi steradians per minute. The active material is deposited on stainless steel.

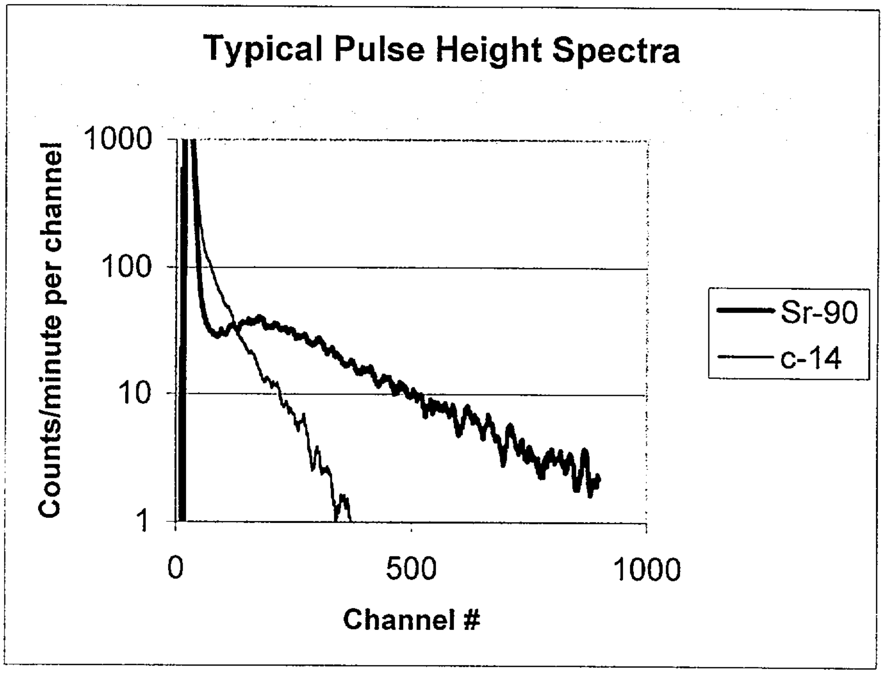

Figure 1. Pulse height spectra typical of the Ludlum 43-89 Probe.

3.0 Detector Area Uniformity. The uniformity of the Ludlum 43-89 probe detector area was determined using a strontium-90 source. The source was supported by a thin mylar sheet and positioned on the probe surface in 1cm increments along the lateral and longitudinal axes. The protective screen normally covering the thin mylar window was in place. Figure 2 shows the lateral profile of the probe surface, while Figure 3 shows the longitudinal profile. From these profiles, the effective area of uniformity of the Model 43-89 probe is found to be 120 cm^2. This compares to an area of 126 cm^2 found by measuring the width and length of the open area in the mounting frame.

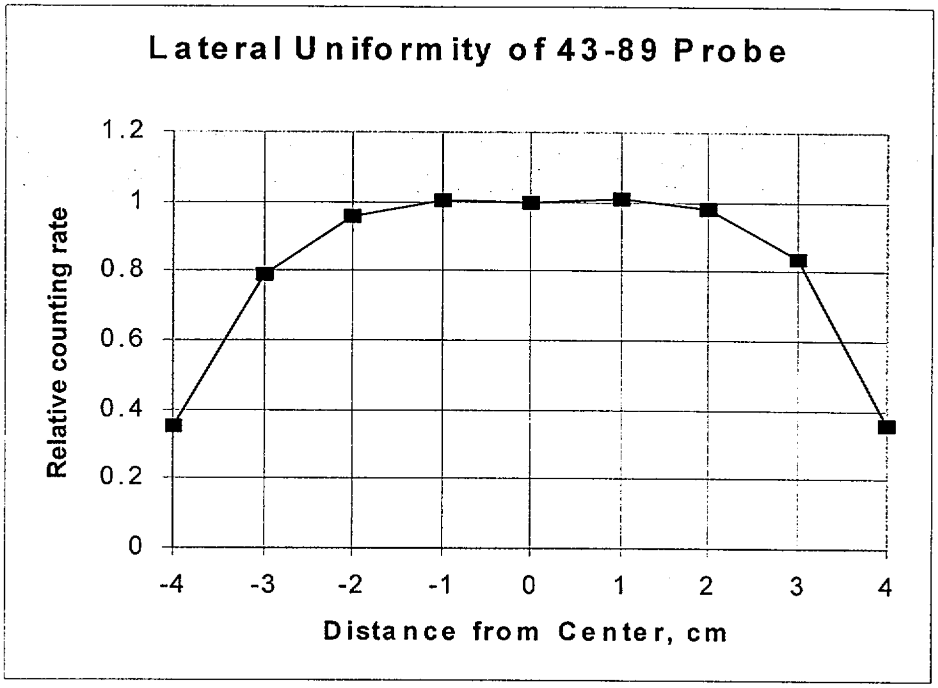

Figure 2. Lateral uniformity test using a Sr-90 source.

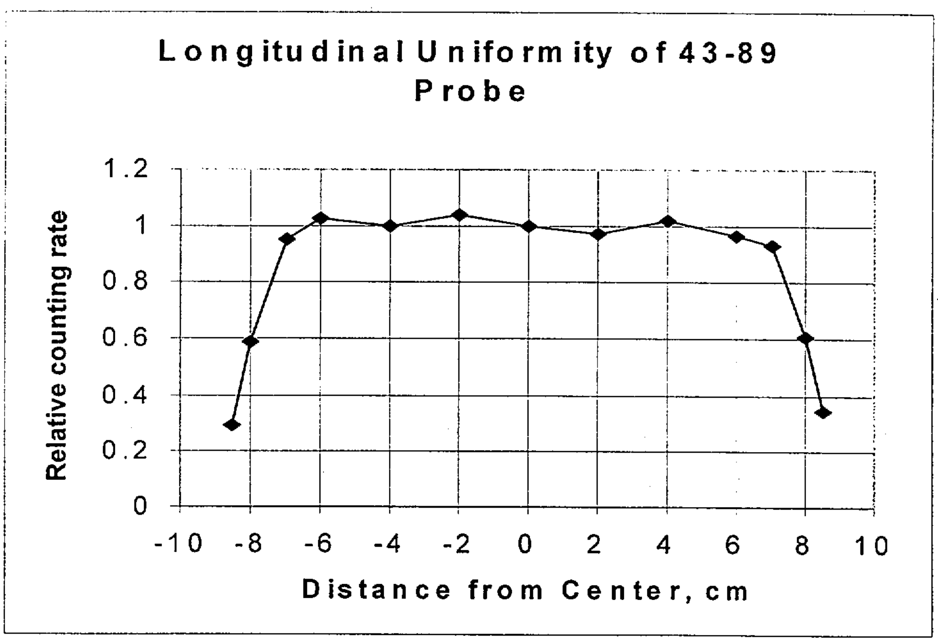

Figure 3. Longitudinal uniformity test using a Sr-90 source.

4.0 Energy Dependence. The energy dependence of the Ludlum 43-89 probe was determined using the six beta sources (C-14, Cl-36, Sr-90, Tc-99, Pm-147 and Pb-210) individually centered on the detector surface. The beta ray energies for the sources used were found in the computer database, RADDECAY, Version 2, March, 1997. Figure 4 displays the resulting probe energy dependence.

The data of Figure 4 demonstrate the low energy cutoff generated by the 7 mg/cm^2 window in combination with the threshold setting. Noting that the graph is plotted in terms of the end point energy as the independent variable, the mean energy for any of the nuclides used in the calibration would be about one-third of the end point. The energy dependence is found not to be critically dependent on the setting of the electronic threshold discriminator.

Inasmuch as the calibration datum was, in each case, stated in terms of electrons/minute into 2-pi steradians, this data is representative of the absolute efficiency without backscattering. A backscattering factor greater than 1.00 would cause this response datum to be increased. The derivation of the backscattering factor is discussed below.

Using the energy dependence data of Figure 4, the sensitivity of the probe to any other beta ray emission can be found by interpolation and extrapolation. In particular, the sensitivity of the probe to the beta rays of uranium-238, thorium-232, and Potassium-40 can be found by tabulating their beta ray emissions and branching ratios along with the interpolated values of sensitivity.

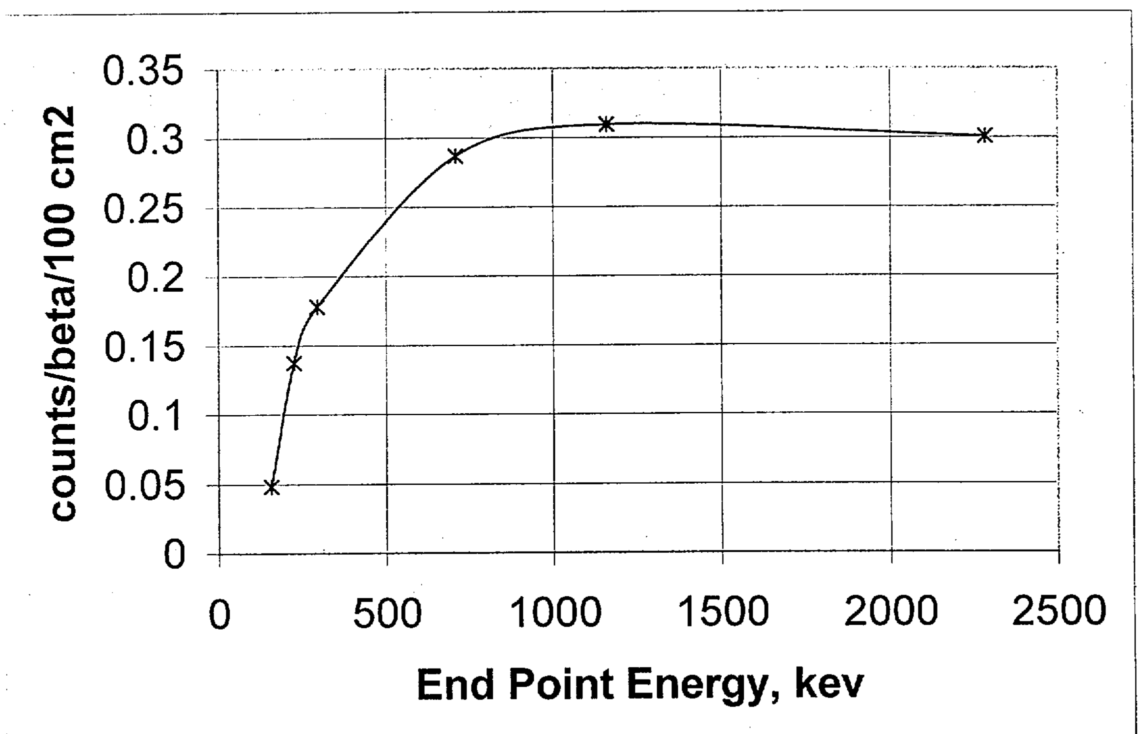

Figure 4. Energy dependence of the Ludlum 43-89 probe for the case of no additional backscattering in the source. These data refer to beta rays emitted in all directions, i.e., 4-pi and are increased by the ratio 120/100 to allow for the fact that the probe has an area of 120 cm^2.

Appendix 1 of this attachment shows the tabular data for Uranium-238 arranged in order by beta ray end point energy. The probability of emission, located in the third column, multiplied by the sensitivity for detection, located in the fourth column, yields the counting rate sensitivity for detection of that particular nuclide. The summation of the sensitivities represents the net counting rate sensitivity per original uranium-238 decay. Those values are shown in Table 1. The sensitivity weighted mean end point energy is 1.30 MeV.

Table 1. Net sensitivity of the Ludlum 43-89 probe to uranium-238 series beta rays and electrons.

	cpm/dpm ^{238}U
Internal Conversion Electrons	0.089
Beta Ray Sensitivity	1.297
Total ^{238}U Sensitivity	**1.386**

In a similar manner, the sensitivity of the probe for thorium-232 beta rays is developed in Appendix 2 of this attachment. The summary values are shown in Table 2. The backscattering factor is developed in 6.0 below. The sensitivity will be increased by the backscattering factor. The sensitivity weighted mean end point energy is 1.15 MeV.

Table 2. Net sensitivity of the Ludlum 43-89 probe to thorium-232 series beta rays and electrons.

	cpm/dpm ^{232}Th
Internal Conversion Electrons	0.1335
Beta Ray Sensitivity	0.7055
Total ^{232}Th Sensitivity	**0.839**

Potassium-40 decays with an 89.5% probability of emitting a 1.314 MeV beta ray. The probe sensitivity for **potassium-40** without backscattering is, then:

$$\mathbf{S(K\text{-}40) = 0.31 \times 0.895 = 0.277} \quad \text{cpm/dpm per } 100 \text{ cm}^2$$

5.0 Detectability. The sensitivity of the Ludlum 43-89 probe was determined by counting the six sources in turn for one minute each, subtracting the background from each and dividing by the source emission as described above. This yields the sensitivity of the probe to each source. The minimum detectable activity (MDA) is then defined as

$$\mathbf{MDA\ (dis/minute) = 4.66 * (bkg\ counts\ in\ 1\ minute)^{1/2} / (sensitivity,\ counts/dis)}$$

For the 1 minute counts used in determining sensitivity, the background counting rate was:

Bkg = 303 counts in one minute

Using this value for the background the MDA for a 1 minute count of the discrete calibration sources may then be calculated in disintegrations per minute. These example data are shown in Table 3.

Source	Sensitivity Ct/dis	MDA[a] dpm
C-14	0.048	1690
Pm-147	0.137	592
Tc-99	0.178	456
Cl-36	0.287	283
Pb-210	0.309	263
Sr-90	0.300	270

[a] For a 1 minute count

Table 3. Example values of the minimum detectably activity (MDA) for the discrete calibration sources and a typical background counting rate of 303 counts/minute. The values are given for a counting time of 1 minute.

Similarly, the minimum detectable activity may be calculated for the weighted decay series. These values are shown in Table 4. The sensitivities are normalized to a probe area of 100 cm^2 and represent the detectability per disintegration of the parent radionuclide.

Source	Sensitivity Ct/dis	MDA[a] dpm
U-238	1.386	58.5
Th-232	0.839	96.7

[a] For a 1 minute count

Table 4. Values of the minimum detectable activity (MDA) for the decay series and a typical background counting rate of 303 counts/minute.

6.0 Backscatter Corrections. Accurate measurement of the value of the surface deposited radioactivity is complicated by backscattering which might be different for the contaminated surface and the calibration surface. The calibration sources were deposited on stainless steel and calibrated in terms of beta rays exiting the thin window.

In the future the method described below in this report may be used as required to determine backscattering factors for additional materials such as:

- Concrete
- Brick
- Wood
- Gypsum Board
- Metals

To exemplify the determination and use of the backscattering information, several experimental determinations were made. The beta ray backscatter was measured using thin layer sources of uranium and thorium deposited on thin mylar sheets. The uranium and thorium thin layer sources were coated with clear Krylon® to protect the sources and maintain secular equilibrium.

The values of the backscatter factor, i.e., the ratio of counts with backscatter to counts without backscatter, is tabulated in Table 5. These backscatter factors are to be taken as examples only as the final values must be subjective to the measuring situation. The uranium data are statistically more certain than the thorium data inasmuch as the source strength was somewhat greater.

Scatterer	Backscatter Factor	
	Uranium	Thorium
Dry Clay Soil	1.16	1.11
Lucite	1.10	1.09
Concrete	1.19	1.19
Aluminum	1.18	1.16
Brass	1.27	1.14

Table 5. Example values of the backscatter determined using uranium and thorium alternately as source material and using several materials of sufficient thickness to offer infinite backscatter.

Backscatter factors are generally smaller at smaller energies and are larger with increasing atomic number.

Scatterer	cpm/dpm/100 cm^2 parent nuclide.		
	Uranium	Thorium	Potassium
Clay Soil	1.61	0.93	0.32
Lucite	1.52	0.91	0.30
Concrete	1.65	1.00	0.33
Aluminum	1.64	0.97	0.33
Brass	1.76	0.96	0.35

Table 6. Example values of the counting rate sensitivity for uranium-238, thorium-232 and their daughters along with potassium-40 in a backscattering geometry. The backscatter factor for uranium was used for the potassium column inasmuch as no thin potassium-40 source was available.

The final sensitivities are then determined by multiplying the values in Tables 1 and 2 by the backscattering factor. These values are shown in Table 6.

Additional types of material will be tested for backscattering as required.

7.0 Composite Sensitivities. The several sensitivities tabulated above form the basis for determining counting rate action levels or release limits. Some determination of the nuclide mix must be available for that calculation. In the case of the Mallinckrodt C-T building mix, data were determined using scraped samples and alpha counting. This weighting is described in Appendix D to the C-T Phase 1 Plan.

8.0 Summary. A generic calibration method has been described for the Ludlum 43-89 beta ray detection probe. The method permits allocation of the sensitivities of individual components of a mixture to the net sensitivity and determination of the backscattering correction for subject surfaces.

The high energy calibrations are consistent with values currently employed by Mallinckrodt for use of this probe. These values are consistent with those reported previously using the Bicron AB100 probe.

Appendix 1. Derivation of the net counting sensitivity of the Ludlum 43-89 probe for the beta rays of uranium-238 and its daughters using the measured energy dependence data of Figure 4 and neglecting backscattering. Internal conversion electron energies and branching ratios are shown in Appendix 1a and beta ray energies and branching ratios are shown in Appendix 1b. **The total uranium series sensitivity considering internal conversion electrons and beta rays is 1.386 cpm/dpm ^{238}U.**

Appendix 1a. Internal Conversion Electron decays from uranium series

	i.c.e. Mev	abundance fraction	Efficiency cpm/dpm	U238 cpm/dpm
Th234	0.087	0.028	0.17	0.005
Ra226	0.088	0.006	0.17	0.001
U238	0.090	0.001	0.172	0.000
U234	0.100	0.001	0.18	0.000
Pb214	0.151	0.053	0.225	0.012
Ra226	0.168	0.012	0.24	0.003
Pb214	0.168	0.002	0.24	0.000
Ra226	0.182	0.003	0.252	0.001
Ra226	0.185	0.001	0.254	0.000
Pb214	0.205	0.075	0.27	0.020
Pb214	0.226	0.009	0.283	0.003
Pb214	0.238	0.003	0.288	0.001
Pb214	0.261	0.091	0.298	0.027
Pb214	0.279	0.013	0.301	0.004
Pb214	0.291	0.003	0.303	0.001
Pb214	0.295	0.001	0.303	0.000
Pb214	0.336	0.016	0.308	0.005
Bi214	0.516	0.007	0.308	0.002
Bi214	0.592	0.002	0.306	0.001
Pa234m	0.694	0.004	0.302	0.001
Bi214	1.027	0.002	0.3	0.001
Bi214	1.323	0.004	0.3	0.001

Uranium Series Sensitivity, cpm/dpm U238 **0.089**

Appendix 1b. Beta Ray decays from uranium series

	E_{beta} Mev	Branching fraction	Efficiency cpm/dpm	U238 cpm/dpm
Pb210	0.016	0.802	0	0.000
Pb210	0.063	0.198	0	0.000
Th234	0.096	0.068	0.02	0.001
Th234	0.096	0.185	0.02	0.004
Pb214	0.185	0.026	0.1	0.003
Th234	0.189	0.725	0.1	0.073
Pb214	0.490	0.145	0.24	0.035
Bi214	0.526	0.362	0.244	0.088
Bi214	0.541	0.004	0.25	0.001
Bi214	0.551	0.002	0.253	0.001
Bi214	0.575	0.001	0.26	0.000
Pb214	0.672	0.207	0.28	0.058
Pb214	0.729	0.227	0.29	0.066
Pa234m	0.747	0.002	0.295	0.001
Bi214	0.762	0.001	0.297	0.000
Bi214	0.764	0.002	0.297	0.001
Bi214	0.788	0.011	0.298	0.003
Bi214	0.822	0.028	0.3	0.008
Bi214	0.977	0.006	0.308	0.002
Bi214	1.003	0.001	0.308	0.000
Pb214	1.024	0.337	0.308	0.104
Bi214	1.061	0.003	0.308	0.001
Bi214	1.066	0.056	0.308	0.017
Bi214	1.077	0.009	0.308	0.003
Bi214	1.122	0.004	0.308	0.001
Bi214	1.151	0.044	0.308	0.014
Bi210	1.161	1.000	0.308	0.308
Bi214	1.182	0.001	0.308	0.000
Pa234m	1.236	0.007	0.308	0.002
Bi214	1.253	0.025	0.308	0.008
Bi214	1.259	0.015	0.308	0.005
Bi214	1.275	0.012	0.308	0.004
Bi214	1.380	0.016	0.308	0.005
Bi214	1.423	0.083	0.308	0.026
Pa234m	1.471	0.006	0.308	0.002
Bi214	1.506	0.177	0.308	0.055
Bi214	1.527	0.003	0.308	0.001
Bi214	1.540	0.179	0.308	0.055
Bi214	1.609	0.009	0.308	0.003
Bi214	1.727	0.034	0.307	0.010
Bi214	1.855	0.010	0.306	0.003
Bi214	1.892	0.079	0.306	0.024
Bi214	1.995	0.002	0.304	0.001
Pa234m	2.281	0.986	0.3	0.296
Bi214	2.661	0.006	0.3	0.002
Bi214	3.270	0.017	0.3	0.005
	Uranium series sensitivity, cpm/dpm U238 =			**1.297**

Appendix 2. Derivation of the net counting sensitivity of the Ludlum 43-89 probe for the beta rays of thorium-232 and its daughters using the measured energy dependence data of Figure 4 and neglecting backscattering. Internal conversion electron energies and branching ratios are shown in Appendix 2a and beta ray energies and branching ratios are shown in Appendix 2b. **The total thorium series sensitivity considering internal conversion electrons and beta rays is 0.839 cpm/dpm ^{232}Th.**

Appendix 2a. Int. Conv. Electron decays from thorium series

	i.c.e. Mev	Branching Fraction	Efficiency cpm/dpm	Th232 cpm/dpm
Th228	0.083	0.019	0.162	0.0030
Ac228	0.094	0.010	0.174	0.0017
Ac228	0.098	0.004	0.180	0.0006
Pb212	0.099	0.006	0.180	0.0011
Ac228	0.100	0.003	0.180	0.0005
Th232	0.106	0.001	0.185	0.0002
Ac228	0.109	0.007	0.190	0.0014
Pb212	0.111	0.002	0.190	0.0004
Th208	0.123	0.001	0.203	0.0001
Ac228	0.124	0.020	0.203	0.0040
Ac228	0.128	0.007	0.208	0.0015
Ra224	0.143	0.004	0.220	0.0009
Th208	0.145	0.000	0.220	0.0001
Pb212	0.148	0.331	0.225	0.0744
Ac228	0.161	0.001	0.236	0.0003
Ac228	0.164	0.010	0.237	0.0023
Th208	0.165	0.001	0.237	0.0002
Ac228	0.169	0.002	0.242	0.0005
Ac228	0.179	0.003	0.250	0.0008
Th208	0.189	0.011	0.260	0.0029
Bi212	0.203	0.000	0.268	0.0001
Pb212	0.210	0.013	0.280	0.0038
Pb212	0.222	0.057	0.280	0.0160
Ra224	0.223	0.005	0.280	0.0014
Ac228	0.229	0.003	0.282	0.0007
Ac228	0.231	0.001	0.283	0.0003
Pb212	0.235	0.013	0.285	0.0038
Ra224	0.237	0.002	0.285	0.0005
Pb212	0.238	0.005	0.285	0.0013
Th208	0.261	0.002	0.298	0.0006
Th208	0.274	0.001	0.300	0.0002
Pb212	0.284	0.002	0.302	0.0007
Ac228	0.353	0.001	0.310	0.0004
Th208	0.423	0.007	0.311	0.0021
Th208	0.495	0.001	0.310	0.0004
Th208	0.495	0.005	0.310	0.0014
Th208	0.507	0.000	0.310	0.0001
Th208	0.567	0.001	0.308	0.0004
Th208	0.579	0.000	0.308	0.0001

Bi212	0.634	0.000	0.306	0.0001
Ac228	0.685	0.001	0.306	0.0005
Th208	0.773	0.001	0.300	0.0003
Ac228	0.801	0.003	0.300	0.0008
Ac228	0.859	0.001	0.300	0.0004
Th208	2.527	0.001	0.300	0.0002
Thorium Series Sensitivity, cpm/dpm Th232 =				**0.1335**

Appendix 2b. Thorium-232 series data for beta rays

	E_{beta} Mev	Branching fraction	Efficiency cpm/dpm	cpm/dpm Th232
Ac228.	0.127	0.002	0.020	0.0000
Pb212	0.158	0.052	0.050	0.0026
Ac228.	0.193	0.003	0.100	0.0003
Ac228.	0.237	0.002	0.160	0.0003
Ac228.	0.244	0.002	0.165	0.0004
Ac228.	0.283	0.001	0.180	0.0002
Pb212	0.334	0.851	0.190	0.1617
Ac228.	0.377	0.002	0.210	0.0005
Ac228.	0.393	0.004	0.210	0.0008
Ac228.	0.401	0.002	0.210	0.0003
Ac228.	0.413	0.016	0.212	0.0034
Bi212	0.440	0.007	0.215	0.0016
Bi212	0.445	0.000	0.215	0.0001
Ac228.	0.449	0.024	0.215	0.0052
Ac228.	0.454	0.015	0.215	0.0033
Ac228.	0.491	0.049	0.239	0.0117
Ac228.	0.494	0.008	0.239	0.0019
Ac228.	0.499	0.013	0.240	0.0031
Bi212	0.567	0.003	0.252	0.0007
Pb212	0.573	0.099	0.260	0.0257
Ac228.	0.598	0.003	0.266	0.0007
Ac228.	0.606	0.080	0.266	0.0213
Bi212	0.625	0.022	0.268	0.0059
Ac228.	0.687	0.002	0.282	0.0006
Th208	0.712	0.001	0.288	0.0002
Bi212	0.733	0.017	0.290	0.0048
Ac228.	0.793	0.001	0.300	0.0004
Th208	0.812	0.001	0.300	0.0002
Th208	0.867	0.001	0.302	0.0002
Ac228.	0.910	0.008	0.304	0.0025
Ac228.	0.962	0.002	0.305	0.0006
Ac228.	0.969	0.033	0.305	0.0101
Ac228.	0.983	0.070	0.305	0.0214
Ac228.	1.014	0.066	0.308	0.0203
Th208	1.031	0.011	0.308	0.0032
Ac228.	1.046	0.002	0.309	0.0007
Th208	1.072	0.002	0.309	0.0006
Ac228.	1.115	0.034	0.310	0.0105
Ac228.	1.121	0.005	0.310	0.0014
Ac228.	1.158	0.002	0.309	0.0006
Ac228.	1.168	0.320	0.309	0.0989
Ac228.	1.193	0.002	0.309	0.0005
Th208	1.284	0.084	0.309	0.0258
Th208	1.517	0.082	0.309	0.0253
Bi212	1.519	0.051	0.309	0.0158
Ac228.	1.618	0.001	0.308	0.0003
Ac228.	1.741	0.120	0.307	0.0368
Th208	1.794	0.177	0.306	0.0543

Ac228.	2.079	0.080	0.302	0.0242
Bi212	2.246	0.310	0.302	0.0935

Thorium series sensitivity, cpm/dpm Th232 **0.7055**