LONG
ISLAND
POWER
AUTHORITY

Shoreham Decommissioning Project

Termination Survey Plan

Revision 3

Prepared by
Termination Survey Section
Radiological Controls Division
Operations and Maintenance Department

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Long Island Power Authority

Shoreham Decommissioning Project

Termination Survey Plan
Revision 3

Reviewed by: Frederick Petrolaro 7/21/94
Signature Date
LIPA Site Review Committee

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Signature Date
LIPA SNPS Resident Manager

Preface

This document describes the methods used by the Long Island Power Authority (LIPA) to demonstrate that radiation and radioactive contamination levels of the Shoreham Nuclear Power Station have been reduced to levels below criteria established for release for unrestricted use. It supplements and updates the description of the proposed final radiation survey presented in the Shoreham Decommissioning Plan (LIPA90).

This Plan describes the technical methods to be used and provides guidance for planning and design of the Termination Survey. It is prepared and controlled under the LIPA Nuclear Management Control Manual, Termination Survey Program Description (LIPA92).[1] The methods described are derived from regulatory guidance, specifically Regulatory Guide 1.86 (USAEC74) and draft NUREG/CR-5849 (BE92); and from recent U.S. reactor facility decommissioning experience (Pathfinder, Saxton, Shippingport, UC Berkeley), taking into account conditions at the Shoreham facility.

Revision 0 of this Plan was issued November 20, 1992 and was used for early survey work beginning in January, 1993. Revision 1, issued in April 1993, incorporated changes necessitated by initial survey experience as well as commitments made to the Nuclear Regulatory Commission (NRC) resulting from staff review of the Plan. Revision 2, issued in December, 1993 corrected typographical errors, and provided updated information on: instruments used, the number of survey units, the final report outline and other minor editing changes. The current revision, Revision 3 has been issued to incorporate modifications to the Shoreham release criteria which account for the potential presence of Tritium and Iron-55 in areas where residual contamination from neutron actvated materials of construction may be present (USNRC94). The revision describes the use of adjustment factors for surface activity measurements to account for Tritium and Iron-55 for comparison of measurement results to release criteria guideline values. It also updates descriptions of instrumentation to add new detector assemblies developed for survey of embedded piping and updates the Shoreham Termination Survey Classification Summary to account for changes in the survey unit listing.

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[1] This Plan is a companion document to the Shoreham Decommissioning Project Termination Survey Program Description (LIPA92). The Program Description describes the organization and management responsibilities for the Termination Survey Program.

SHOREHAM DECOMMISSIONING PROJECT
TERMINATION SURVEY PLAN

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1.0 Historical Background Information

The Shoreham Nuclear Power Station (SNPS) power generation plant consists of a boiling water reactor (BWR) nuclear steam supply system (NSSS) and a turbine generator both furnished by General Electric Company. The balance of the plant was designed by Stone & Webster Engineering Corporation. The plant was designed to provide a gross electrical output of 849 Megawatts (LILCO90).

The SNPS achieved initial criticality in February 1985 following receipt of the initial operating license from the Nuclear Regulatory Commission (NRC). A license to operate at power levels not to exceed 5% of full power was granted and low power testing commenced in July 1985. The plant was operated intermittently at power levels not exceeding 5% of full power until the final critical operation in January 1989. This operating history corresponds to 2.03 effective full power days (EFPD) of average fuel exposure (LIPA90).

Pursuant to the 1989 agreement among The Long Island Lighting Company (LILCO), the State of New York and the Long Island Power Authority (LIPA), power generating operations at (SNPS) were terminated. The irradiated fuel was removed from the reactor vessel pressure in August 1989 and placed in the spent fuel pool.

The DECON alternative was selected for decommissioning SNPS and an order approving the LIPA Decommissioning Plan was issued by the US Nuclear Regulatory Commission (NRC) in June, 1992 (USNRC92). The objective of the approved DECON alternative was to decontaminate the SNPS facility and site and release them for unrestricted use. To accomplish this, the majority of the radioactive portions of the reactor pressure vessel and pressure vessel internals were disassembled, segmented and removed. Contaminated and activated portions of plant piping systems and equipment were decontaminated or removed as described in the LIPA Decommissioning Plan (LIPA90), including subsequently approved changes thereto.

2.0 Site Information

2.1 Site Description

The Shoreham Nuclear Power Station site is located in the Town of Brookhaven, Suffolk County, New York on the north shore of Long Island. The site is 50 miles east of the confluence of the East River and Long Island Sound, near La Guardia Airport.

The developed portion of the site comprises 80 acres, and is located within a larger parcel of 499 acres owned by the Long Island Lighting Company (LILCO).[1] The 499 acre parcel is bounded on the north by Long Island Sound and on the east by the Wading River marshland. It is bounded on the west by a parcel of approximately 429 acres known as the Shoreham West property, also owned by LILCO, and on the south by highway Route 25A. The 499 acre SNPS site property is divided across its midsection in the east-west direction by North Country Road which branches off Route 25A about three miles west of the site and rejoins 25A about three miles east of the site. North Country Road is about 1,500 ft. south of the Reactor Building at its closest point to the developed area of the site. Figure 2.1 shows the SNPS site plan and the location of major buildings on the developed portion of the site.

The site elevation varies from sea level at Long Island sound (the northern boundary of the site) to elevation 200 feet midway between North Country Road and the southern border of the site. Except for the developed area, the site is wooded with wetlands along the east and west boundaries extending as much as 1,300 feet inland from the Sound. The developed portion of the site is fairly level with the exception of several graded slopes, the largest of which is a terraced slope about 30 feet in height which traverses the site in an east-west direction immediately to the south of the Reactor Building. The ground surface covering in the developed area is mostly gravel with smaller portions devoted to lawn and paved areas (sidewalks, loading areas and roadways). The site soil cover in unpaved or undisturbed areas is a mixture of sand and glacial till (gravel). Vegetation cover in undisturbed areas is a mixture of grass and weeds with a few shrubs and small trees.

2.2 Site Conditions for Termination Survey

The Shoreham facility remains largely intact following decommissioning. Dismantlement of structures was confined to small portions within the Reactor, Radwaste and Turbine Buildings. Removal of such structures was performed to provide paths for removal of contaminated piping and equipment. As described in the LIPA Decommissioning Plan (LIPA90), most radioactive piping and equipment was dismantled, removed from the facility and disposed of as radioactive waste at a licensed radioactive waste disposal

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[1] Under the Asset Transfer Agreement, approximately 11 acres of the site which include the power block, adjacent office and support buildings and connecting roadways have been transferred to LIPA for conduct of the decommissioning (LIPA90).

facility. This includes the reactor pressure vessel, reactor pressure vessel internals and major portions of 14 plant systems. Decontamination of plant piping and equipment was performed to a much lesser extent. Equipment items which remain and were decontaminated in place include the reactor vessel bottom head, a portion of the main steam lines in the Reactor Building, the Condensate Storage Tank (CST), and Residual Heat Removal (RHR) system heat exchangers (shell only). It is estimated that greater than 75 percent of the piping and equipment on site during the time of reactor operation will remain after decommissioning is complete.

2.3 Site Areas Covered - Scope of Survey

The Shoreham facility and environs have been evaluated to identify the areas to be covered in the termination survey. The termination survey focuses on the area within the Secured Area fence as shown in Figure 2.1. The area (approximately 20 acres) contains the Reactor, Radwaste and Turbine buildings, and other buildings, facilities, and grounds within the Secured Area fence. The Secured Area is described in the Shoreham Nuclear Power Station Updated Safety Analysis Report (LILCO90). The area within the Secured Area fence coincides with the Restricted Area defined in Shoreham radiological control procedures as the area where access has been controlled and radioactive materials controlled for purposes of protection of individuals from exposure to radiation. Following completion of Shoreham decommissioning and removal of the irradiated fuel from the site, areas outside the Restricted Area may be added to the survey if used for temporary storage or handling of irradiated fuel or other radioactive materials.

Major attention in the termination survey is given to the areas most affected by reactor operations and by decommissioning activities. The area covered by the termination survey has been divided into approximately 390 individual "survey units" for management of the survey. Each survey unit is classified as "affected" or "unaffected" for survey implementation.[2] Affected areas are largely confined to the Reactor and Radwaste buildings, and portions of the Turbine building. Table 2.1, Shoreham Termination Survey Classification Summary, summarizes the breakdown of the facility into affected and unaffected areas. This table is updated from time to time, as needed, and maintained via an approved procedure.

The environs of the facility beyond the area of the site encompassed by the termination survey have been demonstrated to be free of detectable radioactivity from Shoreham operations. This is well documented by the Shoreham Radiological Environmental Monitoring Program (REMP).

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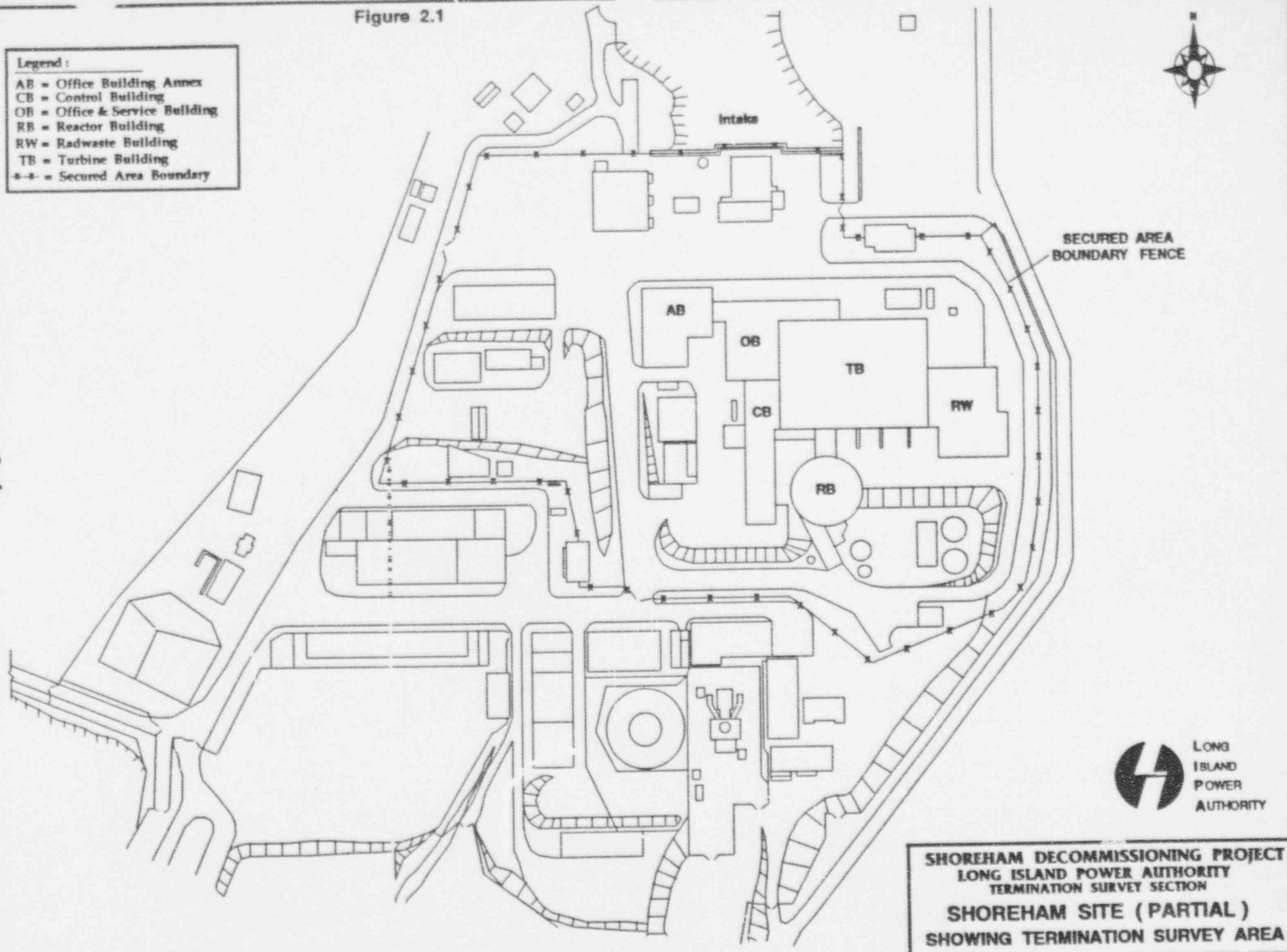
[2] An "affected" area as defined in draft NUREG/CR-5849 (BE92) and used in this Plan is a designation used to indicate that an area (survey unit) has a potential for containing residual radioactive contamination. An "unaffected" area is one which is not expected to contain residual radioactivity based upon the operating history and previous radiological surveys (see Glossary).

Table 2.1

Shoreham Termination Survey Classification Summary

DESCRIPTION	CODE	Total No. of Survey Units	No. of Affected Survey Units	No. of Unaffected Survey Units
STRUCTURES				
Reactor Bldg.	RB	85	84	1
Drywell	PC	14	14	
Suppression Pool	SP	5	5	
Turbine Bldg.	TB	106	8	98
Radwaste Bldg	RW	50	48	2
Control Bldg	CB	4		4
O & S Bldg	OB	4		4
O & S Bldg Annex	AB	4		4
Other Site Bldgs	OS	9	1	8
Structure Totals		281	160	121
OUTSIDE AREAS				
Site Grounds	SG	7		7
Soil Samples	SS	1		1
Structure Exteriors	SE	14		14
Outside Area Totals		22	0	22
PLANT SYSTEMS	SU	82	38	44
TOTALS		385	198	187

Figure 2.1


Legend :
AB = Office Building Annex
CB = Control Building
OB = Office & Service Building
RB = Reactor Building
RW = Radwaste Building
TB = Turbine Building
Secured Area Boundary
Intake
SECURED AREA
BOUNDARY FENCE
AB
OB
TB
RW
CB
RB
N
LONG
ISLAND
POWER
AUTHORITY
SHOREHAM DECOMMISSIONING PROJECT
LONG ISLAND POWER AUTHORITY
TERMINATION SURVEY SECTION
SHOREHAM SITE (PARTIAL)
SHOWING TERMINATION SURVEY AREA
Drawn by: R.F.Y.
Date: 11/9/98
Drawing No.: TSP_001
Figure 2.1

3.0 Termination Survey Overview

3.1 Survey Objectives

The termination survey is designed to demonstrate that licensed radioactive materials have been removed such that residual levels of radioactive contamination are below applicable Regulatory Guide 1.86 limits (USAEC74). Radiation detection instrumentation requirements for the survey are based upon the conclusion that the controlling radioactive species in determining compliance with Regulatory Guide 1.86 release limits are activation products dominated by Co-60. These are beta-gamma emitters as defined in Regulatory Guide 1.86 (USAEC74), hence the limits for beta-gamma emitters shown in Table 3.1, below, apply. Special limits have been established for the hard-to-measure radionuclides Fe-55 and Tritium in areas where residual contamination from neutron activated materials of construction may be present (USNRC94). The release criteria limits for Shoreham are shown in Table 3.1.

The applicable release limits for alpha emitters are also shown in Table 3.1.[1] These are for Natural Uranium, U-235, U-238 and associated decay products. Instruments and methods are being incorporated into the survey which are adequate to measure alpha surface activity at levels below the limits in Table 3.1.

Table 3.1

Acceptable Surface Contamination Levels
(dpm per 100 cm^2)

	Fixed Plus Removable		Removable
Activity	Average	Maximum	
Beta-Gamma	5,000	15,000	1,000
Alpha	5,000	15,000	1,000
Fe-55, Tritium	200,000	600,000	1,000

The release criteria for Shoreham also include guideline values for gamma exposure rate and soil radioactivity concentration. The average guideline value for gamma exposure rate is 5 μR/hr above background, measured at one meter from accessible surfaces in the facility buildings and outdoor areas (LIPA90). The average is calculated over an area not to exceed 10 m^2. In addition, any individual gamma exposure rate measurement shall not exceed 10 μR/hr above background, the elevated area guideline. The guideline value for soil and bulk materials radioactivity concentration at Shoreham is 8 pCi/gm. Bulk materials include activated concrete, sewage sludge, tank bottoms and sediments, radwaste treatment media (e.g. charcoal

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[1] The various release limits are also referred to as guideline values (Be92). See Section 8.0, Glossary, for definitions of guideline value and elevated area guideline.

beds) and any other materials not readily evaluated by direct measurements. No elevated area guideline has been established for soil and bulk materials.

3.2 Identity of Contaminants

The Shoreham Characterization Study (LILCO90a) reported a total facility radioactivity inventory due to Shoreham operations of about 600 Curies (Ci) (not including the irradiated fuel, control blades and readily removable reactor asssembly items). All but a very small fraction (less than one Ci) of this inventory was contained in activated materials of the reactor pressure vessel and vessel internals which have been removed from the site. The calculated radioactivity composition of the activated components (as of July, 1990) is: Fe-55, 69%; Co-60, 28%; and Ni-63, 2%. Minor amounts of other species are calculated to be present, including: H-3, C-14, and Ni-59 (LILCO90a). Laboratory analysis of activated reactor pressure vessel and vessel internals samples indicates the presence of low levels of Mn-54, Zn-65, and Ni-63, in addition to Co-60. Analysis of bioshield wall samples show low levels of Co-60, Mn-54 and Eu-152 (TU92).

In most areas of the facility, the most likely source of residual contamination is activated corrosion products (crud) deposited in piping systems which could have been transported into the facility via leaks or handling of contaminated equipment. Detectable radionuclides in surface corrosion deposits are confined to a small number of radionuclides. An analysis of piping system corrosion products in nine (9) discrete samples shows that these are comprised largely of Co-60 and Fe-55, with a "best fit" (by least squares regression) Fe-55:Co-60 ratio of approximately 0.30 ($\pm$ 0.13) (TU92). The "average" value of the individual Fe-55:Co-60 ratios from these same 9 samples is 0.30. Accounting for isotopic decay from the time of core operation until July 1994, the average value becomes 0.19. This last number is used to adjust individual measurements, as described in Section 5.0, to account for the Fe-55 not detected by survey instruments. Laboratory analysis of plant corrosion product deposits has not identified the presence of reactor-produced alpha emitters above lower limits of detection.

Several areas have been identified where residual contamination sources exhibit composition characteristics which differ from surface corrosion product deposits described above. The Reactor Vessel Bottom Head, the Spent Fuel Pool, the Reactor Biological Shield wall remnants and areas where Reactor Vessel and Vessel internals were cut up and dismantled, may contain significant fractions of hard-to-measure (HTM) radionuclides such as Fe-55 which decays by electron capture, and Tritium which is a low energy beta emitter.

The radioactivity composition for surface contamination measurements in all areas in the termination survey is given in Table 3.2. These results (except for the Spent Fuel Pool) are derived from neutron activitation calculations reported in the Shoreham Site Characterization report (LILCO09a) adjusted for radioactive decay to July 1, 1994. The Spent Fuel Pool composition is obtained from radiochemical analysis of a spent fuel pool cleanup system filter (SCI94). Table 3.2 also contains the adjustment factors, f which are applied to direct and removable surface activity measurements. Application of adjustment factors is described in Section 5.0.

Table 3.2

Radioactivity Composition for Surface Activity Measurements

Area - Description	Nuclides	Fraction of Total Activity	Adjustment Factor
Reactor Bioshield Concrete	Co-60 Fe-55 H-3	0.0278 0.3159 0.6559	35.9
Reactor Bioshield Concrete removable activity	Co-60 Fe-55	0.0809 0.9181	12.4
Reactor Bioshield Steel	Co-60 Fe-55	0.0413 0.9487	24.2
Reactor Bioshield Steel removable activity	Co-60 Fe-55	0.830 0.170	1.2
Dry Cutting Station (total and removable activity)	Co-60 Fe-55 Ni-63	0.3840 0.5635 0.0517	2.6
Spent Fuel Pool (total and removable activity)	Co-60 Fe-55 Ni-63	0.4324 0.4764 0.0868	2.3
Reactor Vessel Bottom Head	Co-60 Fe-55 Ni-63	0.3858 0.5617 0.0521	2.6
All Other Areas	Co-60 Fe-55	0.830 0.170	1.2

3.3 Organization and Responsibilities

An organization, identified as the Termination Survey Section of the Radiological Controls Division, has been created within the LIPA - Shoreham Decommissioning Project organization for planning and implementation of the termination survey. The organization and responsibilities of the Termination Survey Section and the interfaces and responsibilities for all other elements of the Decommissioning Project Organization for the termination survey are described in the Termination Survey Program Description (LIPA92).

3.4 Training

3.4.1 Technician Qualification

A job qualification category for termination survey technicians will be established under the existing Shoreham Station "Health Physics Technician Selection, Training and Qualification Program" procedure. The training and qualification acceptance criteria from this procedure will be adopted. The training will include classroom and functional job performance training on termination survey procedures and specialized instrumentation. The training and qualification process for individual technicians will require from three to five days to complete. Termination survey technicians will generally be selected from the pool of technicians who have previously been qualified to perform HP responsibilities on the Shoreham Decommissioning Project. When new technicians are hired directly to support the survey, their training will include in addition to termination survey training, those portions of the overall site Health Physics Technician training necessary to ensure proper job performance.

3.4.2 Classroom Training

Classroom training includes: an overview of the Termination Survey Program, instrumentation, and procedures. The overview will cover termination survey objectives, survey methods, the role and responsibilities of termination survey technicians, the importance of personnel safety, termination survey quality assurance, the Termination Survey Plan and program implementation.

3.4.3 Functional Training

Functional training will involve hands-on performance of principal HP technician termination survey tasks. A survey of a typical structural and system survey unit will be performed by each technician under the surveillance of a qualified instructor.

3.5 Laboratory Services

Laboratory radioanalytical services of the Shoreham Nuclear Power Station Radiochemistry Section and/or Health Physics Section, both within the Radiological Controls Division will be used in support of the termination survey. The on-site capabilities include gamma spectroscopy (GeLi) of filters, smears and bulk samples; liquid scintillation; gas proportional counting; gross beta-gamma counting of smears; and gross alpha counting of smears. Both Sections operate under approved QA programs and procedures. Additionally, a contract is in place with a qualified vendor for specialized radiological analysis of samples on an as-needed basis. Vendors are selected in accordance with the requirements of the LIPA QA Manual, Appendix N (LIPA92a).

3.6 General Survey Plan

The termination survey is implemented at the individual survey unit level. Three categories or types of survey units have been established: 1) structures, which include building interiors, 2) plant systems and 3) outdoor areas. These categories combine survey units into groups with similar physical characteristics. The survey is planned for measurements to be taken for each survey unit independently. The measurement intensity of each survey unit is based upon its classification as affected or unaffected.

Due to the large scope of the termination survey and the requirement that some survey activities be conducted in parallel with decommissioning work, a systematic approach is necessary. Further, it is essential that key interfaces between survey activities and other decommissioning work activities be identified.

The termination survey planning and implementation process for each survey unit involves the following steps: 1) initial classification; 2) history file preparation and classification review; 3) turnover for termination survey; 4) walkdown; 5) survey design; 6) preparation of modification packages (primarily for plant systems surveys); 7) preparation of work requests and scheduling; 8) preparation of final survey instructions; 9) physical support activities such as erection of scaffolding, system tagout and system breaches for system surveys; 10) performance of the survey; and 11) post survey restoration and control of surveyed areas (isolation of systems after survey completion). These are described in the following paragraphs.

3.6.1 Initial Classification

The classification of the facility into "affected" and "unaffected" areas provides the planning basis for the termination survey. It was conducted using results from the Shoreham Site Characterization Program (LILCO90a) and the recommendations of experienced Shoreham personnel using the classification criteria contained in Appendix A, Survey Design Guidelines. The classification status of all survey units is maintained in the Termination Survey Classification Description, and is controlled by a procedure of the same title.

3.6.2 History File Preparation

The history file is a compilation, in a standardized format, which summarizes the operational and radiological history of each survey unit included in the termination survey. Preparation of the history file involves review of the system description (for system survey units), plant operating records, the Shoreham Characterization Report (LILCO90a), radiological surveys and other relevant information. Specific operating history which could affect the radiological status is sought in this review. The purpose of this process is to provide a substantive basis for the survey unit classification, and hence the level of intensity of the termination survey.

The history file contains a summary description of the survey unit and, for system history files, summarizes relevant operational data. Relevant operational data includes operating lineups to radioactive systems, and other events which could affect the radiological status. Similarly, for structures and outdoor areas, the extent of radioactive materials involvement in the area (if any) is summarized.

Possible outcomes of the review are: the initial classification is verified, or it is modified to reflect the more thorough evaluation. Additionally, the review may suggest specific areas of a structure or components of a system which should be highlighted in the final survey. The review and conclusions are documented in a checklist. The history file also identifies the sources of information used. It may contain summaries, excerpts or complete documents which are useful for survey design. The system history file and its preparation are described in a Termination Survey procedure.

3.6.3 Turnover for Termination Survey

Prior to acceptance of a survey unit (structure, system or outdoor area) for the termination survey, a number of conditions must be satisfied. Decommissioning activities are completed, all tools are removed, housekeeping and area cleanup is completed, decontamination of affected structural areas and system residual components is completed and verified by operational radiological surveys, and scaffolding needed to be left in place for termination survey is identified. Radiological surveys verifying the status of the area, if a structure, and remaining system components, if a system, are provided to the Termination Survey Section. Turnover and control of systems, structures and outside areas is controlled by a Termination Survey procedure.

3.6.4 Walkdown

The walkdown is a key activity in the preparation of the survey design. For systems, it includes review of system flow diagrams and piping drawings, and physical walkdown of the system. Structures and outdoor areas are also physically walked down. A principal objective is to assess the physical scope of the survey unit and to identify potential breakdown into subunits. Special access needs are identified. Potential support requirements for conduct of surveys are identified, such as scaffolding, component disassembly, interference removal, engineering modifications, electrical tagout and system alignment to provide access for surveys. Safety concerns, such as access to confined spaces, high walls, and ceilings, are identified and resolved. It is noted that for survey units involved with decommissioning activities, the walkdown is best completed when the final configuration is known, usually near or after the completion of decommissioning work. Early information is available through Decommissioning work packages and material takeoff lists and drawings.

3.6.5 Survey Design

The survey design results in the identification of the locations (grid blocks, system components) to be surveyed and the type of radiological measurement or sample to be collected at each location. The design is based upon the guidelines in Appendix A. The design of the survey for individual survey units is prepared in accordance with a Termination Survey procedure.

3.6.6 Engineering

Once the survey design is prepared, engineering review and support requirements are developed. In the rare instances where engineering modifications are required for surveys, the modification packages are prepared and the design review implemented through approved engineering procedures.

3.6.7 Work Planning and Scheduling

Upon completion of any required engineering review, the physical modifications are specified. Field work is implemented via the Maintenance Work Request (MWR) process. The MWRs identify all components which require opening, identify all modifications, indicate restoration requirements and indicate whether a system is to be isolated or returned to service. The MWR process is also used to initiate support work and tagouts necessary for surveys of structural and outdoor survey units. The survey unit support work is then placed upon the Project work schedule for performance.

3.6.8 Survey Instructions

The survey instructions are provided to the Lead HP technician assigned responsibility for the specified survey unit. They specify the number and type of radiological measurements to be taken at each location or component identified in the survey design. The instructions identify smear samples and other samples to be collected. The survey instructions identify those survey points (components or other specified locations) where QC verification surveys are required. The survey instructions are prepared by the designated Termination Survey Radiological Engineer in accordance with a Termination Survey procedure.

3.6.9 Field Support

The MWR identifies each component or survey location requiring support work and tagouts. In cases where special surveys are required such as components, embedded piping, or large tanks which are classified as affected, other preparation work may be required. This may include gridding of large tanks once access is provided and safety precautions have been satisfied.

3.6.10 Survey Measurements

Termination survey measurements are conducted in accordance with Termination Survey procedures and the specific survey instructions for the survey unit. The measurements include surface scans, direct measurements of surface contamination, smear samples for removable surface contamination and gamma exposure rate measurements. The measurements are discussed in Section 4.0. Prior to conducting the survey, a walkdown is conducted by the cognizant Termination Survey Radiological Engineer and the Lead HP technician to verify the survey locations and the details of the instructions.

3.6.11 Data Management and Evaluation

Upon completion of field measurements and sample collection (smears, soil and sediment as applicable), all measurement data and sample counting results for each survey unit are reviewed for completeness. The survey measurement data is then entered into a custom database designed to store measurement raw data, perform calculations necessary to convert individual measurements to reporting units, calculate summary statistics and generate data reports for each survey unit. The calculations and evaluations performed are described in detail in Section 5.0, Data Interpretation.

3.6.12 Reporting Results - Release Records

The results of the survey of each survey unit are reported individually via a document called a "release record". Each release record consists of a written summary which describes the survey unit and presents the comparison of the survey results to release criteria guideline values. Attached to the text are data reports which include each measurement result as well as the calculated summary statistics for the survey unit. Maps are also included with each release record to identify the location of survey measurements. In rare cases, a release record may cover a partial survey unit. e.g. those portions of the Radioactive Waste system within the Turbine Building. Also, where several survey units are of similar composition or are very closely related functionally, they may be combined into a single release record for simplification of the reporting process.

3.6.13 Restoration and Isolation

a. Systems

After survey measurements have been taken, reviewed and approved, and QC verification survey measurements have been completed in system survey units, the system is restored and components are replaced as specified in the MWR. If indicated in the MWR, the system is isolated to protect against recontamination. Isolation and

control of plant systems after termination survey is performed under a specific approved procedure.

It is noted that many plant support and service systems will be returned to service after completion of termination survey measurements. Examples are: compressed air, heating and cooling, ventilation and fire protection. When a system is required to remain in service, administrative controls will be used to minimize the possibility of system contamination. These include, but are not limited to, surveillance activities to ensure that the system is not aligned or operated in a manner which could compromise termination survey results.

b. Structures and Outdoor Areas

After measurements are completed in structures and outdoor areas, appropriate controls are used to prevent or minimize possible contamination. This is controlled by an approved station procedure. Numerous structural survey units have been defined within the Radiological Controlled Area (RCA) of the Shoreham facility. These include all of the structural survey units which comprise the Reactor Building, the Turbine Building and the Radwaste Building, representing the majority of the areas within the power block. As sufficient numbers of surveys are completed in contiguous RCA survey units (with completion of the surveys being indicated by complete, approved release records) to allow manageable reduction of the RCA boundaries, these reductions will be made. Removal of such areas from the RCA provides additional assurance that material containing radioactivity is not used or transported through previously surveyed areas.

3.7 Quality Assurance

3.7.1 General Provisions

As indicated in the LIPA Decommissioning Plan and in the Termination Survey Program, quality assurance for the termination survey is subject to the provisions of the Decommissioning Program Quality Assurance Manual (LIPA92a). In addition, the Termination Survey Program itself has established quality control measures as an integral part of the program. Principal measures established to meet quality objectives are:

a. Selection and Training of Personnel

Qualification requirements and responsibilities are established for key personnel performing termination survey tasks. A technician training

and qualification program has been established which includes classroom training and job-functional training. Training and qualification records are maintained on all technicians selected for the termination survey.

b. Instrumentation Selection, Calibration and Operation

An evaluation and testing program was conducted to select radiation detection instrumentation for the survey. Instrument calibration is performed either under approved SNPS calibration procedures using calibration sources traceable to the National Institute of Standards and Technology (NIST), or by qualified vendors with the results traceable to NIST. Measurements are performed using approved written procedures for each instrument. Control of instruments is established by an instrument control procedure.

c. Survey Documentation

Each termination survey measurement is identified by date, instrument, technician, location, type of measurement, and mode of instrument operation.

d. Quality Control - Verification

Replicate measurements are performed independently on a selected sample of survey measurements on an ongoing basis.

e. Written Procedures

All termination survey tasks which are essential to survey data quality are controlled by procedures reviewed and approved by the LIPA Site Review Committee.

f. Mockup of Procedures and Processes

Dry runs and mockups are performed to test principal procedures and methods prior to implementation in the field.

g. Chain of Custody

Written procedures establish responsibility for custody of samples and survey data between the point of measurement or collection until final results are obtained.

h. Records Management

Generation, handling and storage of termination survey design and data packages is controlled by an approved procedure.

i. Data Management Software

Computer programs generated for processing of survey measurement data shall be tested and verified.

j. Independent Review of Survey Results

The release record of each survey unit is given independent review prior to acceptance for final management approval.

k. Control of Surveyed Areas and Systems

Administrative (i.e., procedural) and physical controls are established on areas and systems to minimize the possibility of contamination subsequent to the survey.

l. Control of Vendor Supplied Services

Essential services, such as instrument calibration and laboratory sample analysis, will be procured only from qualified vendors, in accordance with an approved procedure whose internal QA programs are subject to LIPA audit.

3.7.2 Termination Survey Quality Control Procedure

A Termination Survey Quality Control procedure controls essential quality assurance activities not addressed in other procedures. These include:

a. conduct of QC replicate sampling measurements,

b. routine verification of survey measurement data accuracy,

c. control charts for individual instruments,

d. testing of computer data calculation programs,

e. validation of operational survey data used as final survey data,

f. documentation of surveys, and

g. custody of instruments, samples and measurement data.

3.8 Schedule

The termination survey detailed schedule is maintained within the Project Schedule by the Project Controls Division of the Finance and Administration Department and the Work Planning Section of the Operations and Maintenance Department.

The termination survey is scheduled to be completed in several major phases which encompass distinct portions of the facility. The initial phase included only the survey of the SNPS main turbine, and was reported in February, 1993. The first major phase which included the outside area survey units, structural survey units outside the power block, and most systems and structural survey units within the Turbine Building, was reported in September 1993. The second phase includes systems and structural survey units within the Reactor Building Primary Containment and Suppression Pool, and was reported in February 1994. The third phase includes systems and structural survey units primarily within the Radwaste Building, and was reported in June 1994. The final phase includes all remaining survey units, the majority of which are those areas impacted by the removal of irradiated fuel from the facility, primarily the Reactor Building refuel floor and other areas not reported earlier. The final phase is scheduled to be reported in October 1994. Upon completion of each phase the release records will be compiled and the survey units covered will be available for NRC verification surveys.

3.9 Survey Report

The Final Report will be prepared for submission to the Nuclear Regulatory Commission to meet the intent of Regulatory Guide 1.86 (USAEC74) for final survey reporting. The Final Report will follow the guidance of Draft NUREG/CR-5849 (BE92) regarding content. The Final Report will be submitted in stages with an updated report prepared for each of the major survey phases discussed in Section 3.8. Each edition of the Final Report will include the survey measurements obtained during the appropriate phase and evaluation of the results. An update to the overall survey status report as well as other documents which comprise the Final Report will also be provided. The full outline of the Final Report is described in Section 6.0.

4.0 Survey Plan and Procedures

4.1 General

The design approach of the Shoreham termination survey is considerably affected by the final configuration of the facility, which is largely intact with the majority of equipment left in place. The Reactor, Turbine and Radwaste Buildings contain over 200 equipment rooms. These rooms contain approximately 80 plant piping systems and the majority of the systems occupy multiple rooms.

The majority of the survey effort is confined to the areas contained within the Reactor and Radwaste Buildings and those areas in the Turbine Building where radioactive materials were handled. These are classified as affected areas. The remainder of the areas within the scope of the survey are classified as unaffected. All radioactive material handling, movement and storage on the site has been controlled under approved procedures. No detectable activity of SNPS origin has been detected on the site grounds or environs following extensive measurements of site soil and outdoor surfaces in the Site Characterization Program (LILCO90a) and in the REMP program prior to, during, and subsequent to Shoreham operation. [1] The survey plan and procedures are designed accordingly, to focus primarily on remaining plant structures and systems in the affected areas. Instrumentation has been selected and measurement procedures developed to detect and measure surface contamination levels (primarily Co-60) and gamma exposure levels in these affected areas.

4.2 Instrumentation

Radiation detection and measurement instrumentation for the termination survey has been selected to provide reliable operation with adequate sensitivity to demonstrate attainment of the release criteria. An evaluation has been conducted of instruments and detectors produced by several manufacturers. Detectors have been selected based upon detection sensitivity, operating characteristics and expected performance in the field under conditions of use. The detectors selected and their detection characteristics are summarized in this Section. Recording instruments (survey meters) for use with these detectors have also been evaluated. Instrumentation to be used for gamma exposure rate measurements and special purpose measurements is also described.

4.2.1 Instrument Description

The principal instruments selected for termination survey measurements are identified in Table 4.1, Termination Survey Instrument Summary. The detectors used for total surface contamination monitoring are for the most part operated with data logging survey meters.

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[1] Trace amounts of Co-60, on the order of 0.1 pCi/gm have been detected in a sanitary sewage septic field distribution tank located on the owner controlled area of the site in 1993. This is above the Shoreham Station Radiological Environmental Monitoring Program (REMP) lower limit of detectability (LLD) for this isotope .

Table 4.1

Termination Survey Instrument Summary

Information Withheld Per §2.790(a)(4)

4.2.2 Detection Sensitivity

The detection sensitivity of the detectors selected for termination survey measurements has been evaluated. These results are summarized in Table 4.2, Measurement Detection Sensitivities. Special detector configurations have been developed for termination survey of piping which is not readily accessible using conventional survey techniques. Table 4.3 summarizes the detector assemblies developed for survey of various sized piping. Most of these are multiple-GM detector assemblies.

For count rate measurements, the minimum detectable activities (MDA) shown in Tables 4.2 and 4.3 are calculated using the following equation:

(4.1)
$$MDA_{(C.L.)} = \frac{(k_a + k_b)\sqrt{\frac{S}{t_s} + \frac{B}{t_b}}}{E\left(\frac{A}{100}\right)}$$

where:

k_a = critical value at the upper, one-sided (100 - a)% confidence level (C.L.) (for normal statistics, k_a = 1.645 at 95% C.L.)

k_b = critical value at the lower, one-sided (100 - b)% confidence level (C.L.) (for normal statistics, k_b = 1.645 at 95% C.L.)

S = Sample (plus background) count rate (cpm),

B = Background count rate (cpm),

t_s = Sample count time (minutes),

t_b = Background count time (minutes), and

E = Instrument detection efficiency, counts per disintegration, and

A = Detector sensitive area (cm^2).

The MDAs are calculated using the above Equation 4.1 with the indicated values of k for the 95% confidence level and assuming the probability of Type 1 and Type 2 errors to be equal. These MDAs represent detection sensitivities under static conditions, one minute counts with fixed geometries. Detector efficiencies are empirically determined using sources traceable to the National Institute of Standards and Technology (NIST). Beta-Gamma detectors are calibrated with Co-60 and alpha detectors with Pu-239. Nominal background values are used for the calculation, with 30 minute background count times. During scanning type surveys, the detection sensitivity may be reduced. Therefore threshold or practical detection

Table 4.2

Measurement Detection Sensitivities

Information Withheld Per §2.790(a)(4)

Table 4.3

Embedded Piping Detector Assembly Sensitivities

Piping Dia. nominal (ID, in)	Detector Assembly (eff. area) [1]	Bkgnd [2]	Efficiency 4π [3] (area adjusted) [4]	Detection Sensitivity (dpm/100 cm^2)
12	4 - FT126 GMs (504 cm^2)	350 cpm	0.058 (0.295)	210
10	8 - HP260 GMs (124 cm^2)	165 cpm	0.166 (0.206)	210
8	6 - HP260 GMs (93 cm^2)	130 cpm	0.152 (0.141)	270
6	9 - HP260 GMs (140 cm^2)	210 cpm	0.148 (0.207)	230
4	6 - HP260 GMs (93 cm^2)	165 cpm	0.134 (0.124)	350
3	4 - HP260 GMs (62 cm^2)	80 cpm	0.156 (0.097)	310
2	4 - TGM N1003 GMs (27.2 cm^2)	19	0.168 (0.046)	520
1.5	1 - HP190A (end window)	17 cpm	0.005 [5]	2800

Notes. 1. Effective sensitive area is the total sensitive area of all detectors in the assembly. Survey measurements are normally performed by summing the counts from all detectors. The assemblies can also be operated whereby the counts from each detector are recorded separately.

2. Nominal background values; 30 minute counts.

3. The 4π efficiency is obtained from the combined detector response to a Co-60 flexible mylar source in an annular sleeve placed inside a piping spool piece. The source active area is such that it subtends the active area of all detectors in the detector assembly.

4. The area adjusted efficiency is the product of the 4π efficiency and the ratio A/100, where A is the effective sensitive area of the detector assembly.

5. Efficiency obtained by detector response to an annular sleeve source with an effective area of 100 cm^2.

limits have been empirically determined for each detector used in the scanning mode. Scanning mode efficiencies were obtained via a series of measurements whereby the detector was passed over calibrated sources of various dimensions at an established scanning speed to determine an effective scanning "efficiency". The scanning MDAs were then calculated using equation 4.1 with the scanning efficiency substituted for E, the efficiency term. These results are shown in Table 4.2 for the various detectors used for scan surveys.

4.2.3 Calibration and Maintenance

Instruments and detectors used in the Termination Survey are calibrated and maintained at Shoreham Nuclear Power Station according to approved procedures. Detectors for surface beta-gamma measurements are calibrated using NIST traceable Co-60 sources. Specialized instruments, for example, the pressurized ion chamber, are calibrated by the vendors.

4.3 Survey Plan

4.3.1 Classification

Each survey unit is classified into one of two categories which identifies each unit as "affected" or "unaffected". Units identified as affected have a possibility of containing residual contamination and those identified as unaffected have a very low probability of residual contamination. Classification of individual survey units is based on the site characterization study (LILCO90a) and the history of radioactive materials involvement or potential for contamination of the survey unit. Criteria for classification of survey units are given in Appendix A.

4.3.2 Reference Grids

Gridding consists of dividing areas to be surveyed into regular subdivisions for the purposes of identifying survey locations and for use as guides for scanning surveys. For surveys designed on the basis of prescribed sampling plans, whether random or systematic, gridding provides a means of selecting individual measurement locations. Mapping is used to document measurement locations. Detailed guidance for gridding is contained in Appendix A. Figure 4.1 shows grid placement in a structural survey unit classified as affected (an equipment room). Placement of grids in structures is directed by a specific work instruction.

4.3.3 Grid Maps

Grid maps are used for survey design and to document the measurement locations. Figure 4.2, Shoreham Decommissioning Termination Survey Grid Map, shows a grid map of a structural survey unit. It shows the floor and walls up to two meters. This map is identified as the base map for the survey unit. Additional grid maps may be prepared for subunits as needed to plan and document surveys of subunits.

4.3.4 Grid Numbering

Grids are uniquely identified by an ID code or number. The numbering convention is to start at the reference location, usually the southwest corner, and proceed sequentially west to east numbering each row on the floor from south to north in a continuing sequence. Grids are numbered in sequence for each subunit similarly as described for the floor numbering. An individual grid block has a unique identification code as determined by its survey unit (or subunit) ID and the number of the grid block within that unit. The numbering convention is illustrated in Figure 4.2.

4.3.5 Survey Maps

Survey maps are prepared to document the details of survey measurements in circumstances where grid maps are not practical. Survey maps are typically prepared to document detailed surveys of piping and system components.[2] These may be prepared by the technician who performs the survey, or may be specially prepared by direction of Termination Survey Radiological Engineers for detailed surveys of complex components.

4.3.6 Surface Scans

Surfaces are scanned according to the prescriptions in Appendix A. Scanning surveys are performed to screen large areas efficiently to search for areas above the average total surface contamination release criteria and to detect "hot spots", i.e., localized areas above the maximum total surface contamination release criteria. The scanning methods utilized (instrument and survey technique) are capable of detecting 75% of the average total surface contamination release criteria, e.g., 3750 dpm/100 cm^2 for total surface beta-gamma contamination, as shown in Table 4.2. When scanning surveys indicate that contamination levels above the average total surface contamination release criteria may be present, appropriate followup investigation and/or measurements will be performed.

4.3.7 Surface Activity Measurements

Surface activity measurements are taken at measurement locations selected in accordance with the survey design guidelines in Appendix A. The general set of measurements is direct beta-gamma and removable beta-gamma. In areas and systems identified as alpha affected, direct surface and removable surface alpha measurements are taken.

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[2] It should be recalled that system component and equipment exterior surfaces are included in the survey of the structural survey unit in which they reside. The component interior surfaces are surveyed in the survey of the system survey unit to which the component belongs. Survey maps of individual components may thus be prepared for each type of survey.

4.3.8 Exposure Rate Measurements

Gamma exposure rate measurements, when directed by survey design guidelines, are taken at one meter from surfaces at all measurement locations in structures and outdoor areas. In locations where it is not physically possible to locate an instrument one meter from the surface, gamma exposure rate measurements are not taken. The methods for gamma exposure rate measurements are discussed in Section 4.4.3 below.

4.3.9 Soil Sampling

Soil samples will be collected in accordance with Section 7.2.5 of Appendix A. Additional soil samples will be collected in the Termination Survey if a contamination event or spill occurs, or survey measurements indicate outdoor areas of elevated activity above applicable release criteria limits.

4.3.10 Special Sampling and Measurements

a. Sampling of Sediment and Loose Material

Samples of loose paint, dust or other sediment, tank bottoms, sewage sludge, radwaste media, concrete and other bulk materials are collected for laboratory analysis as part of biased sampling and measurements. Such samples may be collected in drain receptacles, sumps, and other catchments in affected areas. Selected storm drain catchments may be sampled in accessible locations on the site. These samples are analyzed by gamma spectroscopy for Co-60. Those samples with detectable activity are quantitatively analyzed and the results compared to the applicable guideline value for soil and bulk materials.

b. Embedded Piping Surveys

Measurements are taken to demonstrate that normally inaccessible piping, e.g., embedded piping, or runs of piping in confined pipe chases, is below the release limits for surface contamination. This technique involves the use of GM detectors, specially calibrated for direct surface measurements within piping interiors. The detectors are mounted in multiple-detector assemblies called "pipe crawlers" (see Section 8.0 Glossary). These assemblies are inserted into piping runs in a controlled manner and measurements of total surface activity are taken systematically over the length of piping.

4.4 Background Level Determination

4.4.1 General Requirements

Backgrounds are established for each type of instrument to be used for surface contamination and gamma exposure rate measurements. Surface contamination

measurements include total surface beta-gamma and alpha, and removable surface beta-gamma and alpha. Gamma exposure rate measurements require determination of the gamma background response of detectors at one meter from surfaces. The background responses of the pressurized ion chamber and microrem instruments must be determined. In addition, backgrounds are determined for specialized detectors and detector systems. These include: large area detectors for floor monitoring and detectors for surveying piping interiors (multiple GM detectors).

4.4.2 Objectives of Background Determinations

The objectives of background determinations for Shoreham Decommissioning Termination Survey measurements are to:

a. establish the reference background mean values for each type of detector used in the Survey;

b. assess the variability in background responses for principal detectors under different applications and conditions of use; and

c. determine the need for correction factors or special measurements to establish the background for Termination Survey measurements in specific locations.

4.4.3 Background Measurements

Several locations have been used to obtain measurement data for establishing reference backgrounds for each type of measurement. Principal criteria for background measurement locations are: similarity to Shoreham facility construction and free of non-natural radioactivity. One on-site building, the Colt Diesel Generator building, was selected due to its similarity in construction to the Reactor, Radwaste and Turbine Buildings. An off-site building, the Shoreham-Wading River fire house, a multistory reinforced concrete building was also selected. Collection of measurements for background determination is performed in accordance with an approved station procedure. The methods for background determinations for each type of Termination Survey measurement are summarized below.

a. Direct Surface Beta-gamma Measurements

To determine background for direct surface beta-gamma measurements, a series of counts of at least one minute duration are taken in sequence. The counts are accumulated by a scaler in the preset time accumulation mode.

Outdoor background measurements are performed on types of surfaces where beta-gamma surface measurements may be taken in the Termination Survey. These include concrete pads, loading docks, pavement, and roofs.

b. Direct Surface Alpha Measurements

The background response of alpha survey instruments is small in comparison with the response when contamination is present. Backgrounds for direct alpha measurements are collected as needed according to an approved procedure.

c. Removable Surface Beta-gamma Measurements

Background determinations of beta-gamma smear counters are made by taking a count of a blank smear, usually on each day of operation. Background determination is performed in accordance with an approved procedure.

d. Removable Surface Alpha Measurements

The background count rate in smear counter alpha detectors is also very low. Background for each counter is determined by counting a blank smear, in accordance with an approved procedure.

e. Gamma Exposure Rate Measurements

A pressurized ion chamber is used to establish gamma exposure rate background at the Shoreham site. The background is used as the baseline for demonstrating that residual gamma exposure rate levels are below 5 μR/hr above background (measured at one meter). The pressurized ion chamber is used as the reference instrument for establishing the gamma exposure rate background and the μR equivalent response of portable microrem meters. The latter are used for the bulk of the Termination Survey gamma exposure rate measurements. A series of paired measurements has been taken with the two instruments to develop the correlation.

A reference value for the gamma exposure rate has been established for the Shoreham site. It is based upon a series of pressurized ion chamber measurements taken one meter above the ground surface in outdoor areas and inside buildings on the site in areas demonstrated to be free of residual contamination from Shoreham operations. The value is 6 μR/hr (5.8 $\pm$ 1.2 μR/hr - one standard deviation). This reference value is used for all survey units, except where actual background conditions are shown to vary significantly, such that area-specific background values are warranted to reduce the bias in reported population average "above background" values.

f. Specialized Measurements

It has been observed that detector background for direct beta-gamma measurements is affected by conditions in the immediate vicinity of the

detector. Significant variations from background reference values have been observed. These variations are caused by the natural radioactivity composition of materials and by shielding effects in some cases. As a result, background measurements for special conditions have been compiled for use in calculating measurement net dpm/100 cm^2 values to reduce the bias in survey unit population statistics. Special condition backgrounds have been compiled for example for: ceramic and clay tile materials, porcelain fixtures, poured concrete materials, roofing materials, large bore piping, embedded piping and others. Guidance for application of special condition backgrounds is provided in an approved procedure.

g. Verification of Background Measurement Population

Each population of background measurements is analyzed using Equation 8-22 of draft NUREG/CR-5849 (BE92) to ensure that the number of measurements in the data set is adequate to characterize the background mean value to within $\pm$ 20% at the 95% confidence level.

4.4.4 Documentation And Control of Background Measurements

Background measurements are collected and recorded in accordance with a Termination Survey procedure. Background results for reference values and special condition backgrounds are compiled in a memorandum which is attached to the survey Final Report.

4.5 Sample Analysis

As indicated in Section 3.5, an in-depth sample analysis capability is available for the Termination Survey. Routine samples of sediment, paint chips and debris will be qualitatively evaluated for the presence of Co-60 via gamma spectroscopy. The need for additional sampling and analysis will be determined on the basis of this initial evaluation.

Figure 4.1
View of Structural Survey Unit Showing Grid Placement

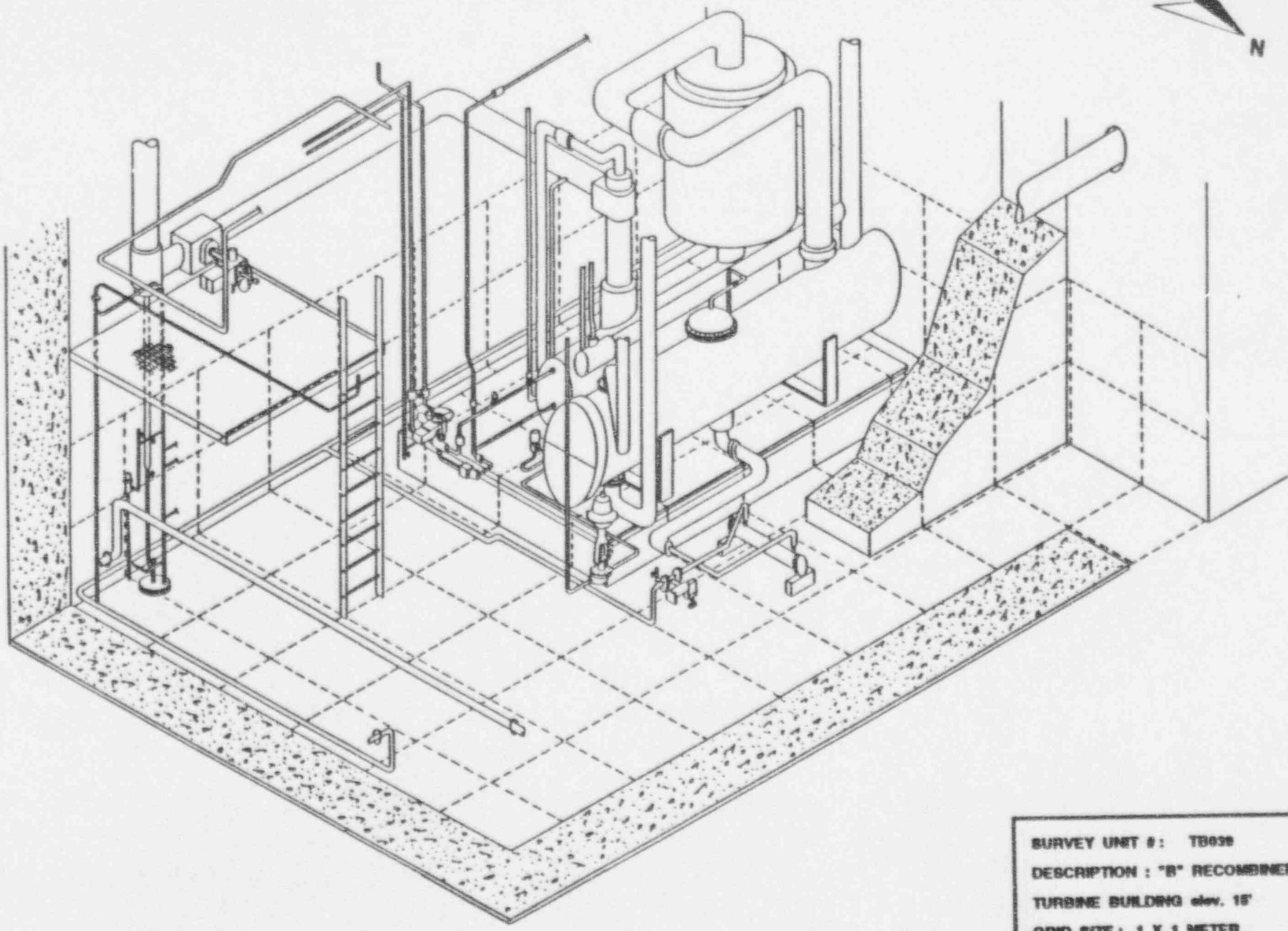
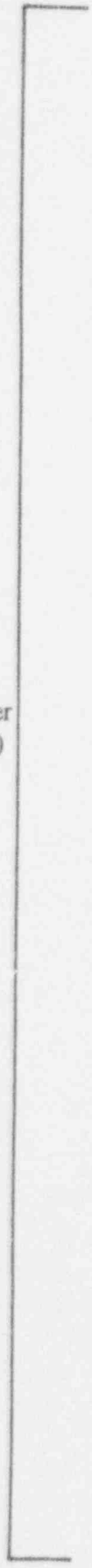


Figure 4.2

Shoreham Decommissioning Termination Survey Grid Map



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5.0 Data Interpretation

All measurements are converted to the appropriate units for comparison with release criteria limit values. Surface activity measurements are converted to units of dpm per 100 cm^2. Gamma dose rate measurements are converted to exposure rate units of μR/hr and the background value subtracted to obtain the net exposure rate. These calculations are performed using equations given in Section 8.0 of draft NUREG/CR-5849 (BE92), indicated below. Additional calculations are made to determine a "critical value" for both total and removable surface activity measurements, to adjust these total and removable surface activity measurements for the presence of the undetected Fe-55 and other hard-to-measure radionuclides where appropriate, and to determine the measurement specific "action level" which indicates the need for additional investigations and/or measurements. Average values for each survey unit (and/or subunit) are compared with the release criteria values. Confidence intervals are estimated for mean values of each survey unit (and/or subunit) at the 95% confidence level. Figures 5.1 through 5.5 describe the flowpaths followed to convert, calculate, analyze and interpret the data for the four (4) basic types of survey measurements made for the termination survey.

5.1 Conversion of Measurements to Reporting Units

5.1.1 Direct Measurements - Total Surface Activity

Measurements of total surface contamination activity are converted from observed gross counts per minute to net activity concentration by subtracting the background counting rate for the instrument and correcting the net count rate for geometry and efficiency to obtain results in dpm/100 cm^2 units. Total surface activity measurement results are reviewed to ensure that the applied background values are appropriate, i.e.,not "too high" so as not to potentially "mask" areas of contamination.

The following calculations and conversions are applicable to measurements of total surface activity:

a. Unit Conversion

(5.1)
$$dpm_{net} = \frac{gcpm - bcpm}{E\left(\frac{A}{100}\right)}$$

where:

dpm_{net} = total surface activity dpm/100 cm^2,

gcpm = detector reading in gross counts per minute (assumed to be entirely due to Co-60 activity),

bcpm = detector background in counts per minute,

E = detector efficiency in counts per disintegration, and

A = area of detector sensitive region (cm^2).

b. Calculation of "Critical Level", L_c (in units of dpm/100 cmý)

(5.2)
$$L_c = 1.96 \frac{\sqrt{s_s^2 + s_b^2}}{E\left(\frac{A}{100}\right)}$$

where:

L_c = Critical Level, defining measurements above normal background distribution, and

s_s = counting error in sample measurement, or

$$s_s^2 = \frac{c}{t^2}$$

where:

c = measurement total counts (sample plus background), and

t = measurement count time.

s_b = counting error in background measurement, or

$$s_b^2 = \frac{B}{t_B^2}$$

where:

B = total background counts, and

t_B = background count time.

E = detector efficiency in counts per disintegration,

A = area of detector sensitive region (cm^2), and

1.96 = 97^{th} percentile value of a one-tailed normal distribution.

c. Adjustment to Account for Hard-To-Measure Radionuclides

Measurements converted using Equation 5.1 which exceed the "critical level" as determined by Equation 5.2 are reported as having surface activity above background levels. This activity is assumed to result from residual Co-60 contamination. Therefore, adjustment for the undetected presence of Fe-55, Tritium and other hard-to-measure (HTM) nuclides is made as follows:

(5.3) $$dpm_{adj} = f\,(dpm_{net})$$

where:

dpm_{adj} = surface activity concentration adjusted to include Fe-55 and other hard-to-measure species (in dpm/100 cm^2), and

dpm_{net} = surface activity concentration from Equation 5.1 (in dpm/100 cm^2), and

f = numerical factor; (see Table 5.1 for factors).

Table 5.1

Adjustment Factors for Surface Activity Measurements

Area - Description	Adjustment Factor, f
Reactor Bioshield Concrete	35.9
Reactor Bioshield Concrete Removable Activity	12.4
Reactor Bioshield Steel	24.2
Reactor Bioshield Steel Removable Activity	1.2
Dry Cutting Station	2.6
Spent Fuel Pool	2.3
Reactor Vessel Bottom Head	2.6
All Other Areas	1.2

The numerical factors, f in Table 5.1 are described in Section 3.2.

d. Calculation of "Action Level", L_R (in units of dpm/100 cmý)

(5.4)
$$L_R = 3 \frac{\sqrt{s_s^2 + s_b^2}}{E\left(\frac{A}{100}\right)}$$

where:

L_R = Action Level, defining those measurements considered significantly above background (and assumed to include Fe-55 and possibly other HTM radionuclides). In survey units (or subunits) classified as "Unaffected Areas", values above this level require investigation, including additional measurements and possible reclassification of the survey unit (or subunit).

s_s, s_b, E, and A are as defined for Equation 5.2, and

3 = 99.7th percentile value of the normal distribution. [1]

5.1.2 Removable Contamination Measurements

Measurements of removable surface activity are converted from gross count rate to units of net dpm/100 cm^2 by subtracting the background count rate of the smear counting detector and correcting the net count rate for detector geometry and efficiency.

The following calculations and conversions are appropriate to measurements of removable surface activity:

a. Unit Conversion - determined by using Equation 5.1.

b. Calculation of "Critical Level", L_c - determined by using Equation 5.2.

c. Adjustment for Fe-55 and other Hard-to-Measure Radionuclides

Similar to the treatment of total surface activity measurements, measurements of removable surface activity which exceed the "critical

[1] The three (3) sigma coefficient represents the 99.7th percentile of the standard normal distribution (two-tailed), taken from NUREG/CR-2082, "Monitoring for Compliance with Decommissioning Termination Survey Criteria", p. 132.

level" are adjusted to include the undetected Fe-55, and possibly other HTM activity, using Equation 5.3.

d. Calculation of "Action Level", L_R

For removable surface activity measurements, the "action level", L_R, is established at 250 dpm/100 cm^2 for all measurements. This value corresponds to 25 percent of the removable residual contamination release criterion as specified in Table 3.1. In survey units (or subunits) classified as "Unaffected Areas", measurements above this level require additional investigation, including additional measurements and possible reclassification of the survey unit (or subunit).

5.1.3 Gamma Exposure Rate Measurements

Most gamma exposure rate measurements are taken with microrem meters. A correction factor is applied to each reading to convert to units of μR/hr, as discussed in paragraph 4.4.3.e. The gamma exposure rate background is subtracted from the gross μR/hr value to obtain net gamma exposure rate in units of μR/hr. Gamma exposure rate measurement results are reviewed to ensure that the applied background values are appropriate, i.e.,not "too high" so as not to potentially "mask" areas of elevated gamma exposure rates.

The following calculations and conversion are applicable to gamma exposure rate measurements:

a. Unit Conversion

(5.5) $$\frac{\mu R}{hr} = 3.06 + 1.07\left(\frac{\mu Rem}{hr}\right) - \left(\frac{\mu R}{hr}\right)_{bkg}$$

where:

μR/hr = net gamma exposure rate in micro R per hour,

μRem/hr = measured gross micro rem per hour,

$(\mu R/hr)_{bkg}$ = background exposure rate in μR/hr,

3.06 = constant term of the correlation factor, and

1.07 = proportional term of the correlation factor determined as described in paragraph 4.4.3.e.

b. Calculation of "Action Level", L_R

For gamma exposure rate measurements, the "action level", L_R, is established at 5 μR/hr above background for all measurements. Measurements above this level require additional investigation, including additional measurements, and a calculation to demonstrate the exposure rate averaged over a 10 m^2 area centered on the measurement location exceeding L_R does not exceed 5 μR/hr above background.

5.2 Comparison With Release Criteria Limits

The method outlined below will be used to demonstrate attainment of the release criteria limits.

5.2.1 Attainment of Release Criteria for Surface Contamination

a. Total Surface Activity (fixed plus removable contamination)

Individual measurements:	Do not exceed elevated area gross activity guideline value (contaminated area not to exceed 100 cm^2). See Table 5.2 for values.
Local area average:	Does not exceed gross activity guideline value averaged over an area of one m^2. See Table 5.2 for values.
Population [2] - random sampling:	Upper limit of confidence interval (Equation 5.6) for the mean value is below gross activity guideline value. See Table 5.2 for values.
Population [2] - biased sampling:	Upper limit of confidence interval (Equation 5.6) for the mean value is below gross activity guideline value. See Table 5.2 for values.

[2] A population for Termination Survey purposes, refers to a survey unit (or subunit if so specified). More precisely, a population represents the collection of all possible values of a parameter, e.g., total surface contamination, being measured through a sample of its members.

b. Removable Surface Contamination

Individual measurements:	Do not exceed 1000 dpm/100 cm^2.
Population - random sampling:	Upper limit of confidence interval (Equation 5.6) for the mean value is below 1000 dpm/100 cm^2.
Population - biased sampling:	Upper limit of confidence interval (Equation 5.6) for the mean value is below 1000 dpm/ 100 cm^2.

c. Calculation of Upper Limit of Confidence Interval

(5.6)
$$u_a = \bar{x} + t_{1-a,\,df} \frac{s_x}{\sqrt{n}}$$

where:

u_a = upper confidence limit of population mean, and

$\bar{x}$ = population mean value

$t_{1-a,\,df}$ = upper confidence level value (as obtained from Appendix B, Table B-1, of draft NUREG/CR-5849); df (degrees of freedom) is equal to n - 1 and "a" is the false positive probability.

s_x = population standard deviation

n = number of measurements in the population

5.2.2 Attainment of 5 microR per hour Criterion

Individual measurements:	Net exposure rate does not exceed 10 μR/hr.
Local area average:	The the net exposure rate does not exceed 5 μR/hr when averaged over an area of 10 m^2.
Population - random sampling:	Upper limit of the confidence interval Equation 5.6) for the mean net exposure rate does not exceed 5 μR/hr.

Population - biased sampling: Upper limit of the confidence interval (Equation 5.6) for the mean net exposure rate does not exceed 5 μR/hr,

5.2.3 Evaluation of Soil Sample Results

Since soil contamination from Shoreham operations has not been detected in the Site Characterization Study (LILCO90a), or in the REMP Program, it is believed to be unlikely that detectable contamination will be found in the soil column on the site in the Termination Survey. As there is no limit or criterion for residual contamination in soil for release of the site for unrestricted use, soil radioactivity concentration measurements will be compared to REMP MDAs and historical REMP measurement result ranges for individual isotopes. For gamma emitters where no REMP data exists, as in the case of Co-60 in soil, the current NRC criterion is adopted and results below 8 pCi/gm average concentration in the top 15 cm of the soil column will be considered to be acceptable. [3]

5.3. Calculation of Gross Activity Guideline Values for Total Surface Activity

When mixtures of radionuclides are involved whose members have different guideline values, the gross activity guideline value (GAG) is substituted for the guideline value. The gross activity guideline values are obtained from the "sum of fractions rule" in Appendix A of NUREG/CR-5849 (BE92), whereby the sum of the ratios of each radionuclide to its guideline value must be ≤ 1. The resulting formula for the GAG is:

$$GAG = \frac{1}{\frac{F_1}{G_1} + \frac{F_2}{G_2} + \ldots \frac{F_n}{G_n}} \qquad (5.7)$$

where:

GAG = Gross Activity Guideline Value in dpm/100 cm^2,

F_i = fraction of total activity due to the ith radionuclide,

G_i = guideline value for the ith radionuclide, in dpm/100 cm^2. The guideline values for principal radionuclides at Shoreham are: Co-60 = 5000; Fe-55 = 200,000; H-3 = 200,000 dpm/100 cm^2 (USNRC94).

[3] The value of 8 pCi/gm in soil is applied to the total concentration of Shoreham-produced gamma-emitting radionuclides.

The elevated area gross activity guideline value is obtained as the product: 3 times the gross activity guideline value. The calculated gross activity guideline values applicable to the various areas are given in Table 5.2.

Table 5.2

Gross Activity Guideline Values

Area	Gross Activity Guideline Value	Elevated Area Gross Activity Guideline Value
Bioshield Concrete	95,900	287,700
Bioshield Steel	76,900	230,700
Dry Cutting Station	11,100	33,300
Spent Fuel Pool	9,400	28,200
Vessel Bottom Head	11,100	33,300
All Other Areas (1)	5,000	15,000

Table 5.2 Note: (1) The gross activity guideline and elevated area gross activity guideline values for all other areas are calculated to be 5,900 and 17,700 dpm/100 cm^2, respectively when the special limits for Fe-55 and Tritium are applied. However, the standard Regulatory Guide 1.86 guideline values of 5,000 and 15,000 dpm/100 cm^2 are used to maintain consistency with earlier results and to provide an additional degree of conservatism.

Figure 5-1

DATA REVIEW FLOWPATH FOR TOTAL SURFACE ACTIVITY - AFFECTED SURVEY UNITS

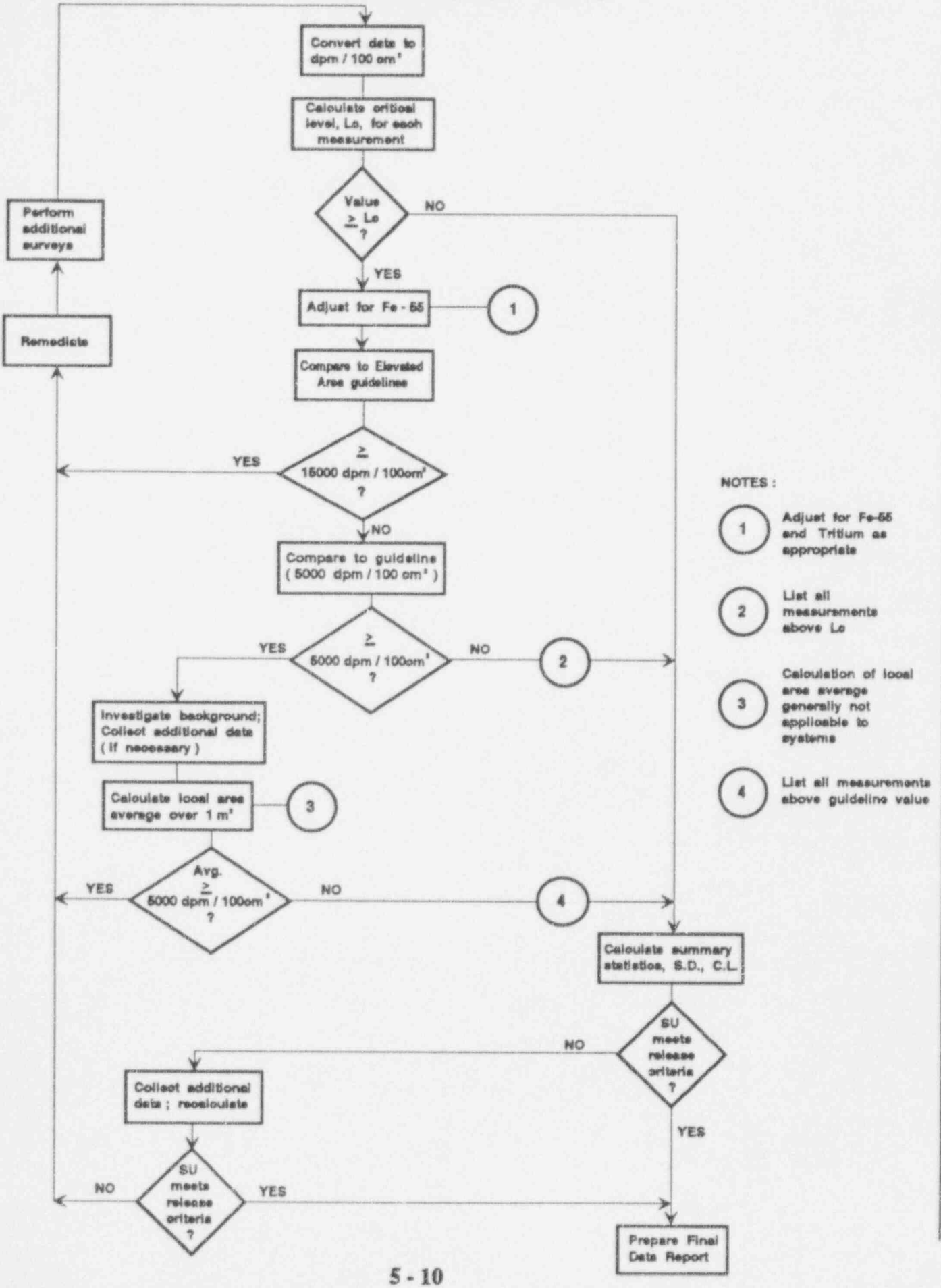


Figure 5 - 2

DATA REVIEW FLOWPATH FOR TOTAL SURFACE ACTIVITY - UNAFFECTED SURVEY UNIT

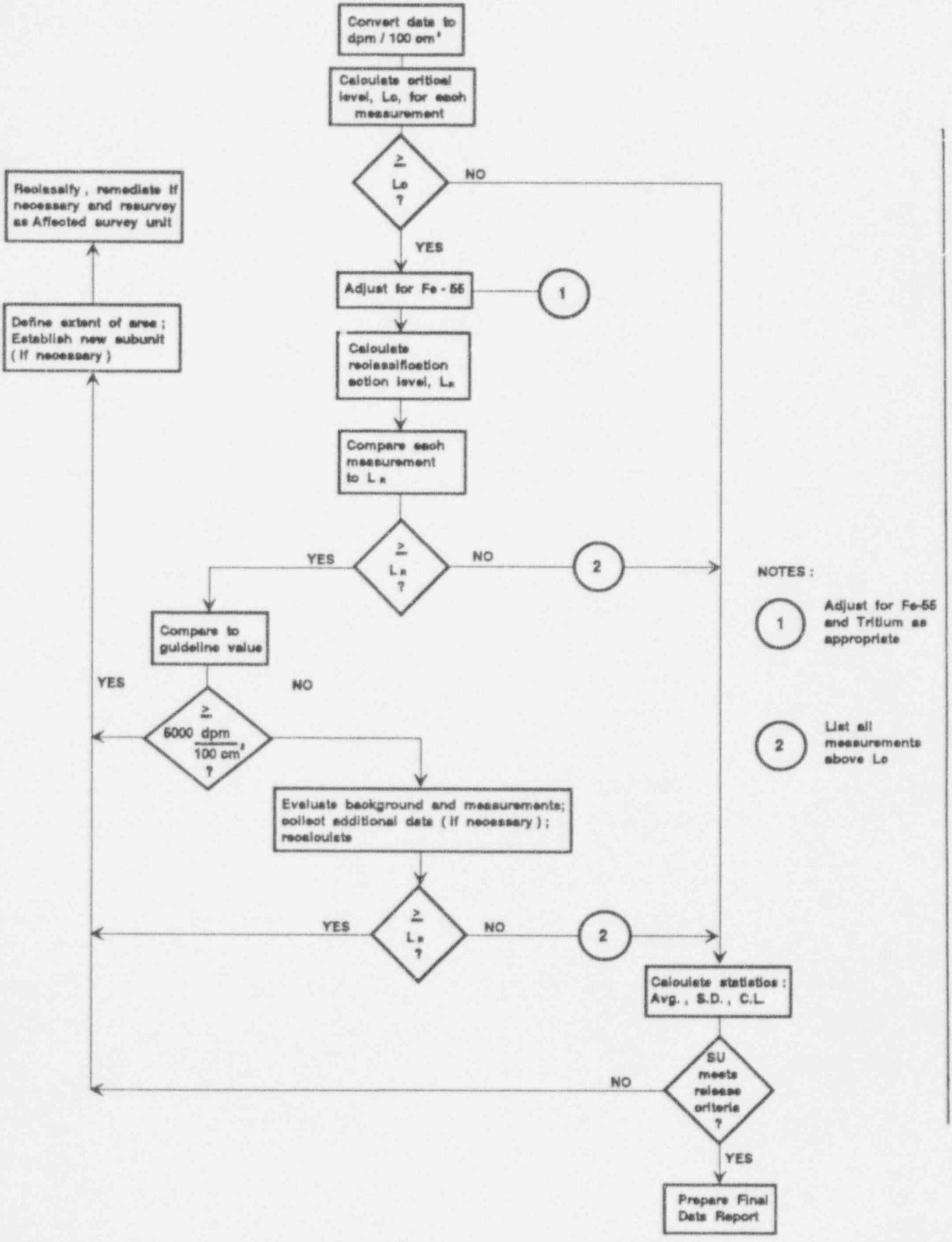


Figure 5 - 3

DATA REVIEW FLOWPATH FOR REMOVABLE SURFACE ACTIVITY

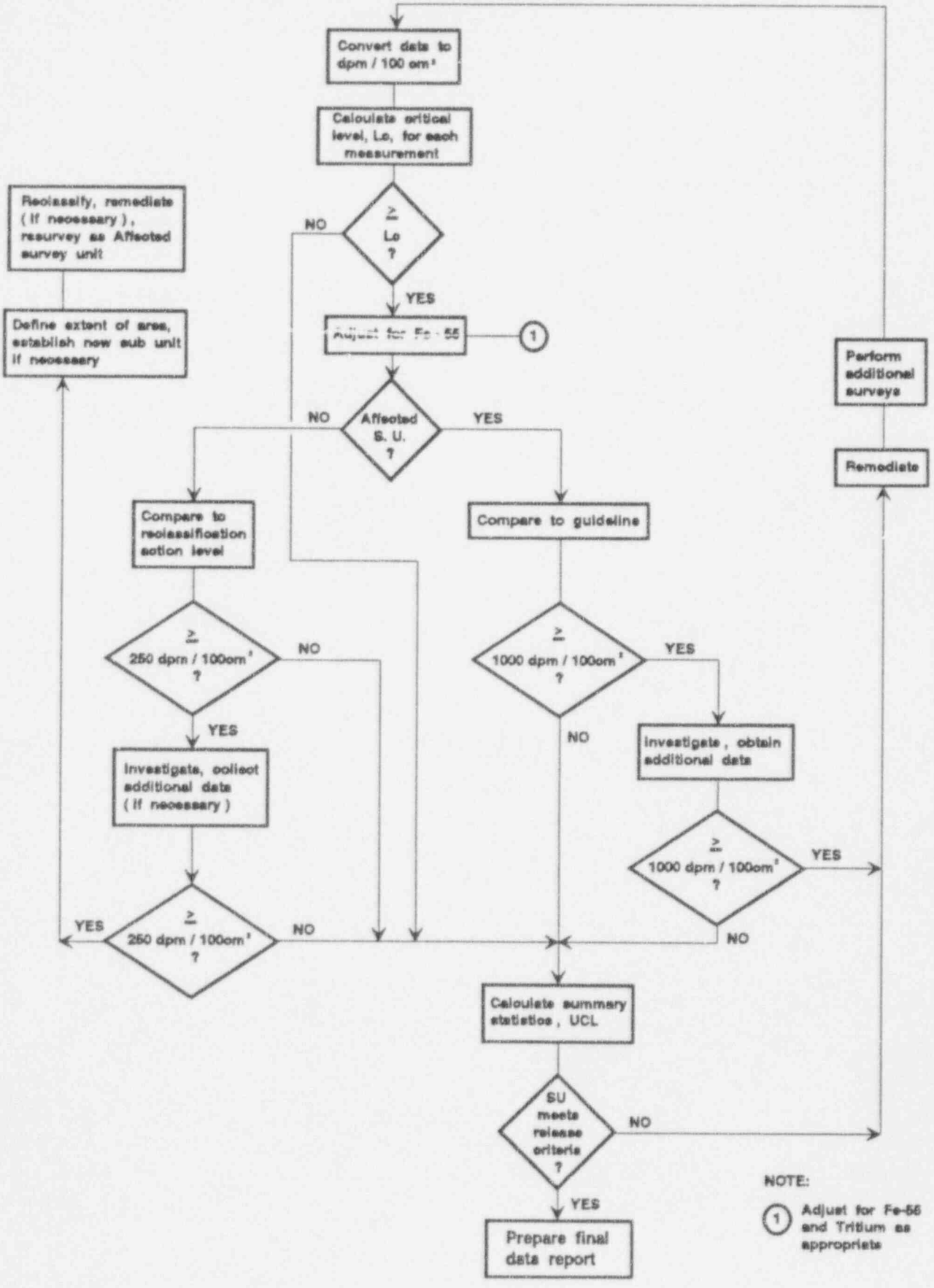


Figure 5 - 4

DATA REVIEW FLOWPATH FOR GAMMA EXPOSURE RATE MEASUREMENTS

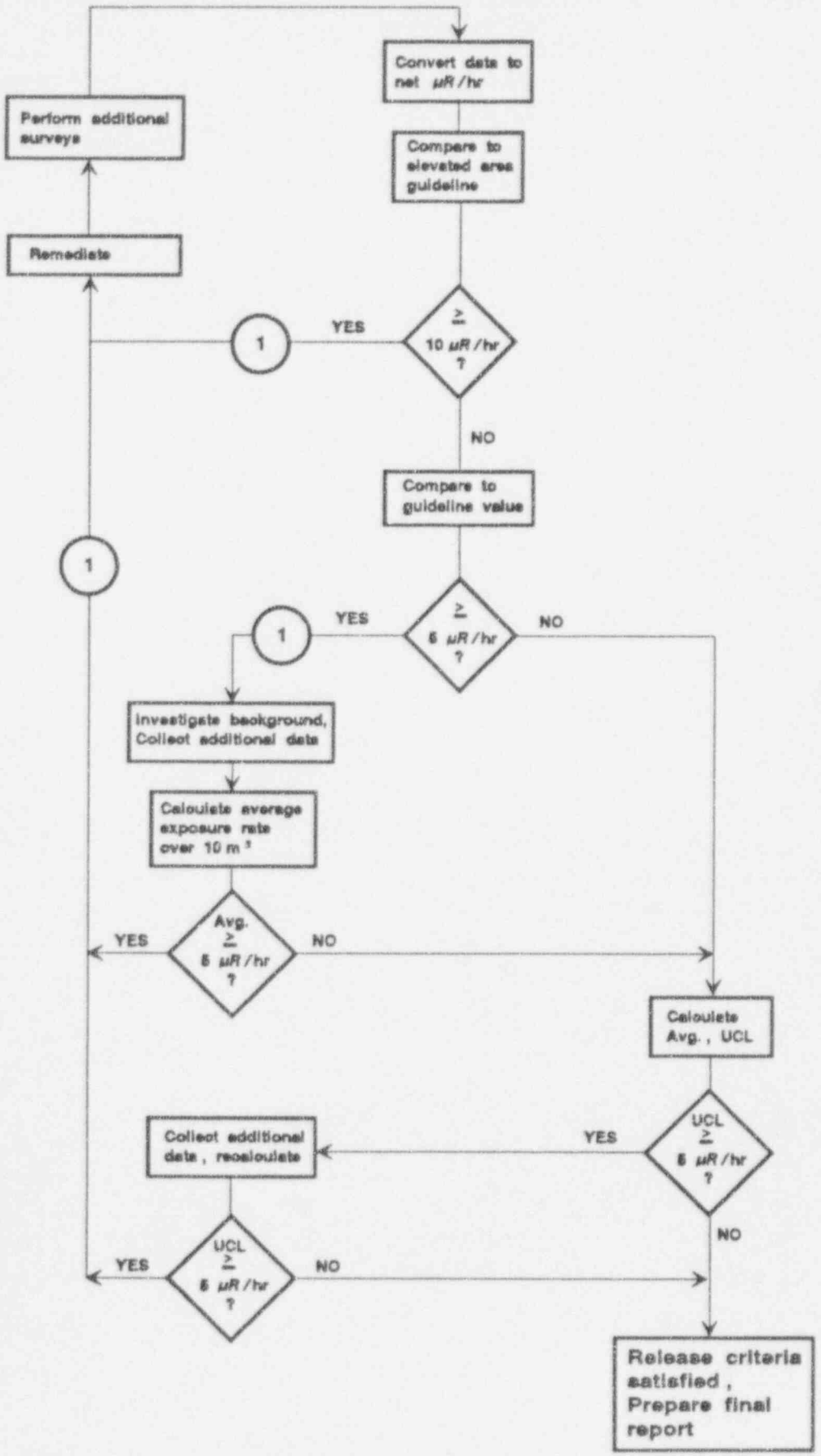
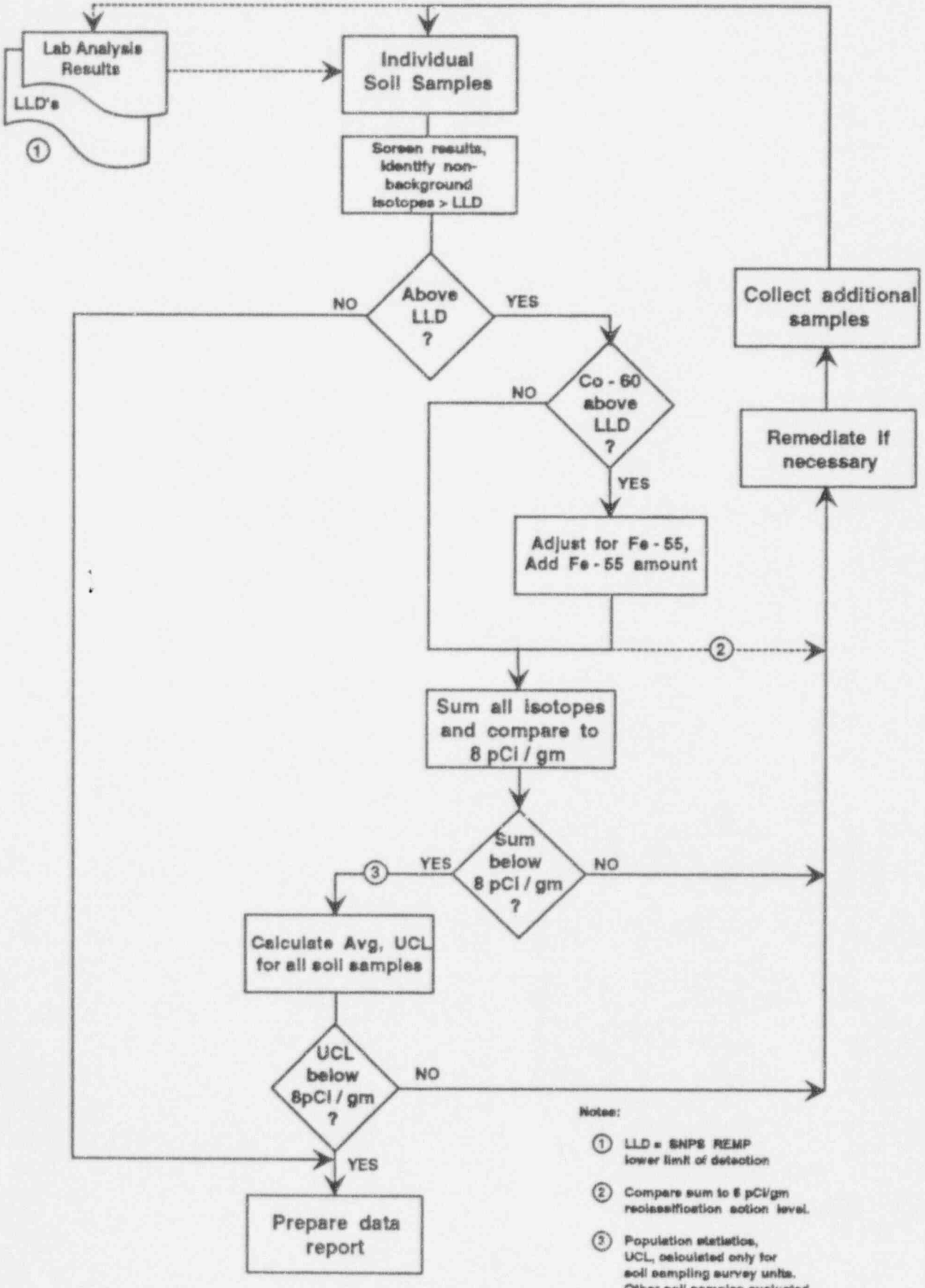


Figure 5 - 5

DATA REVIEW FLOWPATH FOR SOIL SAMPLES



6.0 Final Report

Upon completion of each major phase of the Termination Survey, an updated Final Report will be prepared for submission to the Nuclear Regulatory Commission. This report will meet the intent of Regulatory Guide 1.86 (USAEC74) for final survey reporting. The report will follow the guidance of draft NUREG/CR-5849 (BE92) regarding content.

6.1 Topical Outline

The Final Report will address the following topics. The report will provide adequate data and discussion of each topic to meet the intent of NUREG/CR-5849 (BE92). The following describes the format for the Final Report with regards to document volumes, topical outline and content:

Volume 1

1.0 Background Information
- 1.1 Reason for Decommissioning
- 1.2 Management Approach and Organization

2.0 Site Description
- 2.1 Type and Location of Facility
- 2.2 Ownership
- 2.3 Facility Grounds - Survey Scope
- 2.4 Facility Structures
- 2.5 Plant Systems
- 2.6 Outdoor Areas

3.0 Operating History
- 3.1 Licensing and Operation
- 3.2 Processes Performed
- 3.3 Waste Disposal History and Practices

4.0 Decommissioning and Supporting Activities
- 4.1 Decommissioning Objectives
- 4.2 Site Characterization
- 4.3 Radiological Environmental Monitioring Program (REMP)
- 4.4 Radiological Effluent Reports
- 4.5 Decontamination and Dismantlement Activities

5.0 Termination Survey Methodology
- 5.1 Sampling Parameters
- 5.2 Background Levels
- 5.3 Major Contaminants
- 5.4 Guidelines Established

5.5 Equipment and Techniques Applied
5.6 Survey Process
5.7 Survey Controls
5.8 Data Analysis

6.0 Termination Survey Results
6.1 Survey Results
6.2 Findings
6.3 Final Configuration

7.0 References

8.0 Glossary

Volume 2, (Multiple Books)

Survey Unit Release Records
1. Tabulated Results for Individual Survey Units
2. QC Replicate Surveys
3. Survey Maps

Volume 3

Termination Survey Program Description
Termination Survey Plan
Implementing Procedures
Technical Memoranda and Information

Volume 4, (Multiple Books)

Supporting Documentation

1. Nuclear Quality Assurance Surveillance Reports
2. Nuclear Quality Assurance Audits
3. LIPA Deficiency Reports

As discussed in Section 3.8, an updated Final Report will be submitted upon completion of the each phase of the termination survey. The first of these reports contains the four volumes, as described above, with books included for Volumes 2 and 4 to detail the surveys completed. The second and subsequent submittals of the Final Report consist of updates to Volumes 1 and 3, and additional books to be included in Volumes 2 and 4.

The Final Report provides information which substantiates the survey findings and conclusions. Such information includes, but is not limited to: Reports of Nuclear Quality Assurance Department (NQAD) audits and surveillances of the termination

survey, quality control (QC) survey results, and survey unit release records (including applicable survey maps).

6.2 Reporting of Survey Findings

6.2.1 Summary

Measurement results are reported at several levels of detail. An overall summary of the measurement results and conclusion that the facility meets the release criteria is provided. A tabular data summary shows the results for each major category of survey unit: structures, outdoor areas and plant systems. This tabulation identifies the number of survey units in each category, the maximum individual measurement values and maximum survey unit UCLs for the measurements of each type: total surface beta-gamma, total surface alpha, removable surface beta-gamma and removable surface alpha activity concentration, and gamma exposure rate. A more detailed summary table, is provided which shows the following for each survey unit: number of measurements, maximum, and UCL for each type of measurement. Maximum individual measurement values and upper limit of confidence intervals about the mean (at the 95% confidence level) in units of dpm/100 cm^2 are reported for comparison to the release criteria surface activity limits in Table 3.1. A summary of gamma exposure rate measurements is presented by similar treatment, showing that maximum value is less than 10 μR/hr above background and the mean value (UCL) for each survey unit is less than 5 microR/hr above background. The results for soil sampling are similarly presented.

6.2.2 Summary Data Reporting for Each Survey Unit

Within the release record for each survey unit or subunit, the number of measurements and the upper limit of the confidence interval about the mean (at the 95 % confidence level) are reported in tabular form. These are reported in units of dpm/100 cm^2 for each type of measurement: total surface beta-gamma, total surface alpha, removable surface beta-gamma and removable surface alpha activity concentration. Gamma exposure rate measurement results are reported showing the number of measurements and upper limit of the confidence interval about the mean (at the 95 % confidence level) for each survey unit (and/or subunit). The release limit value for each type of measurement is provided in the release record.

Results of sampling measurements, e.g., sediment, paint, concrete, other debris, are reported in the release record for each survey unit.

Any portions of individual survey units found not to meet any of the various release criteria will necessarily be remediated. The Final Report will identify all areas within the survey which required reclassification and/or remediation.

For each of these areas, the initiating survey results, investigation, corrective actions, and followup survey results will be provided.

6.2.3 Detailed Data Reporting

The result of each measurement taken in the termination survey are tabulated in the individual release record for each survey unit. Table 6.1 shows an example of a typical format for individual measurement tabulations for a structural survey unit. Results of alpha activity measurements are reported in a similar format for those survey units where alpha measurements are taken. Tables of similar format are used to report data for systems and outdoor survey units. Final data reports provide indication of individual measurements which exceed the "critical level", calculated as described in Section 5.0, and the adjustment for Fe-55 activity, where applicable. Furthermore, the release records for survey units classified as "unaffected" provide a listing of individual measurements which exceed the "action level", calculated as described in Section 5.0, and the results of the investigation of these measurements.

Table 6.1

Termination Survey Detail Data Report

page No. 1
Date: 04/07/93

SHOREHAM DECOMMISSIONING PROJECT
Termination Survey Data Report

Survey Unit ID. :SU068 System Code: W23 Survey date 01/29/93
Name: CHLORINATION Building: MULTI BUILDING SYSTEM

total surface activity (beta-gamma)

point	gcpm	bk_cpm	eff	dpm/100cm2	Lc	>Lc	+Fe55	Lr	>Lr
1	33	22	0.017	647	855			1309	
5	23	22	0.017	58	773			1184	
6	26	22	0.017	235	798			1223	
7	31	22	0.017	529	839			1285	
11	22	22	0.017	0	764			1171	
12	22	22	0.017	0	764			1171	
13	29	22	0.017	411	823			1260	
17	16	22	0.017	-352	710			1088	
18	29	22	0.017	411	823			1260	
19	21	22	0.017	-58	756			1157	
23	29	22	0.017	411	823			1260	
24	15	22	0.017	-411	701			1073	
25	27	22	0.017	294	807			1235	
29	24	22	0.017	117	781			1197	
30	17	22	0.017	-294	720			1102	
31	22	22	0.017	0	764			1171	
35	27	22	0.017	294	807			1235	
36	41	22	0.017	1117	915	X	1330	1401	
37	28	22	0.017	352	815			1248	
41	15	22	0.017	-411	701			1073	
42	24	22	0.017	117	781			1197	
43	20	22	0.017	-117	747			1144	
47	21	22	0.017	-58	756			1157	
48	32	22	0.017	588	847			1297	

7.0 References

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USAEC74, U. S. Atomic Energy Commission, Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors," June 1974.

USNRC92, "Order Approving the Decommissioning Plan and Authorizing Decommissioning of Shoreham Nuclear Power Station, Unit 1," Docket No. 50-322, June 11, 1992.

USNRC94, U. S. Nuclear Regulatory Commission, Letter from C. Pittiglio to A. Bortz, "Approval of a Modification of Facility Release Criteria for Tritium and Iron-55 Surface Contamination at Shoreham Nuclear Station Unit 1", Docket No. 50-322, June 7, 1994.

8.0 Glossary

Action Level - A contamination level, denoted L_R (in dpm/100 cm^2, adjusted for Fe-55), used as the investigation threshold in unaffected survey units to evaluate the need for reclassification as affected. The action level is defined as:

Removable surface contamination measurements: 250 dpm/100 cm^2.

Total surface contamination measurements: the upper confidence limit at the 99.7 % confidence level, (3 sigma), of the net count distribution.

soil contamination: 6 pCi/gm.

Affected Area - A designation assigned to a survey unit which indicates that the survey unit has a potential for containing residual radioactive contamination.

Alpha Affected Area - A designation assigned to a survey unit which indicates that the potential exists for alpha contamination.

Biased Sample - A method of selecting survey measurement locations which incorporates a non-random error, i.e., a method which selectively chooses locations for measurements which have a higher probability of contamination than those locations not selected.

Characterization Survey - A radiological survey and supporting evaluations performed to establish the Shoreham Facility baseline radiological condition for planning decommissioning activities. The Characterization Survey activities are described in and controlled by the Site Characterization Program Description.

Component - An individual equipment item, e.g., a valve, pump, tank, motor, etc. which is identified in the Shoreham Composite Component List.

Composite Component List (CCL) - A controlled listing of components installed at Shoreham. The CCL identifies all components other than cable tray and conduit supports in the SNPS design. The CCL also defines which of these components are safety related, which were formerly safety related, as stated in the USAR, and which are not.

Confidence Interval - A range of values derived from a sample such that there is a probability α, that a population parameter being estimated, e.g. a mean value, lies within the range.

Confidence Level - The probability α, associated with a confidence interval which expresses the probability that the confidence interval contains the population parameter value being estimated (MA90).

Critical Level - A calculated value, used as a decision level to determine when a surface contamination measurement is above background, i.e., is due to contamination. It is defined as the upper confidence limit, at the 95 % confidence level, (1.96 sigma) of the observed net count distribution whose mean value value is zero (converted to dpm/100 cm^2).

DECON - The decommissioning alternative which involves prompt removal of radioactive materials to achieve residual contamination and radiation levels which are below limits established to permit the facility to be released for unrestricted use.

Direct Measurement - A radiological survey measurement performed by holding a detector stationary on or close to the surface and recording the response.

Elevated Area Guideline - The value (net value above background) which individual measurements may not exceed under any conditions. For total surface contamination the elevated area guideline is 15,000 dpm/100 cm^2. For gamma exposure rate measurements, it is 10 μR/hr. Elevated area guidelines have not been established for removable surface contamination.

Elevated Area Gross Activity Guideline - Equivalent to the elevated area guideline, but applied to cases where the gross activity guideline applies. It is obtained as the product: 3 times the gross activity guideline.

Fixed Point Measurement - A synonym for direct surface contamination measurement.

History File - A compilation of information prepared for use in planning the termination survey of a survey unit. It summarizes the operational history, characterization survey data, operational surveys and other information to help establish the basis for the design of the termination survey.

Gross Activity Guideline - The guideline which applies when multiple radionuclides are present which have different guideline values. The gross activity guideline value is calculated using the sum of fractions rule in NUREG/CR-5849 Appendix A as shown in Equation 5.7, Section 5.3 of this document.

Guideline value - The principal numerical limits in the facility release criteria. The guideline values are:

total surface beta-gamma contamination: 5000 dpm/100 cm^2,

removable surface beta-gamma contamination: 1000 dpm/100 cm^2,

total surface alpha contamination: 5000 dpm/100 cm^2,

removable surface alpha contamination: 1000 dpm/100 cm^2,

gamma exposure rate: 5 μR/hr, and

soil contamination: 8 pCi/gm.

These limit values are expressed as net values, above the background values for the measuring instruments used.

NOTE: Under certain conditions, individual total surface contamination and gamma exposure rate measurements may exceed the guideline values, as long as they do not exceed the elevated area guideline values and local area average values do not exceed the guideline values.

Maintenance Work Request (MWR) - A form used at Shoreham Nuclear Power Station, controlled by a station procedure, to initiate and track work activities.

NRC - U. S. Nuclear Regulatory Commission

Operational Radiological Survey - A radiological survey performed under Shoreham Health Physics procedures. Operational surveys are distinct from, and usually performed prior to, termination surveys.

Outdoor Area - A category of survey units which includes site grounds, outside surfaces of buildings and small structures located out of doors.

Pipe Crawler - A term used in the Shoreham Decommissioning Project to denote a mechanical device equipped with multiple-detector assemblies (up to nine GM detectors) used to take direct surface radioactivity measurements of piping interior surfaces. The number of detectors used depends on the pipe diameter. The crawlers are manually inserted and maneuvered through piping with flexible fibreglass push/pull rods.

Plant Structures - All Shoreham Nuclear Power Station site buildings and their surfaces (generally identified as civil structures). For purposes of the termination survey, all structures such as platforms, restraints, supports and other physical items not identified in the system MFSK drawings are considered to be structures. External surfaces of piping systems, heating and ventilation systems, tanks, stacks, etc., are also treated as structures in the termination survey.

Population - A collection of all possible values of a radiological parameter being measured in the termination survey. A survey unit (or subunit) is considered to be a population for purposes of drawing inferences regarding the value of a parameter, such as contamination level mean value in the entire survey unit (or subunit), based upon a sample of measured values.

Power Block - The group of major buildings on the SNPS site directly associated with electrical power generation. This group consists of the Reactor, Turbine, Radwaste, and Control buildings.

Process System Index - A listing, controlled by a station procedure, which identifies and assigns a unique identification code to each plant system.

QC Replicate Survey - A radiological survey which consists of repeat measurements at a specified fraction of the survey measurement locations in a survey unit, usually selected at random, to provide an independent check of termination survey measurements.

Random Sample - In survey design, a method for selection of measurement locations whereby each of the individual locations defined in the sample space has an equal probability of being selected. Related terms are: random selection and randomly selected.

Release Criteria - A term used to identify the radiological requirements for release of the Shoreham facility for unrestricted use. These requirements, which consist of specified limits for residual contamination and radiation levels, are specified in the Shoreham Decommissioning Plan.

Release Record - A document compiled for each survey unit (structure, system or outdoor area) which demonstrates that it is suitable for unrestricted use. It contains evaluated survey data and supporting information to provide a concise record of the results and basis for the conclusion that the release criteria are satisfied.

Reporting Units - The units in which each type of survey measurement is expressed for comparison to release criteria limits. For surface contamination measurements the reporting units are dpm/100 cm^2 and for gamma exposure rate measurements the units are μR/hr.

Scan Survey - A qualitative radiological monitoring technique which is performed by moving a detector over a surface (typically within one cm of the surface) at a specified constant speed to detect elevated contamination or radiation levels. Similar terms applied to this technique are: Scan and Surface Scan.

Site Characterization Report - A report (including addenda) which documents the surveys, calculations and evaluations and presents the results of the SNPS Site Characterization Program.

Subunit - A subunit, as used in survey design, is a subdivision of a complex survey unit that incorporates a structure, on item of equipment, or some other feature in order to establish that an appropriate number of survey measurements be made within the subunit, as well as within the survey unit.

Survey Design - The process of determining the type, location, number and frequency (or density) of radiological measurements to be taken in the termination survey.

Survey Design Guidelines - Criteria established to provide the appropriate level of survey intensity for systems, structures and outdoor areas, based upon their classification.

Survey Instructions - Written directions which specify the type and number of measurements to be taken in a survey unit. The survey instructions are in a standard format on forms controlled by a Termination Survey procedure. Each survey package includes survey instructions.

Survey Location - A discrete area or subdivision of a survey unit that is smaller than a subunit but larger than a survey point. In survey design, a survey unit (or subunit) is divided into a collection of survey locations. Specific locations are selected in accordance with the design guidelines based upon the type and classification of the survey unit. In a structural or outdoor survey unit, a location is usually represented by a single grid block. In a system survey unit, a specified length of piping or a component such as a valve is referred to as a survey location. A survey location can contain one or more survey points.

Survey Package - A collection of information in a standardized format for controlling and documenting field measurements taken for the termination survey. A survey package is prepared for each Survey Unit. The survey package includes the survey instructions, a control form, grid map(s), survey measurement data sheets and survey maps.

Survey Point - A smaller subdivision within an area designated as a survey location (grid block, system component) where local measurements are taken, generally referring to an area covered by a detector, or an area of 100 cm^2 when a smear is taken.

Survey Unit - A division of the facility consisting of a grouping of contiguous (usually) structural areas, outdoor areas, or functionally contiguous equipment items. The Survey Unit is the basic entity for management of the termination survey. Three categories of survey units have been established: plant systems, structures and outdoor areas.

Survey Unit Classification Description - A listing of all survey units established for the termination survey which identifies the classification of each as "affected" or "unaffected".

System Final Configuration: - The status of plant systems following completion of the termination survey. The final configuration establishes the nature of the controls required to maintain the integrity of survey results. It also determines the nature of configuration control and engineering review of access methods needed to obtain survey measurements of system component internals. Four categories are established:

(a) Operable - maintained to meet Technical Specifications.

(b) Functional - Essential support, not required per Technical Specifications however, necessary for minimal plant functions, habitability, and preservation concerns.

(c) Protected - not to be operated in the defueled mode. These systems will be left in a de-energized, safe state and laid up in accordance with System Lay-up Implementation Package (SLIP), which specify maintenance and custodial services necessary to protect them pending disposition the LIPA possession only license.

(d) Decommissioned - taken out of service, and completely or partially removed. Remnants of removed systems are abandoned in place.

Systematic Sample - A sample which is obtained by some systematic method as opposed to a random sample; for example, selection from a list using a specified interval for selection (MA90). In a structural survey unit which has been uniformly gridded, a systematic sample could be comprised of every fourth block, for example.

Termination Survey - Radiological measurements, evaluations and supporting activities undertaken to demonstrate that the Shoreham facility satisfies the criteria for unrestricted use. Termination survey field activities may be divided into phases called the Preliminary and Final surveys.

Termination Survey Section - A Section established within the Radiological Controls Division, Operation and Maintenance Department of the LIPA Decommissioning Project Organization to design and implement the termination survey.

Termination Survey Report - A report describing the methods, and results of the Termination Survey. It initiates the NRC review and final inspection of the facility for termination of the facility license. It is also called the Final Report. It will be issued in stages as phases of the Termination Survey are completed. The final update of the Termination Survey Report will document the completed Termination Survey.

Type I Error - In the statistical theory of hypothesis testing, the error incurred by rejecting a hypothesis when it is actually true (MA90). In the termination survey application it is the probability of deciding that the facility (or a survey unit) meets the release criteria, when in fact, it does not. Also called the α error; the error probability is denoted by α.

Type II Error - The probability of deciding that the facility (or a survey unit) does not meet the release criteria, when the true condition is that the facility does meet the criteria. The Type II error probability is denoted by β.

Unaffected Area - A designation used to identify a survey unit which is not expected to contain residual radioactivity from licensed operations based upon the operating history and radiological surveys.

Work Instruction - A document used to guide performance of a task. Work instructions are similar in format and content to a procedure and are issued and controlled under a station Health Physics procedure. A work instruction is approved at the Section Head level.

Appendix A

Survey Design Guidelines

1.0 Introduction

This Appendix provides guidance for preparation of the design of the Shoreham Decommissioning Project Termination Survey. It incorporates the guidance in NUREG/CR-5849 (BE92) with adaptations to account for conditions at the Shoreham facility.

The objective of survey design is to define the measurements necessary to demonstrate that the facility satisfies the release criteria. As it is not feasible to perform detailed radiological surveys of 100 percent of the facility, measurements are performed at selected locations. Due to the physical complexity of the facility with the possibility of residual contamination limited to a relatively small portion of the facility, a stratified sampling approach is utilized. This approach adjusts the intensity of measurements according to the likelihood of encountering residual radioactivity. This approach is implemented at the facility level by classification of individual survey units according to contamination potential. Survey units are further broken down into subunits, if necessary, to provide the appropriate degree of survey intensity for complex structures and plant systems.

2.0 Survey Design Process

2.1 Facility Organization and Classification

At the facility level, the design and organization for the survey consists of the following:

a. division of the facility into discrete entities (survey units) for management of the survey,

b. division of the facility into categories with similar physical characteristics, i.e., structures, systems and outdoor areas,

c. classification of survey units into two major strata based upon potential for residual contamination, i.e., affected and unaffected, and

d. establishment of a reference grid system for identification of measurement locations.

2.2 Survey Design for Individual Survey Units

The Termination Survey is implemented in the field at the individual survey unit level. The design is developed, measurement instructions prepared and measurements completed for each survey unit independently. The guidelines in this Appendix are

applied to each survey unit based upon its classification as affected or unaffected. The guidelines establish the level of measurement intensity for each type of measurement needed to demonstrate that the release criteria are satisfied.

3.0 Facility Breakdown into Survey Units

3.1 Definition of Survey Unit

A survey unit is defined as a division of the facility consisting of like entities for purposes of management of the Termination Survey. Three categories or types have been established: structures, which include building interiors, plant systems, and outdoor areas. These categories allow the grouping of like elements.

3.1.1 Structures

For Termination Survey purposes, structures include the indoor portions of site buildings including the exterior surfaces of plant systems, equipment and furnishings located therein. Structural survey units are established by division of the facility buildings into discrete (usually contiguous) geographical areas taking advantage of existing structural boundaries where possible, i.e., rooms or major elevations of small buildings.

3.1.2 Outdoor Areas

Outdoor areas include all site grounds determined to be within the scope of the Termination Survey, the outside surfaces of buildings and miscellaneous outdoor structures not identified as separate structures. These miscellaneous outdoor structures include equipment storage pads, switchyard transformers and storage tanks, for example.

3.1.3 Systems

Systems included in the Termination Survey as survey units include all nuclear steam supply, reactor control, and process systems, and building service systems associated with the SNPS design and described in the SNPS USAR (LILCO90). Each system identified in the Records Management File Code List is treated as an individual survey unit. A system survey unit encompasses the interior surfaces of fluid carrying piping and components of the system.

3.2 Independence of Structures and Systems

The survey of each individual survey unit is conducted independently. Interior surfaces of piping and equipment which comprise the system survey unit are surveyed independently of the exterior surfaces. Exterior surfaces of equipment and piping are included in the survey of the structural survey unit in which they are located.

3.3 Size and Scope of Survey Units

3.3.1 Structures

A structural survey unit includes the entire surface area of floors, walls, ceilings, and outside surfaces of equipment and furnishings. The size of structural survey units is established by natural boundaries such as rooms in buildings or by building elevation in small buildings. The optimal floor area of a structural survey unit classified as affected is an area of approximately 100 m^2. Units with floor plans as small as 50 m^2 and as large as approximately 300 m^2 can be established if necessary, due to variations in room size or building floor plan size. Unaffected areas in large buildings may be combined to encompass multiple rooms of similar composition into a single survey unit for efficiency in survey administration as long as guidelines for measurement intensity are followed.

3.3.2 Outdoor Areas

The size of individual outdoor area survey units which consist of site grounds is generally determined by features such as roadways, major building boundaries, fences, etc. Building exteriors (all surfaces) generally comprise a single survey unit.

3.3.3 Systems

Each individual system within the scope of the Termination Survey is established as a survey unit. The boundaries of piping and other systems are established by the controlling drawings, usually the "MFSK" series of flow diagrams.

4.0 Classification of Survey Units by Contamination Potential

After breakdown of the facility into survey units, each is classified into one of two strata which identifies each unit as "affected" or "unaffected". Units identified as affected have a possibility of containing residual contamination and those identified as unaffected have a very low probability of residual contamination. Classification of individual survey units is based on the history of radioactive materials involvement or potential for contamination of the overall survey unit.

4.1 Structures and Outdoor Areas

4.1.1 Affected Areas

Structures and outdoor areas are classified as affected for survey design purposes when the following conditions apply:

a. areas with potential contamination based on operating history, or known contamination based on radiological surveys, or

b. areas where radioactive materials were used or stored, and where records indicate spills or occurrences which could have resulted in contamination.

4.1.2 Unaffected Areas

Structures and outdoor area survey units are classified as unaffected if the following conditions apply:

a. not expected to contain residual radioactivity from licensed activities based upon the operating or utilization history and radiological survey results, and

b. not classified as affected.

4.2 Systems

4.2.1 Affected Systems

Systems are classified as affected for survey design purposes if the following conditions apply:

a. the potential for contamination exists based on operating history, or there is a known history of contamination based on radiological surveys, or

b. the system circulated, stored or processed radioactive materials, including: primary coolant, radioactive process or treatment media which were associated with the operation or control of the Nuclear Steam Supply System (NSSS) such that they could become contaminated or experience neutron activation; or where records indicate spills or occurrences which could have resulted in contamination.

4.2.2 Unaffected Systems

System survey units are classified as unaffected if the following conditions apply:

a. not expected to contain residual activity based upon the operating or utilization history and radiological survey results, and

b. do not meet the criteria for classification as affected.

4.3 Reclassification

Survey units may be reclassified subsequent to the initial classification according to the following criteria:

4.3.1 Upward Reclassification

A survey unit (or subunit) must be reclassified from unaffected to affected if at any time in the survey planning, design, or during the actual survey, new information is obtained which indicates that the criteria for an unaffected survey unit are no longer satisfied. An investigation, to determine whether upward reclassification of the survey unit (or subunit) is required, will be initiated if any total surface contamination measurement exceeds the "action level" calculated as described in Section 5.0 of this Plan, if any removable surface contamination measurement exceeds 25 percent of the removable surface contamination limit specified in Table 3.1, or if a single gamma exposure rate measurement exceeds 5 μR/hr. The investigation of a survey unit (or subunit), including additional measurements taken to determine the extent of residual contamination present, should either conclude that the survey unit (or subunit) continues to meet the criteria for an unaffected area or the survey unit (or subunit) will be reclassified as an affected area. If a survey unit is reclassified upward, sufficient measurements must be taken to meet survey design guidelines for affected areas. Only under conditions of such reclassification is it acceptable to classify a subunit as an affected area within a survey unit previously classified as unaffected.

4.3.2 Downward Reclassification

A survey unit may be reclassified downward from affected to unaffected prior to the survey design if additional information is obtained such that the criteria for classification as an unaffected area are satisfied. For example, through the investigation for the history file preparation, additional survey results may become available subsequent to the initial classification which show that no contamination is detected and records indicate that there is no history of radioactive materials use.

4.4 Alpha Affected Areas

Due to the lack of any history of alpha contamination from SNPS operations, alpha surveys are not routinely performed during the Termination Survey. However, in the event that alpha contamination is observed a mechanism is needed to expand the scope of the Survey to ensure that applicable release criteria limits for alpha contamination are satisfied. A survey unit (structural area, system or outdoor area) is classified as an alpha affected area if the following conditions apply:

a. alpha activity greater than 25 percent of release criteria limits has been detected, or

b. the area or system is immediately involved with fuel handling or storage.

In alpha affected survey units, direct measurements and measurements of surface alpha contamination are added to the list of measurements normally performed at the selected measurement locations. Surface scans for alpha activity are performed in the vicinity of the selected measurement locations.

5.0 Reference Grid System and Gridding of Survey Units

The reference grid system for the termination survey establishes a discrete-uniform subdivision of areas covered by the survey. The principal objectives of grid marking are: to establish measurement locations for the survey design and to document where measurements were taken if the need arises to verify measurement results by repeat measurements, or to establish boundaries of areas needing remediation.

5.1 Gridding Affected Areas

5.1.1 Indoor Areas

a. Floors and Lower Walls

Floors and lower walls up to 2 meters from floor level are gridded, i.e., grids are marked on the surfaces. The grid size is one meter square.

b. Upper Walls and Ceilings

Grids are marked on upper walls and ceilings of affected structural survey units if the areas are determined to be suspect as defined in Section 7.1.2. If determined to be suspect, the areas are gridded in accordance with Paragraph a. above. If the area is not suspect, the guidance in Paragraph c. below is applied.

c. Other Surfaces

Grids are marked on surfaces of equipment and other non-regular surfaces in affected structural survey units as needed to aid in the design of surveys and to control and document the location of measurements. If not possible to mark such items into one meter square grids, regularly spaced markings one meter apart may be used.

5.1.2 Outdoor Areas

a. Site Grounds

Site grounds in affected areas are marked off in 10 meter square grids.

b. Roofs and Building Exterior Walls

Roofs and exterior walls in affected areas are marked off into grids not to exceed 5 meters square. Smaller grids (one to three meters square) may be used on small surfaces where the use of five meter grids would result in fewer than 10 grid blocks on a surface.

5.2 Gridding Unaffected Areas

5.2.1 Indoor Areas

Grid marking of surfaces in unaffected indoor structural survey units is not required. However, grids may be marked for reference purposes. The grid size is optional (one to 10 meters square), commensurate with the size of the area. Temporary grid markings may be used to control scan surveys.

5.2.2 Outdoor Areas

Grid marking of surfaces in outdoor areas is not required. If desired for reference purposes, grids may be used. The size is optional, but shall not exceed 30 meters square on site grounds and 10 meters square on building exterior surfaces.

5.3 Gridding for Systems Surveys

Gridding on surfaces of system interiors is not normally performed. Identification of measurement locations is shown on survey maps which contain drawings or diagrams. Interiors of large tanks and vessels in affected systems may be gridded as needed to design the survey. In such cases, the surfaces are marked off into one meter grids. Square grids are used where possible, but tank ends and other non-rectangular surfaces are marked in regular shapes which approximate one meter in area.

5.4 Accuracy of Grid Marking

An accuracy objective which is reasonable for the scale of the SNPS facility is to document a measurement location with an uncertainty of less than one meter. To ensure that this is attained, grid blocks will be marked to an accuracy within 15% of the specified dimension. This also allows for accommodation of areas of irregular shape within a grid block if needed. Remnant areas at the end of rows can be incorporated into the last full size grid block as long as the area of the resultant block is not greater than approximately 25% above the area of the specified grid size.

6.0 Stratification-Breakdown of Survey Units

6.1 Structures

In affected areas, structures are generally divided into subunits for survey design and control. The floor and lower walls comprise a subunit and the upper walls and ceiling a separate subunit. Due to additional structural features and the presence of piping

and equipment, additional subunits may be needed. Separate subunits are usually created to establish "populations" where biased surveys are indicated. For example, piping penetrations in floors and walls where contaminated piping systems were removed are usually grouped into a subunit. A large sump or catchment basin location in a structural survey unit is treated as a separate subunit. Grating, decking, stairs and ladders are usually grouped into a subunit. Equipment skids are usually identified as subunits for survey of the exterior surfaces. Unaffected areas in structures need not be divided into subunits, but in some cases subunits may be used if needed to group elements in complex structures.

6.2 Outdoor Areas

Outdoor site ground areas may be divided into subunits based upon the nature of the surface, e.g., to separate paved and unpaved areas which have different survey requirements.

6.3 Systems

Systems with multiple and separate components are usually divided into subunits for survey design purposes. A typical subunit comprises a large bore component [1] and the adjacent piping. Usually tanks, heat exchangers and other vessels are treated as subunits.

6.4 Classification of Subunits

Subunits may be classified differently from the survey unit in which it resides under certain conditions. A subunit within an affected survey unit may be classified as unaffected if it is a physically separate entity with no likely mechanism for transfer of contamination from the affected portions to the unaffected portions. The typical application of this classification option is the classification of upper walls, ceilings and overhead areas within an affected survey unit as unaffected. The converse is not allowed, however, i.e., classification of an affected subunit within a unit classified as unaffected, except as provided for in Section 4.3.1 for reclassified subunits.

7.0 Measurement Location and Frequency of Measurements

7.1 Structures

This Section provides guidance for determining the frequency of measurements, i.e., the number of measurement locations in structural survey units (and/or subunits). The frequency of measurements in affected areas is determined by:

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[1] Large bore at SNPS refers to piping and the piping system components associated with piping having inside diameters of 3 inches or greater.

a. the sensitivity of scanning surveys for floor and lower walls[2], and

b. the expected level of contamination relative to release criteria limits for upper walls and ceiling areas.

The measurement frequency in unaffected areas is not controlled by scan sensitivity.

Locations for fixed measurements are at or near the center of the one meter grid blocks. Unless specified otherwise in the following paragraphs, the measurements taken at fixed locations are: direct surface beta-gamma, removable surface beta-gamma (smear) and gamma exposure (or microrem) rate measurement at one meter from the surface.

In survey units classified as affected, direct surface and removable surface alpha measurements are taken in addition to the measurements identified above.

7.1.1 Affected Areas - Floors and Lower Walls

Floors and lower walls of affected areas are surface scanned over 100% of the area. Measurements are taken at fixed locations. Measurements are taken systematically in every second grid block. At least 30 measurement locations are selected.

7.1.2 Affected Areas - Upper Walls and Ceilings

In large structural survey units, and those with complex structures and piping in the overhead areas, the survey design is usually implemented by establishing individual subunits for vertical and horizontal surfaces and perhaps additional subunits for equipment and furnishings. Survey intensity of upper walls and ceilings classified as affected areas is determined based upon the suspected level of residual contamination.

a. Not Suspect

The rule for determining than an area is not suspect is: expected contamination levels must be below 50 percent of release criterion for total surface contamination and below 25 percent of the release criterion for removable surface contamination, i.e.,below the following

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[2] The scanning sensitivity value which triggers an increased frequency of fixed measurements is 75 percent of the surface contamination release criteria limit for average total surface contamination: 5000 dpm/100 cm^2. That is, the scanning method must be capable of detecting levels below 3750 dpm/100 cm^2 averaged over an area not to exceed one m^2. As described in Section 4.2 of the Termination Survey Plan, the Termination Survey instrumentation scanning sensitivities are below this value, so the conditional measurement frequencies in NUREG/CR-5849 are not invoked.

values: 2500 dpm/100 cm^2 direct total surface contamination and 250 dpm/100 cm^2 removable surface contamination. If the area is not suspect, a minimum of 30 measurement locations are selected on horizontal surfaces, and 30 on vertical surfaces. The measurement density shall be at least one measurement every 20 m^2 averaged over the area of the subunit. A scan survey is performed in the immediate area of each selected measurement location. The scan survey covers an area of approximately one m^2 or greater. The measurement point locations are usually selected in a biased manner, i.e., focusing on locations where demolition, piping removal or other remediation work occurred and on surfaces where contamination potential is highest.

b. Suspect.

If any area is suspected to be or is above the levels prescribed in Paragraph a. above, the survey protocol is the same as for the floors and lower walls of an affected area. That is, a 100 percent scan with the fixed location measurement frequency of one measurement every second grid block, for a minimum of 30. The area is gridded in 1 m^2 grids.

7.1.3 Unaffected Areas

a. Inside The Power Block

Floors and lower walls of unaffected indoor areas (structures) within the Power Block are scan surveyed over a minimum of ten percent of the area, focusing on walkways and commonly traveled areas. Measurement locations are randomly selected. At least 30 locations or an average of one measurement per 50 m^2 (of floor plus lower wall surface area) are required whichever is greater.

Thirty locations or an average of one measurement location per 50 m^2, whichever is greater, are selected at random from upper walls and ceilings. No scan survey is performed.

b. Outside The Power Block

Floors of unaffected areas outside the Power Block are scan surveyed over a minimum of ten percent of the area, concentrating on traffic areas. At least 30 locations are selected at random, or an average of one per 50 m^2, whichever is greater. Lower walls are not included in the survey of floor areas.

Thirty locations are selected at random from the walls (including lower walls) and ceiling areas, or an average of one per 50 m^2, whichever is greater. No scan survey is performed.

7.2 Outdoor Areas

7.2.1 Affected Areas - Site Grounds

Affected site ground areas are surveyed according to the nature of the surface covering.

a. Paved Areas

Paved areas are surface scan surveyed over 100 percent of the area with a beta-gamma detector. Measurement locations are selected systematically to achieve an average of at least one per 10 m^2 grid such that a minimum of 30 locations are surveyed. Direct and removable surface beta-gamma measurements and gamma exposure rate measurements (at one meter from the surface) are taken at each measurement location.

b. Unpaved Areas

Unpaved areas are scan surveyed over 100 percent of the area using a gamma detector. Gamma exposure rate measurements at one meter from the surface are performed with at least one location per 25 m^2, including at each location where a soil sample is collected, such that at least 30 locations are surveyed.

7.2.2 Unaffected Areas - Site Grounds

Unaffected ground areas are surveyed according to the type of ground surface.

a. General - All Surfaces

A minimum of 30 survey locations are selected at random. At each location (at the approximate center of the grid), a gamma exposure rate measurement is taken at one meter from the surface. Gamma exposure rate measurements are also obtained at each of the randomly selected locations where soil samples are obtained.

b. Paved Areas

The paved areas within each survey unit are evaluated to identify traffic areas and areas with any potential for residual contamination. These areas are scanned for beta-gamma contamination. All such areas are cleared of material as necessary to provide access for the survey. A minimum of 10 percent of paved areas will be scanned. A direct surface beta-gamma measurement is taken at or near the center of each survey location (grid block) identified for survey as described in paragraph 7.2.2.a,above.

c. Unpaved Areas

Unpaved areas are surface scanned over at least 10 percent of the area with a gamma detector (NaI or microrem). Emphasis is placed on traffic areas and roadways.

7.2.3 Affected Areas - Building Exteriors

No building exteriors have been identified which are classified as affected. If classified as affected, they are scan surveyed over 100 percent of the area. A minimum of 30 measurement locations are selected on each of wall surfaces and roofs or an average of one per 20 m^2, whichever is greater. The survey design should focus on likely areas for deposition, e.g., downwind of stacks, vents and drains.

7.2.4 Unaffected Areas - Building Exteriors

The survey of exterior surfaces of buildings and structures which are classified as unaffected is as follows:

a. Exterior Walls Surfaces

The survey of building exterior walls focuses on the area within two meters of the ground surface. The areas are surface scanned over a minimum of 10 percent of the area. Measurement locations are randomly selected for a minimum total of 30 or an average of one per 50 m^2, whichever is greater.

b. Roofs

The survey of building roofs which are classified as unaffected is limited to the roofs of the Power Block and immediately adjacent buildings. If any of these surfaces are classified as affected, they are surveyed in accordance with the guidelines in paragraph 7.2.3 above. Otherwise the roof areas are surveyed at a minimum of 30 location or one per 50 m^2, whichever is greater. Roofs of other buildings in outside areas which are classified as unaffected are not surveyed.

7.2.5 Soil Sampling

Soil samples will be collected from the site grounds for the Termination Survey. Thirty locations will be selected at random from the site. The soil samples will be collected from the top 15 cm of the soil column and will be analyzed as REMP samples. The analysis will include gamma spectroscopy for Co-60.

Additional soil samples are collected as needed to support the Termination Survey of plant systems, some of which extend beyond the Secured Area of the site. Soil samples may be collected, for example, from leach fields, soil adjacent to or below catchments, sumps and building foundations.

Any soil sample containing detectable concentrations of Co-60, or any other non-naturally occurring radionuclides above the REMP detection limit (LLD), will be investigated to determine the source of the contamination, as well as the areal extent and depth of the contamination.

7.3 Plant Systems

7.3.1 Selection of Measurement Locations

Surveys of systems are biased sampling surveys. Selection of measurement locations in both affected and unaffected systems focuses on the locations with the highest probability for contamination considering the system design and operational characteristics. Typically, the inlets, outlets, system interfaces, system crud traps and flow impingement areas are selected. A "location" in a system survey design is generally identified as an area which includes a surface defined by the interior of a component or of a discrete-specified length of piping in piping systems. A component includes any discrete element which is distinguishable from piping. When a component is removed or opened for access for Termination Survey measurements, typically three measurement locations are established: the component interior surfaces and the two piping spool pieces immediately adjacent. In non-piping systems such as ventilation systems, locations are identified as local areas where air impinges, such as inlet and outlet baffles and ducting interior surfaces at bends and cleanouts. The survey design should take advantage of low cost access methods including destructive cutting of piping (where permitted).

7.3.2 Breakdown Into Subunits

Systems, including mechanical systems, such as the main steam turbine, are broken down into subunits by functional areas, components or locations. Large vessels and heat exchangers are usually considered to be subunits of the system in which they are located. In piping systems, each fluid carrying interface to other systems is usually established as a subunit.

7.3.3 Survey Measurements

The interior surfaces of components selected for survey measurements are usually scanned over 100 percent of the accessible area. When possible, locations are selected such that both direct and removable surface contamination measurements can be taken. Gamma exposure rate measurements are not usually taken within systems, except where interiors are large enough for human occupancy, i.e., large tanks and spaces such as the

Condenser Hotwell. At least one direct surface and removable surface contamination measurement is taken at each location.

7.3.4 Measurement Frequency - Affected Systems

In general, the measurement frequency is strongly influenced by the size and complexity of a system. In affected systems, the design guideline is a minimum of 30 measurement locations. Exceptions may arise where the system is limited in physical extent such that 30 discrete locations cannot be identified.

7.3.5 Measurement Frequency - Unaffected Systems

In unaffected systems, a minimum of 10 measurement locations will be selected. Similarly, in unaffected systems, the number of measurements may be limited by the physical extent of the system, so exceptions may be made to the minimum number guideline.

7.3.6 Embedded Piping

Several piping systems, primarily Liquid Radioactive Waste (G11), and Fuel Pool Cooling and cleanup (G41) (P71), have substantial lengths of piping which are embedded in reinforced concrete, i.e., floors and reactor cavity areas. For management of the survey, each system with embedded piping may be divided into subunits for survey design and control purposes. Detailed surveys of embedded piping are performed with multiple GM detector assemblies, called pipe crawlers, inserted into embedded piping runs. All openings and normal access points (cleanouts) will also be surveyed. All runs of piping will be systematically surveyed over at least 25 percent of their length.

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ENCLOSURE 2

ADJUSTMENT FACTORS
for
TERMINATION SURVEY MEASUREMENTS

Adjustment Factors for Termination Survey Measurements

Summary

The principal release criteria guideline values for surface contamination from beta-gamma emitting radionuclides at Shoreham are the Regulatory Guide 1.86 limits of 5,000 dpm/100 cm^2 for average total surface activity and 1000 dpm/100 cm^2 for removable surface activity. For areas where certain other hard-to-measure radionuclides may be present, such as Fe-55 which decays by electron capture, special guideline values apply. When mixtures of radionuclides are involved whose members have different guideline values, gross activity guideline values and elevated area guideline values are substituted for guideline values. Direct beta-gamma measurements above the critical level [1] are multiplied by a "factor" (f) to account for the presence of the hard-to-measure species which may be present. Table 1 identifies the areas where special guideline values apply, and presents the values of gross activity guideline values and the adjustment factor values.

Table 1

Summary of Gross Activity Guideline Values and Adjustment Factors

Description	Gross Activity Guideline Values Total Surface Activity (dpm/100 cm^2)(1)		Adjustment Factor, f	
	Guideline Value	Elevated Area Guideline Value	Total β-γ	Removable β-γ
Bioshield Concrete	95,900	287,700	35.9	12.4 (2)
Bioshield Steel	76,900	230,700	24.2	1.2 (3)
D-S Storage Pool (Cutting Station)	11,100	33,300	2.6	2.6
Spent Fuel Storage Pool	9,400	28,200	2.3	2.3
Reactor Assembly (bottom head)	11,100	33,300	2.6	2.6
All Other (4) (balance-ofplant)	5,000	15,000	1.2	1.2

Table 1 Notes:

(1) Special guideline values have not been established for removable surface contamination. The guideline value of 1000 dpm/100 cm^2 applies.

(2) Residual activity from Tritium is not present in removable form in activated concrete.

(3) Activated steel is not considered to be removable. Any removable surface contamination measured on the Bioshield steel liners is assumed to be similar to balance-of-plant deposits.

(4) Gross activity guideline values are not applied to other areas, as special guideline values were not in effect for the majority of measurements in these areas.

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[1] The critical level L_c, is used to identify measurements which are "above" background. It is defined in the Shoreham Termination Survey Plan as the upper limit of the 95 % confidence interval of the mean value of the net count distribution. For this use the critical level is expressed in units of dpm/100 cm^2.

Radionuclide Composition

To compute gross activity guideline values and adjustment factors for direct surface activity measurements, radionuclide compositions were determined for the areas where special guidelines apply. For each of the areas, the most likely sources of residual contamination were identified. These are shown in column two of Table 2. The radionuclides which comprise residual radioactivity sources for each of the areas are shown in column 3 and the sources of data evaluated to estimate activity ratios are shown in column 4 of Table 2.

Table 2

Determination of Radionuclide Composition

Area Description	Source of Residual Contamination	Principal Radionuclides (1)	Data Sources (2)
Bioshield Concrete	neutron activated concrete	Fe-55, Co-60, H-3	**activation analysis**, core sample analysis
Bioshield Steel Liner	neutron activated steel	Fe-55, Co-60	**activation analysis**, core sample analysis
Dryer-Separator Pit Cutting Station	neutron activated vessel and vessel internals which were cut up in the cutting station	Fe-55, Co-60, Ni-63	**activation analysis**, Engineering records of materials processed
Spent Fuel Storage Pool	sediments derived from fuel, control rods and associated hardware stored in pool, primarily activated corrosion products	Fe-55, Co-60, Ni-63	**fuel pool cleanup system filter sample analysis**
Reactor Vessel Bottom Head	neutron activation of stainless steel cladding on inner surface, normal plant corrosion products on outside surface	Fe-55, Co-60, Ni-63	**activation analysis**, vessel cladding sample analysis

Table 2 Notes:

(1) Radionuclides listed include Co-60 and hard-to-measure species which comprise ≥ 5 % of the total activity. Eu-152 has also been reported at concentrations above 5 % of the total activity in Bioshield concrete samples. It is not accounted for separately, as it is detected by survey instrumentation. Any residual activity due to Eu-152 will be accounted for along with Co-60 and other detectable radionuclides against the guideline values for β-γ emitters.

(2) The principal source of information used to calculate activity compositions is highlighted in bold type.

The principal source of information on radionuclide composition of neutron activated materials of construction is neutron activation analysis reported in the Shoreham Site Characterization Report (LILCO90). However, additional information for Bioshield and Reactor Vessel materials is available from radiochemistry analysis of samples. These include samples of Bioshield and Reactor Pressure Vessel material collected during May 1994. Four samples of Bioshield steel, one of Reactor Pressure Vessel bottom head cladding material and two of Bioshield concrete were collected. Samples of Bioshield steel liner and concrete were sent to three separate laboratories for analysis. Each sample was analyzed for gamma emitters and Fe-55. Activation analysis results are the only source of information for the characterization of materials handled in the Dryer-Separator Pit Cutting Station. Spent Fuel Storage Pool residue was characterized using results from radiochemistry analysis of a Pool Cleanup System filter.

An evaluation was performed to determine the best data source for use in characterization of Bioshield steel and concrete and the Reactor Vessel Bottom Head. This evaluation is summarized in Attachment 1. The attachment summarizes the radiochemistry sample results and shows ratios of Fe-55:Co-60 activity derived from the Bioshield radiochemistry sample data. It is seen that Fe-55:Co-60 ratios for individual samples vary widely, ranging from 11 to 1,500,000 for Bioshield steel and from 1 to 2353 for Bioshield concrete. As an alternative method to estimate Fe-55:Co-60 ratios using radiochemistry results, ratios were obtained using calculated averages of Fe-55 and Co-60 concentrations. These composite Fe-55:Co-60 ratios were compared to Fe-55:Co-60 ratios obtained from neutron activation results. Good agreement is obtained for concrete (7.1 vs 11.4) and poor agreement is obtained for steel (577 vs 23) for the radiochemistry and neutron activation results, respectively. The principal limitation in the radiochemistry data is the high degree of uncertainty in the Fe-55 results. Two of the three laboratories reported no positive confirmation of Fe-55. In these cases, reported MDA values were used to calculate Fe-55:Co-60 ratios.

After a review of all the available data on Bioshield and Reactor Vessel Bottom Head radionuclide composition, it is concluded that the activation analysis results provide the most consistent basis for the characterization. This is also consistent with other radionuclide characterizations performed for Decommissioning purposes such as for Radioactive Waste shipment.

Calculation of Gross Activity Guideline Values

The gross activity guideline values are obtained from the "sum of fractions rule" in Appendix A of NUREG/CR-5849 (BE92), whereby the sum of the ratios of each radionuclide to its guideline value must be ≤ 1. The resulting formula for the GAG is:

$$GAG = \frac{1}{\frac{F_1}{G_1} + \frac{F_2}{G_2} + \ldots \frac{F_n}{G_n}}$$

where: GAG = Gross Activity Guideline Value in dpm/100 cm^2,

F_i = fraction of total activity due to the ith radionuclide,

G_i = guideline value for the ith radionuclide, in dpm/100 cm^2. The guideline values for each nuclide identified in Table 2 are: Co-60 = 5000; Fe-55 = 200000; H-3 = 200000; and Ni-63 = 5000 dpm/100 cm^2.

The calculations are summarized in Table 3. The radionuclide composition for each area is shown in columns 2 and 3. The factor F, in column 3, is the fraction of the total activity represented by each nuclide. The adjustment factor f, whose values are shown in column 4, is obtained as the quotient $1/F_{Co}$, where F_{Co} is the fraction of total activity due to Co-60 in each area. It is applied to direct activity measurements to account for the total activity which may be present. Column 5 shows the individual fractions, F_i/G_i, which appear in the denominator of the GAG equation. It also shows the sums of the fractions. The GAGs shown in column 5 are obtained as the reciprocal of the sum of the fractions.

Table 3

Gross Activity Guideline Value Calculations

Area	Nuclides	F	f	F_i/G_i	GAG (dpm/100 cm^2)
Bioshield Concrete (1)	Co-60 Fe-55 H-3	0.0278 0.3159 0.6559	35.9	0.00000557 0.00000158 0.00000328 sum 0.00001043	95923 (95900)
Bioshield Concrete Removable (1)	Co-60 Fe-55	0.0809 0.9181	12.4	0.00008090 0.00091810 sum 0.0009990	1001 (1000)
Bioshield Steel (1)	Co-60 Fe-55	0.0413 0.9487	24.2	0.00000826 0.00000474 sum 0.00001300	76902 (76900)
Dry Cutting Station (1)	Co-60 Fe-55 Ni-63	0.3840 0.5635 0.0517	2.6	0.00007680 0.00000282 0.00001034 sum 0.00008996	11116 (11100)
Spent Fuel Pool (2)	Co-60 Fe-55 Ni-63	0.4324 0.4764 0.0868	2.3	0.00008648 0.00000238 0.00001736 sum 0.00010622	9414 (9400)
Vessel Bottom Head (1)	Co-60 Fe-55 Ni-63	0.3858 0.5617 0.0521	2.6	0.00007716 0.00000281 0.00001042 sum 0.00009039	11063 (11100)
All Other	Co-60 Fe-55	0.830 0.170	1.2	0.00016600 0.00000085 sum 0.00016685	5993 (5900)

Table 3 Notes:

(1) Data from Attachment 2.

(2) Data from Attachment 3.

(3) Gross activity Guideline values rounded down to nearest 100 dpm/100 cm^2.

(4) Minor constituents not included (F_i < 5 %).

References

BE92, J. D. Berger, "Manual for Conducting Radiological Surveys in Support of License Termination", prepared for U. S. Nuclear Regulatory Commission, NUREG/CR-5849 Draft, June, 1992.

LILCO90, Long Island Lighting Company, "Shoreham Nuclear Power Station Site Characterization Program Final Report", May 1990 (Addendum 1, June 1990; Addendum 2, October 1990; Addendum 3, June 1992).

List of Attachments

1. Evaluation of Fe-55:Co-60 Ratios in Bioshield and RPV Materials.
2. Activity Ratio Calculations for Termination Survey Special Areas
3. Scientech, Inc. "Radioactive Sample Analysis Report; Shoreham SFSP Paper Filter", 5/27/94.

I. Vendor Sample Radiochemistry Analysis Results

Location	Scientech	Teledyne	INEL
Fe-55 pCi/gm			
D8 Inner Liner	757	100 *	600 **
D8 Outer Liner	110 *	200 *	1500 **
D8 Concrete	30	10 *	30 **
D11 Inner Liner	12300	300 *	1000 **
D11 Outer Liner	353	500 *	1000 **
D11 Concrete	45	8 *	12 **
Vessel Bowl Shavings	5960	300 *	
Co-60 pCi/gm			
D8 Inner Liner	1.37	8.83	1.26
D8 Outer Liner	0.16	1.85	0.001
D8 Concrete	0.39	5.33	0.05
D11 Inner Liner	1.87	1.05	1.58
D11 Outer Liner	0.06 *	0.001 *	0.005
D11 Concrete	14.3	5.85	0.0051
Vessel Bowl Shavings	1.85	2.92	

II. Fe-55:Co-60 Ratios Obtained From Radiochemistry Results

Location	Scientech	Teledyne	INEL
D8 Inner Liner	553	11	476
D8 Outer Liner	688	108	1500000
D8 Concrete	77	2	600
D11 Inner Liner	6578	286	633
D11 Outer Liner	5883	500000	200000
D11 Concrete	3	1	2353
Vessel Bowl Shavings	3222		

Notes: * Results less than reported MDA, MDA values used to calculate Fe-55:Co-60 ratios.

** Uncertainty > ± 100 % and not reported as isotope found. The MDA values shown are used to calculate Fe-55:Co-60 ratios.

III. Comparison Of Fe:Co Ratios From Radiochemistry and Activation Analysis of Bioshield Materials

	Fe-55 pCi/gm		Co-60 pCi/gm	
Source	Steel	Concrete	Steel	Concrete
Scientech D8-IL	757		1.37	
Scientech D8-OL	110		0.1600	
Scientech D8-Conc		30.0		0.39
Scientech D11-IL	12300		1.87	
Scientech D11-OL	353		0.06	
Scientech D11-Conc		45.0		14.3
Teledyne D8-IL	100		8.83	
Teledyne D8-OL	200		1.85	
Teledyne D8-Conc		10		5.33
Teledyne D11-IL	300		1.05	
Teledyne D11-OL	500		0.001	
Teledyne D11-Conc		8		5.85
INEL D8-IL	600		1.26	
INEL D8-OL	1500		0.001	
INEL D8-Conc		30		0.05
INEL D11-IL	1000		1.58	
INEL D11-OL	1000		0.005	
INEL D11-Conc		12		0.0051
Trimmed Mean *	632	20.5	1.022	2.905
Std Dev	435	9.5	0.76	3.11
Activation Analysis Results pCi/gm				
Calculated Avg	1467.5	27.6	63.9	2.43
Estimated Error **	1467	27.6	63.9	2.43

Fe-55:Co-60 Ratios	Steel			Concrete		
	Mean		Std Dev	Mean		Std Dev
Radiochemistry	618.4	±	459.9	7.1	±	7.6
Activation Analysis	23.0	±	23.0	11.4	±	11.4

Notes: *Trimmed mean is obtained from data set with high and low values eliminated.

** Error of activation analysis concentrations estimated to be 100 %.

Part I: Bioshield

A. Total activity - in Curies on July 1, 1990

	H-3	C-14	Fe-55	Co-60	Ni-59	Ni-63	Total
Concrete	0.00968	0.0000019	0.0108	0.000555	0.000000	0.000002	0.02103
Steel	0.00098	0.0000018	0.371	0.00942	0.000004	0.000578	0.38198

B. Activity decayed to July 1, 1994

Concrete	0.007721	0.0000018	0.00371876	0.000327	0.000000	0.000002	0.01177
Steel	0.000781	0.0000017	0.12774654	0.005561	0.000004	0.000560	0.13465

C. July 1, 1994 activity concentration in pCi/gm

	Mass gm							
Concrete	134650668	57.342	0.014	27.618	2.433	0.000	0.018	87.4248
Steel	87049350	8.980	0.021	1467.519	63.887	0.048	6.443	1546.89

D. Activity ratios as of July 1994 (individual isotope pci/gm over total pCi/gm)

Concrete	0.655899	0.0001596	0.31590419	0.027834	0.000001	0.000201	1
Steel	0.005805	0.0000132	0.94868534	0.041300	0.000031	0.004165	1

E. Activity ratio for concrete removable activity only (assumes H-3 not available)

July 1994 activity	0.0139552	27.6178909	2.433396	0.000131	0.017583	30.0829
Activity ratios	0.000464	0.918058	0.080890	0.000004	0.000584	1

Notes:

1. Activity data from Shoreham Site Characterization report; TLG report L01-22-001, April, 1990.
2. Eu-512 not included in concrete activity data - calculated to be < 0.002 of total activity.
3. Tritium in activiated concrete is bound in water of hydration and not considered to be removable.

Attachment 2

Part II: Cutting Station

A. Total Activity - in Curies on July 1, 1990

Component	Weight-gm	H-3	C-14	Fe-55	Co-60	Ni-59	Ni-63	Total
Core shroud	22888316	0.0381	0.0043	118.662	47.3917	0.0283	3.9	170.024
Jet Pumps	7409731	0.0018	0.0002	5.5189	2.2043	0.0013	0.1815	7.908
Core Spray Sparg	380065	0.000000	0.0000000	0.00102	0.000408	0.000000	0.000033	0.001
Top Guide Plate	4976868	0.0744	0.0084	232.1502	93.6201	0.0553	7.6298	333.538
Core Support	5891167	0.0017	0.0002	5.2119	2.0817	0.0012	0.1714	7.468
Vessel Clad	4411866	0.000029	0.0000033	0.0921	0.0369	0.000021	0.00303	0.132
Vessel Wall	111216810	0.000165	0.0000016	0.328	0.0115	0.000003	0.000428	0.340
Steam Dryer	25909090			0.0125	0.0096			0.022
Moist Separator	49895454			0.0521	0.0401			0.092
Total	232979367							519.526

B. Activity decayed to July 1, 1994 multiplied by weight fraction of component.

Component	Wt_fraction	H-3	C-14	Fe-55	Co-60	Ni-59	Ni-63	Total
Core shroud	0.0982	0.00299	0.00042	4.01405	2.74870	0.00278	0.37177	7.141
Jet Pumps	0.0318	0.00005	0.00001	0.06044	0.04139	0.00004	0.00560	0.108
Core Spray Sparg	0.0016	0.00000	0.00000	0.000001	0.000000	0.00000	0.00000	0.000
Top Guide Plate	0.0214	0.00127	0.00018	1.70759	1.18069	0.00118	0.15815	3.049
Core Support	0.0253	0.00003	0.00001	0.04538	0.03108	0.00003	0.00421	0.081
Vessel Clad	0.0189	0.00000	0.00000	0.00060	0.00041	0.00000	0.00006	0.001
Vessel Wall	0.4774	0.00006	0.00000	0.05391	0.00324	0.00000	0.00020	0.057
Steam Dryer	0.1112	0.00000	0.00000	0.00048	0.00063	0.00000	0.00000	0.001
Moist Separator	0.2142	0.00000	0.00000	0.00384	0.00507	0.00000	0.00000	0.009
Totals	1	0.00440	0.00061	5.88630	4.01120	0.00404	0.53998	10.447

C. Activity Ratios as of July 1994

Fraction of total	0.000421	0.000059	0.563469	0.383975	0.000386	0.051690	1

Notes : 1. Includes all items cut up in the dryer separator pit except CRB XY rack. The contribution of the rack is negligible because it's weight fraction is 0.03 and the Fe:Co ratio is approxately 1.

2. Activity data from Shoreham Site Characterization Report, primarily April 1990 TLG report L01-22-001.
3. Moisture Separator & Steam Dryer activity all due to surface contamination; Fe-55 estimated as 1.3 x Co-60.
4. Surface activity included in the activity shown for each component if data avaliable from Site Char Report.
5. Mn-54 not included due to limited data and short half-life (312 d).

Attachment 2

Part III: Vessel Cladding

	H-3	C-14	Fe-55	Co-60	Ni-59	Ni-63	Total
July 1990 Activity (Ci)	0.000029	0.0000033	0.0921	0.0369	0.000021	0.00303	0.132
July, 1994 Activity (Ci)	0.00002	0.000003	0.0317	0.0218	0.00002	0.0029	0.0565
Fraction of total	0.00042	0.00006	0.56169	0.38585	0.00039	0.05207	1

Note: Activity data from Shoreham Site Characterization Report; April 1990 TLG report L01-22-001.

Part IV: Fuel Pool

	Cs-137	Mn-54	Fe-55	Co-60	Co-57	Ni-63	Total
May 1994 Activity (μCi/gm)	0.000055	0.0000779	0.0325	0.0295	0.000168	0.00592	0.06822
Fraction of total	0.00081	0.00114	0.47639	0.43242	0.00246	0.08678	1

Note: Activity data from Scientech Interim Radioactive Sample Analysis Report, 5/01/94; not corrected for decay.

Part V: Decay Factors	H-3	C-14	Fe-55	Co-60	Ni-59	Ni-63
Half-life (yr)	12.26	5720	2.6	5.26	80000	92
Decay Factor	0.798	1.000	0.344	0.590	1.000	0.970

Note: Decay period is four years - from July 1, 1990 to July 1, 1994.

Attachment 2

Attachment 3

SCIENTECH, INC.

205 PERRY PARKWAY, #10 ■ GAITHERSBURG, MD 20877 ■ PHONE: 301-977-4480 ■ FAX: 301-840-2182

Radioactive Sample Analysis Report

Plant Name	:	Shoreham
Sample Type	:	SFSP PAPER FILTER
Reference Date/Time	:	05/01/94 12:00
Receipt Date	:	05/06/94
Reporting Date	:	05/27/94
Sample Number	:	24017
Plant Sample ID	:	94A-05-014
Volume/Weight	:	2.6730E+02 gram
Purchase Order Number	:	C93L1099
Project Number	:	1-011-92-214

Measured Concentration (uCi/gram)

Nuclide	Value	% Unc.	Nuclide	Value	% Unc.
H-3	<6.6E-05		Ru/Rh-106	<1.1E-04	
C-14	<1.1E-06		Ag-108m	<1.9E-05	
Cr-51	<1.7E-04		Ag-110m	<3.1E-05	
Mn-54	7.79E-05	33	Sn-113	<2.6E-05	
Fe-55	3.25E-02	15	Sn-117m	<1.4E-05	
Co-57	1.68E-05	32	Sb-124	<2.8E-05	
Co-58	<4.3E-05		Sb-125	<6.0E-05	
Fe-59	<5.9E-05		I-129	<1.6E-05	
Ni-59	<1.3E-03		I-131	<3.8E-05	
Co-60	2.95E-02	15	Cs-134	<2.7E-05	
Ni-63	5.92E-03	10	Cs-136	<4.6E-05	
Zn-65	<1.0E-04		Cs-137	5.52E-05	38
Se-75	<1.3E-05		Ce-139	<1.0E-05	
Sr-85	<2.4E-05		Ba/La-140	<2.1E-05	
Y-88	<1.3E-05		Ce-141	<1.9E-05	
Sr-89	<2.4E-06		Ce/Pr-144	<7.0E-05	
Sr-90	<2.3E-06		Eu-152	<3.3E-05	
Nb-94	<3.1E-05		Eu-154	<1.7E-05	
Nb-95	<4.0E-05		Eu-155	<2.7E-05	
Zr-95	<6.7E-05		Np-237/Pu-242	1.04E-07	66
Tc-99	6.24E-06	25	Pu-238	4.37E-07	60
Ru-103	<2.7E-05		Pu-239/240	2.74E-07	33
			Am-241	<1.4E-06	
			Pu-241	<2.7E-05	
			Gross Alpha	<4.0E-06	
			Cm-242	<3.7E-07	
			Cm-243/244	<1.1E-06	

Radiometrics Manager _____

Laboratory Manager _____

The data contained in this report were produced and documented in accordance with approved quality control and quality assurance procedures. All of the results are decay corrected to the reference date listed above. Indicated errors are two (2) standard deviations based on counting statistics only.

FAXED 5-27-94