3.0 IDENTIFICATION OF REMAINING DISMANTLEMENT ACTIVITIES

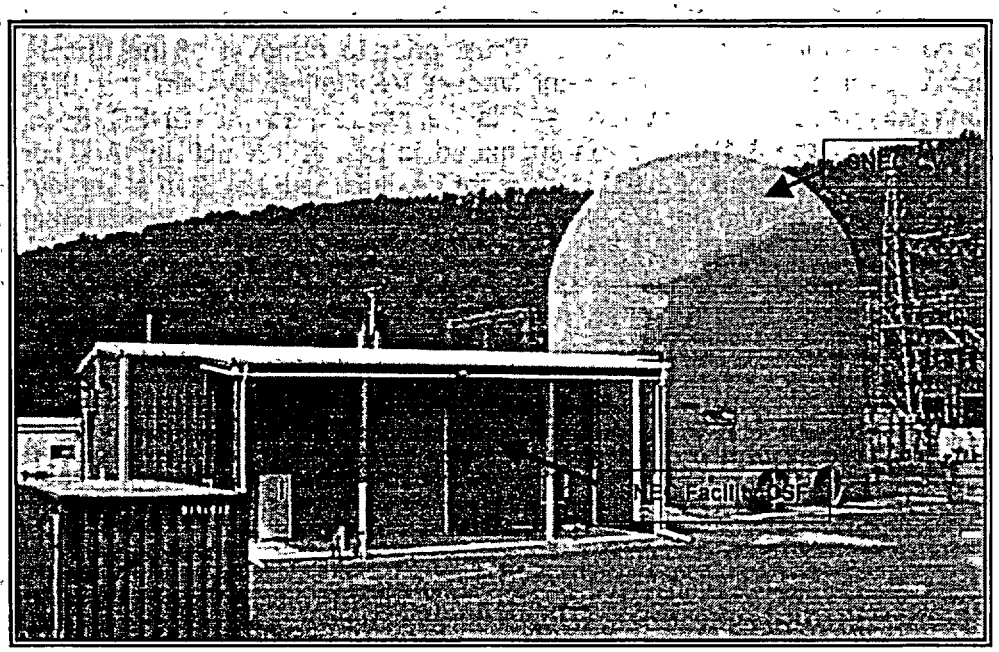
3.1 INTRODUCTION

As discussed previously, this final phase of the SNEC Facility Decommissioning Project commenced in April 1998 following NRC approval of a License Amendment authorizing decommissioning. Since that time, the greater part of 1998 was devoted to removing and shipping the SNEC Facility Large Components (i.e. Reactor Vessel, Pressurizer and Steam Generator). Following removal of the large components, the latter part of 1998 and the beginning of 1999 were devoted to removing the bulk of the remainder of the systems from the SNEC Facility. Following that, concrete decontamination commenced and continued into 2000 when it was determined that complete removal of the concrete from the Containment Vessel (CV) would be required in order to release the site in accordance with NRC Regulations (10 CFR Part 20.1402, Radiological Criteria for Unrestricted Use). Concrete removal is ongoing.

As described in Section 2.2.4 the site is divided into eight areas in describing radiological conditions. These same area designations will be used in describing remaining site dismantlement activities.

FIGURE 3-1

Photograph of SNEC Facility CV & DSF Buildings



3.2 REMAINING TASKS

3.2.1 Area 1 (CV Basement) Remaining Site Dismantlement Activities

Area 1 will be completely removed to the steel CV liner.

3.2.2 Area 2 (Primary Compartment) Remaining Site Dismantlement Tasks

Area 2 will be completely removed to the steel CV liner.

3.2.3 Area 3 (Auxiliary Compartment) Remaining Site Dismantlement Activities

Area 3 will be completely removed to the steel CV liner.

3.2.4 Area 4 (Operating Floor) Remaining Site Dismantlement Activities

Remaining tasks in Area 4 include removal of miscellaneous steel, removal of the polar crane and its girders and disconnection of the Containment Ventilation System. Additionally, some volumetrically contaminated concrete remains to be removed.

The polar crane is the last large component to be removed from the SNEC Facility. Plans for Polar Crane removal have not been finalized. It can be removed using conventional cutting and rigging techniques and will most likely be dispositioned as radioactive waste.

The Containment Vessel and the Containment Vessel Ventilation System are special cases. The SNEC Facility License requires the interior of the Containment Vessel to be maintained as an exclusion area and the ventilation system to be operating during activities that have the potential to cause a measurable release. Thus, prior to removal of the upper dome of the Containment Vessel and the Containment Vessel Ventilation System the SNEC Technical Specifications will need to be revised. A change request to the SNEC Facility Technical Specifications (Reference 3-1) has been submitted to permit this activity. Additionally, prior to the removal of the upper dome of the Containment Vessel, the Final Status Survey (FSS) for the bottom head of the Containment Vessel will be completed and the area backfilled prior to removal of the upper dome. During removal of the upper dome, the backfilled portion of the lower Containment Vessel will be protected by an impermeable membrane and the area resurveyed after completion of the removal of the upper dome.

3.2.5 Area 5 (Pipe Tunnel) Remaining Site Dismantlement Activities

As described in Section 2.2.2, Area 5 consisted of a concrete tunnel, which extended 235 degrees around the CV. The tunnel has been removed except for the portion under the Material Handling Bay (MHB) section of the Decommissioning Support Facility.

The portion of the tunnel under the MHB will be removed and the area surveyed and remediated as appropriate to meet site release levels.

3.2.6 Area 6 (Reactor Compartment and Storage Well) Remaining Site Dismantlement Activities

Area 6 will be completely removed to the steel CV liner.

3.2.7 Area 7 (Exterior of Containment Vessel Dome) Remaining Site Dismantlement Activities

Remaining tasks in Area 7 consist of removing the CV Dome to at least three feet below grade and completing the requisite Final Status Survey on the remaining below grade portions of the CV. Remediation, if needed, will be performed as appropriate.

3.2.8 Area 8 (SNEC Facility Yard Areas) Remaining Site Dismantlement Activities

Remaining tasks in Area 8, with the exception of the Saxton Steam Generating Station (SSGS) area will consist of some soil removal and removal of utilities located outside the CV.

The SSGS was used by the SNEC Facility to support its operation, generate electricity and to provide dilution flow for radioactive effluent liquid discharges. Surveys in this area indicate there was some residual contamination that needed to be removed. This included water and sediment removal, some concrete decontamination, and extensive embedded piping surveying and removal. These activities were completed in 2001. The small amount of remaining embedded piping will be addressed during the FSS design process.

3.3 ACCESS CONTROL MEASURES

Since all decommissioning activities may not be completed prior to the start of final status survey work, measures as described in Section 5.3 will be implemented to protect survey areas from recontamination during and after the final status survey. In all cases, decommissioning activities creating a potential for the spread of contamination will be completed within each survey area prior to starting a final status survey in the area. Additionally, decommissioning activities that create a potential for the spread of contamination to adjacent survey areas will be evaluated and controlled using barriers, covers, or restricting or rescheduling activities.

3.4 10 CFR 50.59 REVIEW

10 CFR 50.59 "Changes, Tests and Experiments" permits licensees to make changes to the facility as described in the Safety Analysis Report (as updated), make changes in procedures as described in the Safety Analysis Report (as updated), and conduct tests or experiments not described in the Safety Analysis Report (as updated) without obtaining a license amendment under certain conditions.

With respect to the SNEC Facility this License Termination Plan serves to support a License Amendment to perform the Final Status Survey and terminate the license. In addition as indicated in Section 3.2.4 above a change request to the SNEC Facility Technical Specifications (Reference 3-1) has been submitted to allow removal of the upper Containment Vessel dome and the Containment Vessel Ventilation System. The remaining tasks described in Section 3.2 are bounded by the Definition for Decommissioning Activities in the SNEC Facility Technical Specifications (Reference 3-3) and are thus permitted by the Technical Specifications. Additionally these tasks have been reviewed against the criteria of 10 CFR 50.59 and are bounded by the SNEC Facility USAR (Reference 3-4).

Thus, although some changes will be made to the facility as described in the SNEC Facility USAR, e.g. Removal of the Polar Crane, these activities were anticipated to be performed. Further, greater than 99% of the curies present at the SNEC Facility at the commencement of decommissioning in 1998 have been safely removed from the site and shipped for proper

disposition. Finally, each activity requiring a procedure as defined by the SNEC Facility Technical Specifications is reviewed to ensure it satisfies the criteria of 10 CFR 50.59. Thus, with the exception of the removal of the Containment Vessel upper dome and the Containment Vessel Ventilation System,, none of the remaining tasks identified in Section 3.2 requires prior NRC approval in accordance with 10 CFR 50.59.

3.5 DECOMMISSIONING TASKS REQUIRING COORDINATION WITH OTHER FEDERAL OR STATE REGULATORY AGENCY

Prior to License Termination, tasks requiring coordination with any other Federal or State Regulatory Agency include disposal of the water in Area 8. Additionally, some water treatment will be necessary for water collecting in excavated areas e.g., the yard area North of the CV. Discharge of this water will be controlled radiologically in accordance with the requirements of the ODCM (Reference 3-2). Non-radioactive sampling and analysis will be in accordance with a program accepted by the Commonwealth of Pennsylvania. Additionally, the Army Corps of Engineers gave its consent for locating temporary on-site excavations and infiltration galleries.

3.6 SUMMARY

The tasks remaining for site dismantlement are permitted by the SNEC Facility License Technical Specifications and USAR. Based on current estimates, these tasks will be completed in the second quarter of 2003 for an estimated exposure of approximately 37 person-rem (Table 3.1). Of this total approximately 35 person-rem has been expended to date. Estimated remaining low level radioactive waste generation is provided in Table 3.2.

3.7 REFERENCES

3-1 SNEC Facility Technical Specification Change Request No. 62, April 2002.

3-2 SNEC Procedure E900-PLN-4542.08 "Off-Site Dose Calculation Manual"

3-3 SNEC Facility Technical Specifications

3-4 Updated Safety Analysis Report for Decommissioning the SNEC Facility

Table 3.1

SNEC Facility Decommissioning Person-Rem Estimate

Task	Person-Rem
Asbestos Remediation	2.97
System Dismantlement	12.83
Large Component Removal	7.38
Structure D&D	4.17
Waste Management	2.13
Miscellaneous Support Activities	2.41
Scaffold & Shielding	5.32
Characterization Activities	0.63
Total	**37.84**

Table 3.2

SNEC Facility Low Level Radioactive Waste Projection

Type	Quantity
Metal	50,000 lbs.
Soil	500 ft^3
Water	1600 Gal
Sediment	50 ft^3
Concrete	5,750,000 lbs.
Dry Active Waste (DAW)	2000 ft^3
Debris	15,000 lbs.

4.0 REMEDIATION PLANS

4.1 INTRODUCTION

The ultimate goal of the decommissioning project at the SNEC Facility is to release the site in accordance with NRC regulations (10 CFR Part 20.1402, Radiological Criteria for Unrestricted Use). Release of the site requires assurance that any future use of the site will not result in exposing individuals to unacceptable levels of radiation. The site release criteria require the Total Effective Dose Equivalent (TEDE) to an average member of the critical population group from residual contamination to be less than 25 mrem/year. In addition, plant-related contamination in groundwater and surface water with the potential to be used as a source of drinking water will be evaluated against the 4 mrem/year dose criteria in the National Primary Drinking Water Standards contained in 40 CFR 141. Remediation of some areas of the site will be necessary in order to meet these release criteria. Chapter 2 of the LTP provides information on the radionuclides present at the SNEC Facility as well as their location, activity and radiation levels. Section 5.4.4 of the LTP describes the data investigation process used to evaluate the effectiveness of remediation to determine whether further investigation or remediation is required to ensure that an area meets the site release criteria. Section 4.3 and 4.4 identify the remediation methods that may be used and describe the areas on site that may be subject to remediation

4.2 RADIOLOGICAL CONTROL PROGRAM CHANGES

The Radiological Controls Program will continue to be fully integrated into all aspects of operations at the SNEC Facility throughout the license termination process. The program will ensure that operations at the SNEC Facility are performed in accordance with the ALARA philosophy of protecting plant personnel and the general public.

The SNEC Radiation Protection Plan establishes basic policies, philosophies, and objectives to maintain doses to workers and members of the public as low as reasonably achievable (ALARA). Specific details as to how the plan is implemented are incorporated into Radiological Controls procedures. These procedures are established and will be maintained in accordance with the SNEC Facility Technical Specifications. GPU Nuclear does not anticipate significant changes to Radiological Controls procedures during the license termination process. However, any changes will be performed in accordance with the SNEC Facility Technical Specifications and 10 CFR 50.59 process.

4.3 REMEDIATION METHODS

In conjunction with the near completion of Site Dismantlement Activities as described in Chapter 3, remediation will be performed in accordance with the general decontamination and dismantlement considerations of Section 5.0 of the SNEC Updated Safety Analysis Report (Reference 4-1). Any area of the site that does not meet the release criteria will be remediated as appropriate. In keeping with the principles of ALARA, additional residual levels of radioactivity may be reduced commensurate with the minimization of total risk. As an example, because of the effectiveness of some remediation methods, some areas may be remediated to residual dose levels well below 25 mrem/year at little or no additional cost.

4.3.1 Building and Structure Surfaces

Many decontamination methods exist to decontaminate surfaces of buildings and structures. Several factors determine the choice of decontamination method for a given application, including the extent of the contaminated area, surface material, depth of contamination, and access considerations. Typical technologies that may be used are listed below. All methods will be utilized with proper regard to the potential for airborne contamination and industrial safety, and may be used in conjunction with containment enclosures or HEPA filtration units in order to minimize airborne radioactivity (as appropriate).

- Abrasive Vacuum Blasting with Recyclable Blast Media - This method is highly effective for surface contamination up to a depth of 6 mm. Unlike conventional abrasive blasting, this method uses a recyclable blast media such as steel shot to minimize the volume of waste that is generated.

- Concrete Sectioning - This method can be used to remove large pieces of contaminated concrete where other methods would be too slow. Cutting may be performed with an abrasive blade or a diamond wire saw.

- Needle Gunning - This method is used for removing surface contamination in areas that are too small or inaccessible for other methods.

- Manual Removal of Building/Structural Materials - Contaminated building materials such as steel and loose concrete will be cut and manually removed and disposed of at an appropriate low level radioactive waste burial or treatment facility.

- Concrete Scabbling - Contaminated concrete structures are routinely scabbled to remove surface activity deposits. Scabbling involves mechanical pounding of the surface to remove surface deposits and some subsurface contamination.

4.3.2 Surface Soils, Gravel and Asphalt

Soil remediation will involve the removal of soil, gravel or asphalt as necessary to meet the site release criteria. Asphalt and gravel will be treated in the same way as soil. Soil, gravel and asphalt will be removed with excavation equipment, and due care will be taken to prevent the spread of contamination during excavation and handling, including controls to minimize the creation of fugitive dust materials. All radiologically contaminated soil, gravel and asphalt determined to be above release criteria will be disposed of at a low-level radioactive material processing or disposal facility.

4.3.3 Sediment

As discussed in Chapter 3, sediment in the SSGS was removed as part of site dismantlement activities. No additional sediment contamination has been found that exceeds the release criteria, including sediment impacted by licensed discharges monitored as part of the SNEC Radiological Environmental Monitoring Program (REMP). Any additional sediment found to be contaminated from plant operations will be removed if the levels of contamination exceed the release criteria. Contaminated sediment may be removed by dredging or by other manual excavation techniques.

4.3.4 Sub-floor Soil

Normally, when radionuclide concentrations in sub-floor soil or beneath structures are found that exceed the applicable release criteria, the concrete floors or structures will be cut or broken up and the underlying soil excavated and removed along with any contaminated structural material. However, because of the difficulty in excavating beneath some existing structures, remediation of sub-floor soil may take place after the structure has been demolished. Remediation may also be performed with the structure intact. Several factors influence the approach that is used, including structural stability issues, groundwater level, the extent and depth of the contamination, the accessibility of the work area to excavation equipment, and the need to protect the work area from the elements as well as the safety of the workers. If it is necessary to first demolish a structure before accessing the below grade soil or other contaminated materials, the part that is above grade and the accessible parts below grade (interior floors, walls, ceilings) will be decontaminated first (as deemed appropriate). A final status survey will then be performed on these areas to verify that release criteria have been met. NRC approval of the results will release the surveyed parts of the structure for conventional demolition and disposal. After the debris is removed, radiological controls will be re-established and the radiologically contaminated sub-floor soil will be excavated as necessary.

4.4 AREAS TO BE REMEDIATED

Site characterization to determine remaining areas that need remediation is ongoing and will continue throughout decommissioning. The site release criteria and the principles of ALARA will be applied on a case by case basis to identify the extent of the remediation necessary. The following areas have either already been remediated, identified for remediation, or may undergo further analysis as deemed appropriate:

- All Outdoor Radioactive Material Storage Areas - These areas may change as site land use needs change. Initially, all Potentially Contaminated Areas (PCA) will be surveyed and be shown to meet site release criteria or be remediated and then surveyed prior to site release. In addition, any subsurface area previously used for underground liquid radioactive waste storage will be sampled and surveyed as appropriate.

- Soil and/or Gravel Under or Near Existing Site Facilities - Some contaminated materials have already been removed from beneath site structures during the soil remediation project. Additional soil and/or gravel will be remediated as necessary if future site characterization finds contamination in excess of the site release criteria. In addition, soil deposits around or under existing structures, such as the old steam tunnel and concrete slabs exterior to the CV, will be sampled, analyzed, and remediated as necessary during dismantlement operations.

- Structural Decontamination - Buildings and structures which have been identified for remediation include the Containment Vessel, Steam Tunnel, Saxton Steam Generating Station and surrounding concrete slabs. If additional structures are found to be contaminated in excess of the site release criteria, they will be decontaminated when they are no longer required to support site decommissioning operations.

- Storm Drain Out-fall – Section 2.0 in the SNEC Facility Site Characterization Report describes various contamination events that have occurred over the course of the plant's life that may have resulted in discharges of low levels of radioactivity. However, no event is listed that describes above permissible release of liquid to the storm drain and weir system. In addition, the weir and 10" drain line emptying to the river have been removed.

4.5 REFERENCES

4-1 Updated Safety Analysis Report for Decommissioning the SNEC Facility – Revision 2, February 1998.

5.0 SNEC FACILITY FINAL STATUS SURVEY PLAN

5.1 INTRODUCTION

The SNEC Facility Final Status Survey Plan (FSSP) has been prepared using the guidance provided in applicable regulatory guidance documents described in Section 5.1.1 below. Ultimately, this plan will be used to develop lower tier procedures and/or work instructions to accomplish the Final Status Survey for the SNEC Facility.

5.1.1 Purpose

The FSSP describes the final survey process that will be used to demonstrate that the SNEC Facility and all additional near site impacted areas meet radiological criteria for license termination. 10 CFR 50.82(a)(9)(ii)(D) (Reference 5-1), and Regulatory Guide 1.179 (Reference 5-2) have been used as guides in the preparation of this plan. This plan incorporates the site release criteria provided in 10 CFR 20.1402 (Reference 5-3) and addresses concerns of NUREG-1727, the NMSS Decommissioning Standard Review Plan, (Reference 5-4), and NUREG-1505 (Reference 5-6). Other documents, such as Draft NUREG-1549 (Reference 5-9), were also reviewed in the process of preparing this plan.

5.1.2 Scope

The final site survey will encompass structures, land areas, and any remaining facility systems which, because of licensed activities, were originally contaminated or had the potential to be contaminated. Areas that exhibited the highest contamination levels were located within the SNEC Containment Vessel (CV), as illustrated in Chapter 2 of this License Termination Plan (LTP). As of the date of the SNEC Facility LTP submittal, the majority of all contaminated systems, components, and soils will have been removed from the site. Continued remediation in selected areas will ensure these areas satisfy unrestricted release criteria before the Final Status Survey (FSS) process begins.

5.1.3 Summary

The SNEC Facility FSSP describes the final survey process and the methodology used to develop guideline values against which residual radioactivity levels remaining at the SNEC Facility at the time of the FSS will be compared. The final survey process is described as a series of steps – survey preparation, survey design, data collection, data assessment, and final survey report preparation. However, in practice, this is an iterative process in that once the results from one step are known they may prompt repeating one or more previous steps. In addition, the process is designed to be flexible in that modifications to the survey process may be made as more information is collected.

FSS activities begin when dismantlement and decontamination activities are believed to be complete. Each survey area is divided into survey units that are classified according to their potential for retaining residual radioactivity, or in accordance with known contamination levels. Survey data collected from each survey unit are collected according to data collection requirements and frequencies established for each classification. When residual radioactivity is measured above pre-set levels, an investigation is performed. Based on the results of the investigation, the survey unit may be remediated, reclassified, resurveyed or determined to meet the release criteria

There are three principal types of survey results collected during the FSS effort. They are 1) scan measurement data, 2) fixed-point measurement data, and 3) sampling of volumetric materials for laboratory analysis. *In-situ* gamma-ray spectrometry may also be included in the release survey process as well as the results of any special measurements or analysis. Statistical testing criteria for special measurements will be applicable to the survey methods used. All collected data are first verified to be of adequate quantity and quality, capable of supporting underlying assumptions necessary for statistical testing. Where necessary, previous survey steps are re-evaluated. Each survey unit will then be tested and compared to the release criteria. To meet the release criteria, the survey data must pass the statistical tests applied. When the data set fails statistical testing criteria, the survey unit is not suitable for unrestricted release without further actions.

Upon completion of FSS activities, a final survey report will be prepared which summarizes the data. The report will document the conclusion that the SNEC Facility and near site areas meet the 10 CFR 20.1402 release criteria and may be released for unrestricted use.

5.2 SURVEY OVERVIEW

This section describes the scope and methodology of the final survey process. It includes quality assurance measures and access control procedures. It also describes how implementation of this plan will demonstrate that the remaining structures and site areas meet the 10 CFR 20.1402 criteria for unrestricted release. Also described herein, are the methods used to develop guideline values against which residual radioactivity levels will be compared.

5.2.1 Identity of Radiological Contaminants

The radionuclide inventory at the SNEC Facility was estimated during the initial site characterization process, which was conducted between 1995 and 1996. Those data are compiled in the SNEC Facility Site Characterization Report (Reference 5-7). Station Work Instructions, site procedures, and Survey Requests have since been used to collect additional site characterization data. This more recently collected information is summarized in Chapter 2 of this plan. All of the data were reviewed and a final radionuclide listing was developed. Refer to Chapter 6, Section 6.2.2.3.

5.2.2 Site Release Criteria

5.2.2.1 Radiological Criteria for Unrestricted Use

These site release criteria correspond to the radiological criteria for unrestricted use given in 10 CFR 20.1402, which are:

- Dose Standard

  Residual radioactivity, distinguishable from background radiation and resulting in a Total Effective Dose Equivalent (TEDE) to an average member of the critical group will not exceed 25 mrem/y, including that from groundwater sources of drinking water.

- ALARA Standard

  Residual radioactivity will be reduced to levels that are As Low As Reasonably Achievable (ALARA), as addressed in Section 6.4.

5.2.2.2 Conditions Satisfying the Site Release Criteria

Levels of residual radioactivity that correspond to the allowable radiation dose and ALARA levels are calculated (derived) by analysis of various scenarios and pathways (e.g., direct radiation, inhalation, ingestion, etc.). These derived levels are referred to as "derived concentration guideline levels" (DCGLs), and form the basis for the following conditions which if met, satisfy the site release criteria (DCGL is defined in Section 5.8):

- All measurements of residual radioactivity above background from all survey units are equal to or below the $DCGL_W$*. No further testing is required, or
- Individual measurements from small areas within a selected survey unit that exceed the $DCGL_W$, do not exceed the $DCGL_{EMC}$*. In addition, the sum of the fractions for both $DCGL_W$ and $DCGL_{EMC}$ in applicable sections of a survey unit are less than unity for all site survey units, and
- All survey data pass applicable statistical testing criteria.
- In all cases, remediation is performed (as applicable) to reduce the levels of residual radioactivity concentrations to ALARA values.

NOTE: *The $DCGL_W$ is the permitted average concentration in the survey unit. The $DCGL_{EMC}$ is the elevated measurement comparison DCGL.

5.2.3 Development of Derived Concentration Guideline Levels

5.2.3.1 Dose Modeling

Dose models based on NUREG/CR-5512, Volume 1 (Reference 5-8) and RESRAD (Reference 5-10) were used to calculate site DCGLs. GPU Nuclear, Inc. also employed URS Corporation to assist in developing suitable subsurface dose models for the SNEC site. These dose models translate residual radioactivity levels into potential radiation doses to the public and are defined by three factors: 1) the scenario, 2) exposure pathways, and 3) the critical group. The scenarios described in NUREG/CR-5512 address the major exposure pathways of direct exposure to penetrating radiation and inhalation and ingestion of radioactive materials. These scenarios also identify the critical group. The critical group is the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity within the assumptions of the particular scenario.

Two representative scenarios were selected for the SNEC Facility. They are 1) building occupancy, and 2) residential farming. These scenarios, described below, represent reasonable and plausible human activities and future uses of the SNEC Facility site.

5.2.3.1.1 Building Occupancy Scenario

Because surface decontamination operations may not completely remove surface radioactivity, a scenario describing surface contamination is considered. This scenario accounts for exposure to both fixed and removable thin-layer radioactivity for a structure. This scenario also assumes that individuals occupy the building in a passive manner without deliberately disturbing the residual radioactivity on building surfaces. Occupancy of the building is assumed to begin immediately after license termination. The exposure duration is assumed for a full work year (2000 hours) continuing for seventy (70) years. The critical group consists of the building occupants, who are the people who work in the building following license termination. This scenario is described in Chapter 6, Section 6.2.1.

5.2.3.1.2 Residential Farming Scenario

Soil at the site is contaminated from licensed operations, accidental spills, and long-term accumulation of material in the soil from effluent releases. It is also contaminated from intentional disposal or burial of concrete or other structural debris such as pavement, masonry or structural steel. The residual radioactivity is assumed to be distributed in a surface soil layer covering the site on property that is used for residential and light farming activities, or is buried beneath this surface. The scenario assumes continuous exposure via multiple exposure pathways to the critical group. The critical group is the resident farming family who lives on the site following site remediation, grows some portion of their diet on the site and drinks water from a source at the site. This scenario is described in Chapter 6, Section 6.2.2.

5.2.3.2 Derived Concentration Guideline Level (DCGL)

The surface contamination and radionuclide concentration levels of structures, land areas, and facility systems remaining at the time of the FSS are compared to DCGLs calculated using the dose models. A DCGL is defined as "that concentration of residual radioactivity distinguishable from background radiation which if distributed uniformly throughout a survey unit, would result in a Total Effective Dose Equivalent (TEDE) of 25 mrem/y to an average member of the critical group." The average member of the critical group is the individual who is assumed to represent the most likely exposure situation based on the assumptions and parameter values used in the dose model calculation. SNEC Facility volumetric DCGLs are calculated using a deterministic approach that incorporates any input parameter that were probabilistically determined to be sensitive during the analysis. This approach ensures that the annual TEDE dose to the average member of the critical group (expected within the first 10,000 years after license termination) will be at or below 25 mrem. DCGLs are presented in terms of surface or volumetric radioactivity concentrations and are expressed in units of dpm/100 cm^2 or pCi/g.

SNEC Facility personnel have proposed a comprehensive set of volumetric DCGLw limits that were developed from the lowest calculated results from both the surface and subsurface regions within or near the SNEC site. SNEC management believes that this all-inclusive set of DCGLw values considers all viable pathways of exposure. These proposed DCGLw values consider both the surface and subsurface conditions of the site, modeled for locations that exhibit differing site dependent geological and hydrological characteristics. Soils and other materials were modeled down to bedrock and take into account the former Spray Pond area, Saxton Steam Generating Station footprint and the SNEC CV area. Surface DCGLw values are modeled down to one meter but the lowest values determined from the surface or subsurface models were then selected to represent the SNEC site 25 mrem/y DCGLs. This method ensures that relocated site soils will never deliver more than the previously described dose limit.

Modeling was also intended to provide DCGL values to meet the administrative goal of 4 mrem/y drinking water. These values in many cases, are more restrictive than the 25 mrem/y DCGLw values developed using all pathways modeling previously described. It is SNEC management's intent to consistently acknowledged this lower limit (4 mrem/y DW) when designing Final Status Surveys for site property. To accomplish this, both sets of values are listed in the SNEC LTP. The 25 mrem/y TEDE limits developed using all pathways modeling will be used to meet license termination requirements and therefore are site specific NRC reviewed and approved DCGLs. The second set of values relating to the 4 mrem/y drinking water goal, is provided as a SNEC Facility commitment to the general public.

A deselection process was used to reduce the applicable site radionuclide listing to eleven radionuclides which can be seen in Table 5-1. The dose from all deselected radionuclides will be accounted for in the final site survey design process by adjusting down the applicable DCGL as per section 6.2.2.3.

5.2.3.2.1 NRC Screening DCGLs

The NRC screening DCGLs are intended to be used as the principal means of releasing structures at the SNEC Facility. However, since the site is radiologically complex (i.e., multiple radionuclides are found in various distributions throughout the site) screening DCGLs are seldom, if ever applied directly to determine compliance with the site release dose criteria. Rather, they are used to develop surrogate ratio, gross activity, or elevated measurement comparison DCGLs ($DCGL_{EMC}$), or are applied using the unity rule. The gross activity or surrogate ratio DCGLs are termed "effective" DCGLs in the sections that follow. Where the broad conservatism inherent in the screening DCGLs prove unrealistically restrictive, a set of site specific DCGLs may be developed and used in their place (with regulatory approval).

NRC screening DCGLs for surface contamination, are presented in the second column of Table 5-1, and were calculated by entering standard default conditions into the NRC's dose modeling computer software "DandD". Alpha emitting radionuclides were calculated using the same default parameters provided in the DandD computer software (surface concentrations only).

Table 5-1

SNEC Facility DCGL Values [a]

Radionuclide	25 mrem/y Limit Surface Area (dpm/100cm^2)	25 mrem/y Limit (All Pathways) Open Land Areas (Surface & Subsurface) (pCi/g)	4 mrem/y Goal (Drinking Water) Open Land Areas [b] (Surface & Subsurface) (pCi/g)
Am-241	2.7E+01	9 9	2.3
C-14	3.7E+06	2	5.4
Co-60	7.1E+03	3.5	67
Cs-137	2.8E+04	6.6	397
Eu-152	1.3E+04	10.1	1440
H-3	1.2E+08	132	31.1
Ni-63	1.8E+06	747	1.9E+04
Pu-238	3.0E+01	1.8	0.41
Pu-239	2.8E+01	1.6	0 37
Pu-241	8.8E+02	86	19.8
Sr-90	8.7E+03	1.2	0 61

NOTES:

(a) While drinking water DCGLs will be used by SNEC to meet the drinking water 4 mrem/y goal, only the DCGL values that constitute the 25 mrem/y regulatory limit will be controlled under this LTP and the NRC's approving license amendment

(b) Listed values are from the subsurface model. These values are the most conservative values between the two models (i e, surface & subsurface).

5.2.3.2.2 Site-Specific DCGLs

Where DandD cannot be easily modified to incorporate the use of site-specific data, other computer dose modeling software, such as the US Department of Energy's RESRAD (Reference 5-10) and RESRAD-BUILD (Reference 5-11) software, may be used to generate site-specific DCGL values. The best approximation of real site conditions was obtained by replacing some default-input parameters in the RESRAD computer code, with site-specific parameters. Site-specific parameters, representing actual site conditions, were obtained from contracted geological services. Site-specific K_d data were empirically derived by Argonne National Laboratories (ANL) from samples of SNEC site soils and construction debris. These empirical values reduce overlapping conservatism and produce a more realistic estimate of site-specific conditions. Site-specific values used in developing site-specific DCGLs are documented in Chapter 6. Site-specific DCGLs are listed in the third column of Table 5-1.

5.2.3.2.3 Surrogate Ratio DCGLs

Surrogate ratio DCGL values may be established for areas where ratios between radionuclide concentrations are reasonably consistent. Establishing ratios between radionuclide concentrations allows other radionuclide concentrations to be predicted and therefore assumed present in the mix at a fixed fraction of another radionuclide. Likewise, a surrogate ratio DCGL allows the DCGLs specific to "hard-to detect" (HTD) radionuclides in a mix to be expressed in terms of a single radionuclide which is more easily measured. The measured radionuclide is called the surrogate radionuclide.

To obtain the best approximation of the actual ratio between radionuclides present in any area, a sufficient number of measurements or samples, spatially separated throughout the area must be collected. The number of measurements or samples needed to determine the ratio is based on the chemical, physical, and radiological characteristics of the radionuclides expected to be present in the area. The surrogate ratio may also be determined by using the most conservative ratio present in the data set (i.e., the ratio that will produce the highest calculated dose per unit concentration). One may also consider the variability in the sampling results by determining the upper 2 sigma bound for each radionuclide present in applicable sample data. These 2 sigma values would then be used with the mean value of the surrogate radionuclide to form a conservative mix for a survey area. In conclusion, selecting a conservative surrogate ratio ensures that potential exposures from individual radionuclides are not underestimated. The surrogate method will be used only when similar physical and geological characteristics are present in a survey unit.

The general equation used for a simple surrogate situation involving two isotopes such as Cs-137 and Sr-90 (as an example), is shown below:

$$DCGL_{Cs,\ Modified} = DCGL_{Cs} \times DCGL_{Sr} / (((C_{Sr} / C_{Cs}) \times DCGL_{Cs}) + DCGL_{Sr})$$

Where:

C_{Sr}/C_{Cs} = surrogate ratio of Sr-90 to Cs-137 (NUREG-1575, Equation 4-1 (Reference 5-5))

Post-remediation sampling may be performed to re-establish surrogate ratios when there is reason to believe that the remediation process may have significantly changed the ratios of the original radionuclide concentration mix. However, the surrogate ratio will usually be established using data collected prior to remediation. Additional sample data may be collected or an evaluation performed when there is reason to believe a surrogate ratio is non-representative of the survey area. In all cases, SNEC will use the best representative data for the survey area. A radionuclide mix justification will be documented in each survey package.

5.2.3.2.4 Gross Activity DCGLs

Where multiple radionuclides are present, a gross activity DCGL may be developed. The gross activity DCGL enables field measurements of gross activity rather than the determination of individual radionuclide activity, for comparison to the radionuclide-specific DCGL. The gross activity DCGL, or $DCGL_{GA}$, for surfaces or volumes with multiple radionuclides is calculated using the following equation (NUREG-1575, Reference 5-5, Equation 4-4):

$$Gross\ Activity\ DCGL = 1/((f_1 / DCGL_1) + (f_2 / DCGL_2) + \ldots (f_n / DCGL_n))$$

Where:

f_1 = relative fraction of the total activity contributed by radionuclide 1

f_2 = relative fraction of the total activity contributed by radionuclide 2

f_n = relative fraction of the total activity contributed by the nth radionuclide in the mix

In situations where an area has unknown or a highly variable concentration of radionuclides throughout, it is acceptable to select the most conservative mix of radionuclides present when developing a gross DCGL. Remediation efforts at the SNEC Facility will seek to remove contamination from concrete surfaces by removing significant near surface volumes of concrete. On other structural surfaces (e.g., support steel), abrasive decontamination techniques may be used to remove surface deposited contamination. Additionally, by aggressively chasing cracks and other structural anomalies, little (if any) surface deposited radionuclides should remain. Those that do remain will be at a significantly reduced fraction of their previous level. Characterization data and/or post-remediation sampling data will be used to adjust gross DCGL values before performing the FSS. In either case, a radionuclide mix justification will be documented in each survey package.

Gross activity measurements will be performed in both reference areas (background areas) and areas to be surveyed.

Open land areas will be scanned by semi-automated contractor supplied measurement equipment and/or by hand-held NaI detector instrumentation. Samples will also be obtained. When necessary, spectra of the components of a radiation field may be used to differentiate between site related radioactive materials and natural occurring radioactivity.

5.2.3.2.5 Elevated Measurement Comparison (EMC) DCGLs

Elevated Measurement Comparison (EMC) DCGLs are DCGL values that have been modified by reducing the dose model area size in the dose modeling computer code. The effect is to allow higher levels of contamination to exist in smaller well-defined on-site areas. Assuming the residual radioactivity is concentrated in a much smaller area rather than uniformly over the entire survey unit is the basis for developing the $DCGL_{EMC}$. The methodology used to calculate the $DCGL_{EMC}$ is given in Appendix 5-1. Area factors for individual radionuclides have been calculated and are presented in Tables 5-15 and 5-15A.

5.2.3.2.6 Unity Rule

Typically, each radionuclide specific DCGL corresponds to the release criteria (e.g., regulatory limit in terms of dose or risk). However, in the presence of multiple radionuclides, the total of the DCGLs for all radionuclides would exceed the release criteria. In this case, the individual DCGLs need to be adjusted to account for the presence of multiple radionuclides contributing to the total dose. One method for adjusting the DCGLs is to modify the assumptions made during exposure pathway modeling to account for multiple radionuclides. The surrogate measurement method discussed previously, describes one method for adjusting the DCGL to account for multiple radionuclides. Additionally, another method includes the use of the unity rule.

The unity rule, represented in the expression below, is satisfied when radionuclide mixtures yield a combined fractional concentration limit that is less than or equal to one (1) (NUREG-1575, Equation 4-3):

$$(C_1 / DCGL_1) + (C_2 / DCGL_2) + \ldots (C_n / DCGL_n) \leq 1$$

Where:

C = concentration of radionuclide (1, 2, ...,n)

DCGL = guideline value for each individual radionuclide (1, 2, ...,n)

A higher sensitivity will be needed in these measurement methods, as the values of C become smaller. In addition, this may influence statistical testing considerations by increasing the number of data points necessary for application of a specific statistical test.

5.2.3.2.7 Handling of Multiple Source Terms

When determining DCGLs in areas where there are multiple source terms, Equation 6-1 will be used.

5.2.4 Facility and Site Classification

Not all areas of the site have the same potential for residual radioactivity and, accordingly, do not need the same level of survey effort to demonstrate compliance with the site release criteria. Using the criteria given below, different sections of the site are grouped into impacted and non-impacted areas based on the potential for residual radioactivity to be present. Classification of site areas is based on professional judgment, operational history (Historical Site Assessment (HSA) information, Reference 5-19), site characterization data, operational surveys performed in support of decommissioning, and routine surveillance. See the site facility diagrams Chapter 2, and the SNEC site map (Figure 5-1), which is located at the end of this chapter.

5.2.4.1 Non-Impacted Areas

Non-impacted areas have no reasonable potential for the presence of residual radioactivity from licensed activities. These areas do not need any level of survey coverage since there was no radiological impact from site operations. No surveys are performed in these areas other than those used to determine a reference area (background).

5.2.4.2 Impacted Area

Impacted areas are areas that have a reasonable potential for the presence of residual radioactivity from licensed activities. Impacted areas are subdivided into three classes described below.

5.2.4.2.1 Class 1 Areas

Class 1 areas are areas that have or have had (prior to remediation), a potential for radioactive contamination (based on site operating history), or known contamination (based on previous radiological surveys).

Examples of Class 1 areas are:

- Areas previously subjected to remedial actions
- Locations where leaks or spills are known to have occurred
- Former burial or disposal sites
- Waste storage sites
- Areas with contaminants in discrete solid pieces of material at high specific activity
- Areas containing contamination more than the $DCGL_W$ before remediation

5.2.4.2.2 Class 2 Areas

Class 2 areas are those that have or have had, a potential for radioactive contamination but are not expected to contain radioactive material greater than the $DCGL_W$. Examples of Class 2 areas are:

- Locations where radioactive materials were present in an unsealed form,
- Potentially contaminated transport routes,
- Areas downwind of stack release points,
- Upper walls and ceilings of some buildings or rooms subject to airborne radioactivity,
- Areas where low concentrations of radioactive materials were handled, and
- Areas on the perimeter of radioactive material control areas.

5.2.4.2.3 Class 3 Areas

Class 3 areas are any impacted areas that are not expected to contain any residual radioactivity, or are expected to contain levels of residual radioactivity at a small fraction of the $DCGL_W$ This would again be based on site operating history and previous radiological survey information. Examples of Class 3 areas are:

- Buffer zones around Class 1 or Class 2 areas,
- Areas with a very low potential for residual contamination, but where insufficient information exists to justify a non-impacted classification.

5.2.4.3 Initial Classification

The initial classifications of the SNEC Facility are given in Table 5-2. They are based on site characterization data, the results of the Historical Site Assessment, and recommendations and concerns of SNEC Facility personnel knowledgeable of site conditions. Site characterization data and radiological history information on Table 5-2 survey areas are summarized in Chapter 2. When there was an uncertainty regarding the preliminary classification of a SNEC Facility impacted area, the area was initially assumed a Class 1 area until determined otherwise.

Table 5-2

Initial Classifications of Site Areas

Survey Unit Designations of the SNEC Facility and Surrounding Impacted Areas										
Survey Unit Number [a]	Description	Classification			Survey Unit Area (m^2) [b]				Number of Survey Units [b]	Type of DCGL Applied [c]
		1	2	3	Floor	Walls	Ceiling	Other		
MISCELLANEOUS AREAS & ITEMS										
MA1	Airborne Monitoring Stations		X					<10	1	1
MA2	SSGS Discharge Tunnel Outfall (Land Area)			X				600	1	2
MA3	Weir Outfall		X					25	1	2
MA4	Weir Outfall Buffer			X				200	1	2
MA5	Northeast Dump Site			X				7000	1	2
MA6	Northwest Open Land Area			X				4100	1	2
MA7	Northwest Open Land Area		X					100	1	2
MA8	Miscellaneous Concrete Slabs (Around Site)			X				<100	1 each	1
CONTAINMENT VESSEL (CV) - INTERIOR & EXTERIOR STEEL SHELL										
CV1-X	Interior Vertical Wall of CV Shell < ~804 5' El	X					392		4	1 [e]
CV2-X	Internal Support Ring Areas	X				65			22 [d]	1 [e]
CV3-X	Interior Curved Bottom of CV Shell	X						255	3	1 [e]
CV4-X	Exterior Wall – 802 6' El up to Cut-off	X				16 [g]			1	1 [e]
CV5	Exterior Wall 1 Meter Below Class 1 Area (Down to 797.6' El)		X			10			1	1 [e]
CV6	External Rock Anchor Support Ring Assembly Area	X				66			1 [d]	1 [e]
[f] MATERIAL HANDLING BAY (MHB) - SNEC AREA										
MH1	Floors & Walls Up to 2 Meters (Interior)	X			22	20			1	1
MH2	Upper Walls & Ceiling (Interior)		X			63	22		1	1
MH3	Roof			X			24		1	1
MH4	Exterior Walls			X		56			1	1

NOTES:

(a) "X" designates a sequential number starting with 1, and defines a survey unit within a survey area

(b) This data was estimated with best available information No survey unit, regardless of its classification will exceed 10,000 square meters.

(c) NRC Default Surface DCGLs = 1, Site Specific Volumetric DCGLs = 2

(d) Survey units were established as the ring areas became available to field personnel doing the survey work.

(e) Activation of CV steel liner to be addressed when region is accessible

(f) This facility may be removed prior to performing Final Status Survey.

(g) Based on projected cut-off at 804 5' El

Table 5-2 (continued)

Initial Classifications of Site Areas

Survey Unit Designations of the SNEC Facility and Surrounding Impacted Areas										
Survey Unit Number [a]	Description	Classification			Survey Unit Area (m^2) [b]				Number of Survey Units [b]	Type of DCGL Applied [c]
		1	2	3	Floor	Walls	Ceiling	Other		
[d] PERSONNEL ACCESS FACILITY (PAF) - SNEC AREA										
PF1	Floors & Walls Up to 2 Meters (Interior)	X			36	49			1	1
PF2	Upper Walls & Ceiling (Interior)		X			116	36		1	1
PF3	Roof			X			40		1	1
PF4	Exterior Walls			X		133			1	1
[d] DECOMMISSIONING SUPPORT BUILDING (DSB) - SNEC AREA										
DB1-X	Floors & Walls Up to 2 Meters (Interior)	X			212	121			5	1
DB2	Upper Walls & Ceiling (Interior)		X			290	212		1	1
DB3	Roof			X			225		1	1
DB4	Exterior Walls			X		325			1	1
DB5	DSB Carport Slab		X		62				1	1
DB6	DSB Carport Roof/Ceiling			X			124		1	1

NOTES:

(a) "X" designates a sequential number starting with 1, and defines a survey unit within a survey area

(b) This data was estimated with best available information. No survey unit, regardless of its classification will exceed 10,000 square meters

(c) NRC Default Surface DCGLs = 1, Site Specific Volumetric DCGLs = 2

(d) This facility may be removed prior to performing Final Status Survey

Table 5-2 (continued)
Initial Classifications of Site Areas

Survey Unit Designations of the SNEC Facility and Surrounding Impacted Areas										
Survey Unit Number [a]	Description	Classification			Survey Unit Area (m^2) [b]				Number of Survey Units [b]	Type of DCGL Applied [c]
		1	2	3	Floor	Walls	Ceiling	Other		
SAXTON STEAM GENERATING STATION (SSGS), INTAKE & DISCHARGE TUNNELS										
SS1	Floor of Discharge Tunnel (first ~150')	X			120				1	1
SS2	Floor of Discharge Tunnel (next ~235')		X		175				1	1
SS3	Floor of Discharge Tunnel (last ~315')			X	234				1	1
SS4	Ceiling of Discharge Tunnel (first ~150')		X				120		1	1
SS5	Ceiling of Discharge Tunnel (last ~550')			X			400		1	1
SS6-X	Walls of Discharge Tunnel (first ~150')	X				290			3	1
SS7	Walls of Discharge Tunnel (last ~550')			X		600			1	1
SS8-X	In DT – Seal Chambers (1, 2, & 3)	X						230	3	1
SS9	Spray Pump Pit Floor	X			120				1	1
SS10	Spray Pump Pit Walls Below 795' El.		X			20			1	1
SS11	Spray Pump Pit Walls Above 795' El.			X			100		1	1
SS12	SSGS Boiler Pad (811' El.)			X	1800				1	1
SS13	SSGS Firing Aisle (806' El.)			X	560	80			1	1
SS14-X	SSGS Basement Area Floor (790' El.)	X			360				4	1
SS15	SSGS Basement Walls – East End	X				100			1	1
SS16	SSGS Basement Walls Up to 2 Meters		X			240			1	1
SS17	SSGS Basement Walls > 2 Meters			X		350			1	1
SS18	Top of Seal Chambers		X		70				1	1
SS19-X	Section of SSGS Intake Tunnel Floor		X		493				2	1
SS20-X	Section of Intake Tunnel Walls		X			2150			3	1
SS21	Section of Intake Tunnel Ceiling			X			493		2	1

NOTES:

(a) "X" designates a sequential number starting with 1, and defines a survey unit within a survey area.

(b) This data was estimated with best available information. No survey unit, regardless of its classification will exceed 10,000 square meters

(c) NRC Default Surface DCGLs = 1, Site Specific Volumetric DCGLs = 2

Table 5-2 (continued)

Initial Classifications of Site Areas

Survey Unit Designations of the SNEC Facility and Surrounding Impacted Areas										
Survey Unit Number [a]	Description	Classification			Survey Unit Area (m^2) [b]				Number of Survey Units [b]	Type of DCGL Applied [c]
		1	2	3	Floor	Walls	Ceiling	Other		
SAXTON STEAM GENERATING STATION (SSGS) SPRAY POND AREA										
SP1	Open Land Area		X					6600	1	2
SNEC FACILITY SITE OPEN LAND AREA										
OL1-X	SNEC Facility Site & Near Site Area	X						11000	11	2
GPU ENERGY (PENELEC) SITE OPEN LAND AREA										
OL2-X	Westinghouse and Adjacent Areas [e]	X						5700	6	2
OL3	Warehouse Burn Area	X						200	1	2
OL4-X	Buffer Zones		X					5600	4	2
REMAINING IMPACTED OPEN LAND AREA										
OL5-X	Site Road Access Areas		X					20500	9	2
OL6-X	Stack Release Area (NNE)		X					14600	3	2
OL7-X	Stack Release Area (SSW)		X					12700	2	2
OL8-X	Buffer Zones			X				47900	5	2
[d] WAREHOUSE (LARGE GARAGE - SOUTH) - PENELEC AREA										
WA1-X	Floors & Walls Up to 2 Meters (Interior)		X		450	290			2	1
WA2	Upper Walls & Ceiling (Interior)			X		292	450		1	1
WA3	Exterior Walls			X		374			1	1
WA4	Roof		X				418		1	1

NOTES:

(a) "X" designates a sequential number starting with 1, and defines a survey unit within a survey area.

(b) This data was estimated with best available information. No survey unit, regardless of its classification will exceed 10,000 square meters.

(c) NRC Default Surface DCGLs = 1, Site Specific Volumetric DCGLs = 2

(d) This facility may be removed prior to performing Final Status Survey

(e) Includes substation yard drainage area.

Table 5-2 (continued)

Initial Classifications of Site Areas

Survey Unit Designations of the SNEC Facility and Surrounding Impacted Areas										
Survey Unit Number [a]	Description	Classification			Survey Unit Area (m^2) [b]				Number of Survey Units [b]	Type of DCGL Applied [c]
		1	2	3	Floor	Walls	Ceiling	Other		
[d] GARAGE (SMALL GARAGE - SOUTHWEST) - PENELEC AREA										
GA1-X	Floors & Walls Up to 2 Meters (Interior)		X		109	122			4	1
GA2-X	Upper Walls & Ceiling (Interior)			X		297	109		2	1
GA3	Exterior Walls			X		180			1	1
GA4	Roof		X				116		1	1
LINE SHACK - PENELEC AREA										
LS1-X	Floors & Walls Up to 2 Meters (Interior)		X		290	177			5	1
LS2-X	Upper Walls & Ceiling (Interior)		X			191	412		7	1
LS3	Exterior Walls			X		343			1	1
LS4	Roof		X				324		1	1
LS5	Roof Drainage System		X					<10	1	1, 2
PENELEC SWITCHYARD BUILDING & YARD STRUCTURES										
PS1	Interior			X	55	89	55		1	1
PS2	Exterior Walls and Roof			X		151	68		1	1
PS3	Switchyard Units – Base Pads			X	<500				1	1

NOTES:

(a) "X" designates a sequential number starting with 1, and defines a survey unit within a survey area.

(b) This data was estimated with best available information No survey unit, regardless of its classification will exceed 10,000 square meters.

(c) NRC Default Surface DCGLs = 1, Site Specific Volumetric DCGLs = 2

(d) This facility may be removed prior to performing Final Status Survey.

5.2.4.4 Changes in Classification

Changes in classification are based on survey data and other relevant information that indicates a different area classification is more appropriate. Changes in area classifications which decrease an area classification will be in accordance with License Condition 2.E.(i).

5.2.5 Final Survey Process

In general, FSS activities do not commence in the area to be surveyed until decontamination activities are believed to be complete and radioactive waste materials are removed. The FSS process begins with survey area preparation activities such as gridding and review of final remediation support survey information, as well as survey area walk-downs. Survey design calculations and the issuance of Survey Requests to field survey teams follow this phase. Field survey teams then collect the data and assemble the survey results in an organized and understandable format in accordance with site procedures. Data assessment and documentation concludes this process.

5.2.5.1 Survey Design Overview

Survey design, as described in Section 5.4, identifies relevant components of the FSS process and establishes the assumptions, methods, and performance criteria to be used. Areas ready for FSS are classified as Class 1, Class 2 or Class 3 and are divided into survey units. Systematic scan and static measurements are prescribed according to a pattern and frequency established for each classification. Investigation levels are established which, if exceeded, initiate an investigation of the survey data. A measurement from the survey unit that exceeds an investigation level may indicate a localized area of elevated residual radioactivity. Such locations are marked and investigated to determine the area and the level of the residual radioactivity present. Depending on the results of the investigation, the survey unit may require remediation, and/or re-survey or re-classification.

When necessary, a two-stage sampling process may be used IAW Reference 5-20. This sampling approach allows a second set of samples to be taken to meet the requirements of the statistical design of the survey. When used, this process will be incorporated as an option in the original survey design for the area.

Quality Control (QC) measurements are prescribed to identify and control measurement error and uncertainty attributable to measurement methods or analytical procedures used in the data collection process. QC measurements provide qualitative and quantitative information to demonstrate that measurement results are sufficiently free of error and accurately represent the radiological condition of the SNEC Facility.

5.2.5.2 Survey Data Collection

As deemed appropriate, a final post-remediation survey is performed using similar instrumentation, quality control and survey techniques to be used in the FSS process. The review of the final post-remediation survey data is then carried out to verify that residual radioactivity levels are acceptable and that no additional remediation will be needed in the survey unit. If an area of elevated residual radioactivity is identified, and remediation is determined to be ALARA, the area is remediated and re-surveyed to ensure meeting FSS requirements. The data collected during the final post-remediation survey (when performed),

provide a sound basis for interpreting radiological conditions that may be encountered during the FSS process.

Following the collection of acceptable post-remediation survey results (as applicable), the FSS is performed. It ensures that any remaining residual radioactivity meets the 25-mrem/y TEDE site release dose criteria. Measurement results stored as FSS data constitute the FSS of record and are included in the data set used to determine compliance with the site release criteria.

5.2.5.3 Survey Data Assessment

Survey data assessment, described in Section 5.6, is performed to verify that the final survey data are of adequate quantity and quality. Graphical representations and statistical comparisons of the data are made, which provide both qualitative and quantitative information about the data. Assessments are performed to verify that the data support the underlying assumptions necessary for the statistical tests. If the quality, quantity, or one or more of the assumptions are called into question, previous survey steps are re-evaluated. Statistical tests are then applied and conclusions are drawn from the data as to whether the survey unit meets the site release criteria.

5.2.5.4 Survey Results

Survey results are documented in history files, survey unit release records, and in the FSS report. A FSS report is prepared that summarizes the data and states the conclusions of the survey process.

5.2.6 Project Management

The planning and implementation of the FSS process is performed by SNEC Facility personnel supplemented by the Decontamination and Decommissioning Engineering group of GPU Nuclear, Inc. Aspects of the final survey project are outlined below.

5.2.6.1 Final Survey Organization

The Program Director, SNEC Facility - serves as the primary decision-maker (often called the Project Coordinator/Manager (PC/M)) and is responsible for the overall implementation of the FSS process.

The SNEC Facility Radiation Safety Officer (RSO) - is the lead individual on the FSS team. He is responsible for managing the FSS process and will act as a technical consultant to the PC/M.

The SNEC Site Supervisor - leads the Dismantlement Organization at the SNEC site. In addition, he provides craft support to the FSS team e.g., scaffold erection, to permit surveys of difficult to access areas.

The Manager Decontamination and Decommissioning Engineering - provides engineering support to the FSS Team and Dismantlement Organization.

Oversight of the process to assure compliance with the Quality Assurance/Quality Control (QA/QC) Program is provided under contract with qualified third party contractors.

5.2.7 Quality Assurance and Quality Control (QA/QC)

QA/QC is an integral part of all FSS activities. The objective of QA/QC, as applied to the FSS program, is to ensure the survey data collected, are of the type, and quality needed to demonstrate that the site is suitable for unrestricted release. Proper application of QA/QC activities will ensure that:

1. The elements of this plan are correctly implemented as prescribed,
2. The quality of the data collected is adequate, and
3. QA/QC reviews will identify deficiencies so that corrective actions, when needed, are implemented in a timely manner and confirmed to be effective.

The basis of the SNEC Facility FSS QA/QC program with respect to on-site sampling and analysis, has been derived from the applicable sections of Regulatory Guide 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operations) - Effluent Streams and the Environment" (Reference 5-12). The program is implemented at the SNEC Facility through procedure E900-QAP-4220.02, "SNEC Count Room Quality Assurance Program".

The SNEC Facility Decommissioning Quality Assurance Plan (DQAP) (Reference 5-13) is discussed in Section 3.0 of the SNEC Technical Specifications, and is applicable to all final status survey activities. Examples of the QA program application are described in the subsections that follow.

5.2.7.1 Training

Training is conducted to achieve initial proficiency and to maintain that proficiency throughout the final survey process. Personnel performing surveys receive training to qualify in the procedures being performed. Training includes:

1. Overview and objectives of the FSS process
2. Procedures governing the conduct of the final survey
3. Operation of the appropriate field and laboratory instrumentation
4. Collection of final survey measurements and samples
5. Survey data evaluation and documentation

The extent of training and qualifications is commensurate with the education, experience, and proficiency of the individual and the scope, complexity, and nature of the activity. Records of training are maintained in accordance with approved facility procedures (see Table 5-3).

5.2.7.2 Written Procedures

All FSS tasks, that are essential to survey data quality, shall be implemented and controlled by approved procedures or work instructions. Final Status Surveys will be performed in a controlled, deliberate manner, providing assurance of accurate results. Applicable provisions of the SNEC Facility Decommissioning Quality Assurance Plan (1000-PLN-3000.05, Reference 5-13), and the SNEC Radiation Protection Plan (E900-PLN-4542.01, Reference 5-14), apply to all FSS activities. Other implementing procedures have been or are under development to support FSS activities. The following matrix (Table 5-3) provides procedure titles relating to SNEC decommissioning and FSS issues, including FSS design and report generation.

Table 5-3

SNEC Procedure Matrix Listing

Area Of Applicability	Procedure Title	Number
Quality Assurance Plan	Saxton Nuclear Experimental Corporation Facility Decommissioning Quality Assurance Plan	1000-PLN-3000.05
Embedded Pipe and Components	Treatment of Embedded Piping and Components	E900-ADM-4500.58
Off-Site Dose	SNEC Facility Off-site Dose Calculation Manual	E900-PLN-4542.08
Radiation Protection	SNEC Radiation Protection Plan	E900-PLN-4542.01
ALARA	ALARA Review Program	E900-ADM-4500.35
Sample Analysis	Operation of the Packard Tri-Carb 2550 Liquid Scintillation Analyzer	E900-OPS-4524.46
Gamma-ray Spectroscopy	Operation of the Portable Gamma Spectroscopy System	E900-OPS-4524.43
Instrumentation	SNEC Rad Con Instrument Operations Manual	E900-OPS-4524.42
Gamma Scans of Samples	Operation of the SNEC Gamma Spectroscopy System	E900-OPS-4524 33
FSS Survey/Sampling	Survey Methodology to Support SNEC License Termination	E900-IMP-4520.04
Gridding Areas	Establishing a Reference Coordinate Grid System	E900-IMP-4520.03
Sample Preparation	Preparation of Sample Materials for Analysis	E900-IMP-4520.02
Isolation of Areas	Post Remediation Isolation	E900-ADM-4500.54
Source Accountability	SNEC Radioactive Source Accountability and Control Program	E900-ADM-4500 53
Decommissioning Safety	SNEC Facility Regulatory Review Process	E900-ADM-4500.52
ALARA	SNEC ALARA Program	E900-ADM-4500.47
Calculations	SNEC Facility Calculations	E900-ADM-4500.44
Training	Training Requirements for SNEC Facility Workers	E900-ADM-4500.42
Chain of Custody	Chain of Custody for Samples	E900-ADM-4500.39
Air Sampling	Environmental Air Sampling	E900-ADM-4500.30
Environmental Measurements	Environmental Monitoring	E900-ADM-4500.22
Radiological Surveys	Radiological Surveys: Requirements & Documentation	E900-ADM-4500.12
Procedures & Work Instructions	SNEC Facility Procedures and Work Instructions	E900-ADM-4500.07
FSS Design	Final Site Survey Planning	E900-ADM-4500.59
Records	Records Retention	E900-ADM-4500.04
Quality Control	SNEC Count Room Quality Assurance Program	E900-QAP-4220.02
Radiological Instrumentation	Quality Assurance Program for Radiological Instruments	E900-QAP-4220.01
Contamination Control	Alpha Control Program	6575-ADM-4500.51
Soil Erosion	SNEC Soil Erosion and Sedimentation Control Plan	6575-PLN-4542.02
Process Control	SNEC Facility Process Control Program	6575-PLN-4542.09

NOTE: This procedure listing is subject to change

5.2.7.3 Instrumentation Selection, Calibration and Operation

Proper selection and use of instrumentation will ensure that sensitivities are sufficient to detect radionuclides at the minimum detection requirements specified by the survey design which will assure the validity of the survey data. Instrument calibrations will be performed in accordance with approved GPU Nuclear, Inc. and/or vendor supplied procedures, using calibration sources traceable to the National Institute of Standards and Technology (NIST), or by qualified contractors providing results traceable to NIST. Operation of all survey instrumentation will be established by approved procedures.

5.2.7.4 Sample Chain of Custody

One of the most important aspects of sample management is to ensure that the integrity of the sample is maintained; that is, that there is an accurate record of sample collection, transport, analysis, and disposal. This ensures that samples are neither lost nor tampered with and that the sample analyzed in the laboratory is actually and verifiably the sample taken from a specific location in the field. SNEC has developed procedures for sample chain-of-custody using NUREG-1575 guidelines (Reference 5-5).

5.2.7.5 Quality Control Surveys

Quality control surveys will be made for all structures, facilities and open land areas. These measurements will be performed in accordance with Section 5.4.5. Quality control surface contamination measurements will be compared to the original FSS measurements. If the same conclusions are reached without any exceptions, the original FSS results for these survey units will be accepted as satisfactory. All discrepancies between FSS survey results and those obtained from quality control re-survey efforts shall be resolved with documented resolutions. If quality control measurements fall outside of their acceptance criteria, a documented investigation will be performed which may result in re-survey, re-sampling or other management actions.

5.2.7.6 SNEC Facility Sample Analysis

High-resolution gamma spectroscopy is used to identify gamma-emitting isotopes, in addition to other appropriate measurement methods used to determine the total radionuclide composition of sample materials Sample data results will be reviewed in accordance with existing requirements to ensure a reasonable interpretation of results.

Based on characterization data described in Chapter 2 of this plan, samples obtained at the SNEC Facility may contain one or more of the radionuclides listed in Table 5-4. Minimum Detectable Concentrations (MDC) determined during analysis will be performed using count times and instrumentation sensitive enough to detect typical environmental levels for these radionuclides (as appropriate).

Table 5-4

SNEC Facility Radionuclides of Concern

Open Land Areas & Structures

Am-241	Ni-63
C-14	Pu-238
Co-60	Pu-239
Cs-137	Pu-241
Eu-152	Sr-90
H-3	

NOTE: This data was derived from existing information (i e, Historical Site Assessment, characterization and scoping surveys).

SNEC Facility samples will be analyzed in accordance with the following requirements:

1. All samples will be analyzed using a Gamma Spectroscopy System.

2. If consistent radionuclide ratios have not been determined from existing data (i.e., scoping or characterization results), at least 5% of samples from the final status survey will include analysis for all radionuclides of concern, including Transuranic and Hard-To-Detect (HTD) radionuclides as listed in Table 5-4.

3. Inter and intra laboratory cross-checks will be performed on at least 5% of those FSS samples which were analyzed onsite.

4. QC samples will be taken per Section 5.4.5.1 and analyzed per SNEC procedure E900-QAP-4220.02, "SNEC Count Room Quality Assurance Program".

5.2.7.7 Access Control of Surveyed Areas and Systems

Administrative and physical controls for access to surveyed areas will be established to preclude the possibility of re-contamination subsequent to completion of the final status survey in that area.

5.2.7.8 Control of Vendor Supplied Services

Quality-related services, such as instrument calibration and laboratory analysis, are procured from qualified vendors whose internal QA program is subject to approval in accordance with the SNEC Facility DQAP. Vendors or contractors that are not listed on an approved vendor listing shall perform all necessary work in accordance with SNEC procedures and work instructions. If vendor services include survey or sampling data to be used as FSS results, these services will be reviewed and approved in accordance with the SNEC procedure (E900-ADM-4500.44, Reference 5-15).

5.2.7.9 Audits and Independent Reviews

Periodic audits are performed to verify that survey activities comply with established procedures and other aspects of the SNEC Facility DQAP and to evaluate the overall effectiveness of the program. The audits are conducted in accordance with approved procedures and performed by individuals who are independent of the activities being audited. Audit results are reported to

responsible management in writing, and actions to resolve identified deficiencies are tracked and appropriately documented. Qualified personnel will perform an independent review of the Final Status Survey Report. This review will ensure that FSS results are performed and documented in accordance with appropriate methodology, and that all conclusions reported are accurate and correctly presented.

5.2.8 Survey Records and Documentation

Generation, handling, and storage of FSS design information and survey data are controlled by approved procedures. Survey records and documentation are maintained as quality records and decommissioning records in accordance with approved facility procedures. Where possible, they are also maintained as electronic media.

At a minimum, each final status survey record will include:

1. Date and time survey was performed
2. Instrumentation used and calibration due date(s)
3. Survey location (grid location or other reference markings)
4. Type of measurement performed (scan survey, fixed-point measurements etc.)
5. Survey team member(s) involved
6. Name of field supervisor(s) responsible for reviewing survey data
7. Survey and Sample Request numbers

Generation, handling and storage of the original final status survey design and data packages shall be in accordance with the SNEC Records Retention procedure (E900-ADM-4500.04, Reference 5-16) and Radiological Surveys: Requirements & Documentation procedure (E900-ADM-4500.12, Reference 5-17).

5.2.9 Calculations

Formal calculations that support License Termination activities are prepared in accordance with the SNEC Facility Calculations Procedure (E900-ADM-4500.44, Reference 5-15). These calculations provide sufficient details with respect to purpose, method, assumptions, design input, references and units such that a person technically qualified in the subject can review and understand the analysis as well as verify the adequacy of the results without frequently consulting the originator. Calculations may be used for activities such as survey design, dose modeling, and computer code verification.

5.2.10 Schedule

Final status surveys are planned, scheduled, and tracked as a part of the overall decommissioning planning process. The schedule is dependent upon the progress and completion of several decommissioning activities and review and approval of the License Termination Plan. Presently, survey data collection is expected to begin in the fourth quarter of

2002. Final survey activities are planned and will be discussed with the NRC in advance to allow scheduling of the required public meeting on the License Termination Plan.

5.2.11 Stakeholders

The stakeholders for the SNEC decommissioning project include those organizations and concerned individuals listed below:

- Citizens Task Force (CTF)
- Concerned Citizens for Saxton Safety (CCFSS)
- Liberty Township
- Huntington and Bedford Counties
- The Commonwealth of Pennsylvania
- FirstEnergy Companies
- Applicable Contractors
- US Army Corps of Engineers

5.3 FINAL POST REMEDIATION SURVEYS

The professional judgment of the SNEC Facility staff will be used to implement final post remediation surveys in areas where former contamination levels required extensive remediation or in other areas as deemed appropriate. Properly designed, post remediation surveys can facilitate the transfer and control of areas, and minimize the impact of planned or ongoing dismantlement activities in adjacent areas.

5.3.1 Walk-down

A walk-down of the survey unit is performed prior to isolation. The principle objective of the walk-down is to assess the physical state of the survey unit and the scope of work necessary to prepare it for final survey. During the walk-down, requirements are identified for accessing, isolating, and controlling the survey unit. Support activities necessary to conduct the final survey, such as scaffolding, interference removal, and electrical tag-outs, are identified. Safety concerns such as confined space entry, high walls, and ceilings are identified. For systems, the walk-down includes a review of system flow diagrams and piping drawings. The walk-down is performed when the final configuration is known, usually near or after the completion of dismantlement activities.

5.3.2 Isolation Criteria

The following criteria will be satisfied prior to acceptance of a survey unit by the FSS team. The physical aspects of these criteria are verified during the walk-down.

1. Planned dismantlement activities within the post remediation survey unit are completed.

2. Planned dismantlement activities affecting or adjacent to the post remediation survey unit are completed, or are evaluated and determined to not have a reasonable potential to introduce radioactive material into the post remediation survey unit.

3. An operational radiation protection survey of the post remediation survey unit is completed and all outstanding items are addressed.

4. Planned physical work in, on, or around a post remediation survey unit, other than routine surveillance or maintenance, is complete.

5. Tools, non-permanent equipment, and material not needed for survey data collection are removed.

6. Housekeeping, clean up, and remediation of the survey unit are completed.

7. Scaffolding, temporary electrical and ventilation equipment and components, and other material or equipment needed for survey data collection is radiologically clean and left in place.

8. Transit paths to/through the post remediation survey unit are eliminated or re-routed.

9. Appropriate measures are instituted to prevent the re-introduction of radioactive material into isolated area from ventilation systems, drain lines, system vents, and other potential airborne and liquid contamination pathways.

10. Measures are instituted to control access and egress and otherwise restrict radioactive material from entering the survey unit.

5.3.3 Transfer of Control

Once a walk-down has been performed and the isolation criteria are met, control of activities within the post remediation survey unit is transferred from the dismantlement organization to the FSS team. The need for localized remediation within the isolated area may be identified after transfer of control. Localized remediation may be performed under the control of the FSS organization. However, if large areas require remediation, the isolated area may be transferred back to the dismantlement organization for further decontamination.

5.3.4 Isolation and Control Measures

Prior to performing the FSS, the post remediation survey unit is isolated and controlled. Routine access, equipment removal, material storage, and worker and material transit through the area without proper controls are no longer allowed. One or more of the following administrative and physical controls will be established to minimize the possibility of introducing radioactive material from ongoing decommissioning activities in adjacent or nearby areas.

1. Personnel training

2. Installation of barriers to control access to the area(s)

3. Installation of postings with access/egress requirements

4. Locking or otherwise securing entrances to the area

5. Installation of tamper-evident seals or labels

Isolation and control measures are implemented through approved facility procedures and remain in place through the FSS data collection process until license termination.

5.4 SURVEY DESIGN

The survey design identifies relevant components of the FSS process, and establishes the assumptions, methods, and performance criteria to be used. The methodology for planning a FSS, including a FSS in the subsurface region is identified in the applicable site procedure. Survey design is summarized in Table 5-5.

The application of survey design criteria to structures and land areas will vary based on the type of survey media and the relative potential for elevated residual radioactivity. For facility systems, many of the survey design criteria applicable to structures and land areas do not apply or are dictated by the physical system layout and the accessibility to the system piping and components. To accommodate these factors, the survey design integrates both non-systematic (random) and judgmental (biased) methods to data collection to achieve the overall objective of the final survey process. Survey design will be performed in accordance with SNEC procedures E900-ADM-4500.59, "Final Site Survey Planning" and E900-ADM-4500.58, "Treatment of Embedded Piping and Components". When necessary, a two-stage sampling process may also be used IAW Reference 5-20.

Each survey design package will address the following areas of interest:

1. A brief overview describing the final status survey design;

2. A description and map or drawing of impacted areas of the site, area, or building classified by residual radioactivity levels (Class 1, Class 2, or Class 3) and divided into survey units, with an explanation of the basis for division into survey units. Maps should have compass headings indicated;

3. A description of the background reference areas and materials, if they will be used, and a justification for their selection;

4. A summary of the statistical tests that will be used to evaluate the survey results, including the elevated measurement comparison, if Class 1 survey units are present, a justification for any test methods not included in MARSSIM, and the values for the decision errors (and) with a justification for values greater than 0.05;

5. A description of scanning instruments, methods, calibration, operational checks, coverage, and sensitivity for each media and radionuclide;

6. For *in-situ* sample measurements made by field instruments, a description of the instruments, calibration, operational checks, sensitivity, and sampling methods, with a demonstration that the instruments, and methods, have adequate sensitivity;

7 A description of the analytical instruments for measuring samples in the laboratory, including the calibration, sensitivity, and methodology for evaluation, with a demonstration that the instruments and methods have adequate sensitivity;

8. A description of how the samples will be collected, controlled, and handled;

9. A description of the final status survey investigation levels and how they were determined;

10. A summary of any significant additional residual radioactivity that was not accounted for during site characterization;

11. A summary of direct measurement results and/or soil concentration levels in units that are comparable to the DCGL and, if data is used to estimate or update the survey unit;

12. A summary of direct measurements or sample data used to both evaluate the success of remediation and to estimate the survey unit variance.

Table 5-5

Survey Design Summary

Specification	Class 1		Class 2		Class 3		Plant Systems
	Structures	Land Areas	Structures	Land Areas	Structures	Land Areas	
SUGGESTED SURVEY UNIT & REFERENCE GRID SIZES							
Size Range⇒	Up to 100 m^2	Up to 2,000 m^2	100 to 1,000 m^2	2,000 to 10,000 m^2	≤ 10,000 m^2	≤ 10,000 m^2	N/A
Reference Coordinate Grid[a]⇒	1 to 2 m	10 to 20 m	1 to 2 m	10 to 20 m	5 to 10 m	20 to 50 m	N/A
SCAN MEASUREMENTS							
Scan Coverage⇒	100%		10 to 100%[b] for Floors & 10 to 50% for Upper Walls & Ceilings		Judgmental, up to 10%		Variable[c]
Scan Area Selection⇒	Accessible surface areas		Judgmental, systematic along transects or of randomly selected grids		Judgmental, random		Judgmental; accessible[d] surface area
STATIC MEASUREMENTS							
Number of Measurements⇒	Calculated using the methodology of **Appendix 5-2** (Default Value is 30[e] for Small Survey Area Sizes, see Section 5 4 3 2)						30[f]
Location Selection⇒	Random starting point, systematic spacing[g]				Random		Accessible Points
Spacing (L)⇒	$L = (A/n)^{1/2}$ is for a square grid (see Section 5.4 3 2) A = total survey unit area; n = # of measurements				N/A		N/A
Type of Survey⇒	Surface Cont.	Soil Meas [h]	Surface Cont	Soil Meas [h]	Surface Cont	Soil Meas [h]	Surface Cont [i]

NOTES:

a) A square or triangular grid system pattern is used and multiple grid patterns are employed as necessary unless survey needs dictate otherwise.

b) Where scanning coverage greater than 50% is judged appropriate, the survey unit may be reclassified as a Class 1 survey unit.

c) Performed according to the scan coverage for the class of survey unit (where possible). The amount of accessible surface area dictates the percentage of surface area scanned

d) Includes health and safety considerations.

e) This number is sufficient for structural survey units less than ~10 square meters and land areas less than ~100 square meters in area, and may be used for embedments such as brackets, unistrut or sections of piping.

f) As allowed by plant system size and accessibility to system interior surfaces.

g) Except when statistical tests are not applied (i e , post remediation survey data for a survey unit are all less than DCGL) When systematic spacing is not practical, a random survey/sampling pattern will be used.

h) Subsurface samples will be obtained from randomly selected locations as well as biased locations (see FSS Design procedure (Table 5-3)

i) Scale and sediment samples will be collected from embedments (e g., piping, unistrut) as appropriate

5.4.1 Survey Units

As reflected in Table 5-5, impacted areas are divided into survey units to facilitate survey design. A survey unit is a physical area of specified size and shape with similar characteristics and potential for residual radioactivity for which data analysis and statistical analysis are performed. A separate decision is made for each survey unit as to its acceptability for release.

5.4.1.1 Survey Unit Size

Professional judgment is used to divide facilities and areas into appropriately sized survey units. Survey unit sizing is sufficient to assure that the total number of data points, based on the measurement frequency, will allow statistical evaluation of the data. Considerations for establishing survey units are physical characteristics, concentration levels, isotopic ratios, and previous remediation efforts, as well as spatial and logistical considerations.

Survey units are sized to ensure data points are relatively uniformly distributed among areas of similar potential for residual contamination. Small survey units are developed to ensure a conservatively established coverage of an area, location or residual system piping or embedment.

Survey units conform to site physical characteristics to the extent practical. Survey units have relatively compact shapes unless an unusual shape is appropriate because of site operational history or site topography. Where possible, existing facility or site characteristics such as horizontal and vertical structural supports (beams, concrete pour seams, or piping runs) are used to define the boundaries of the study. Suggested survey unit sizes are given in Table 5-5. Survey unit sizes may be adjusted as necessary as long as assumptions used to develop area dose models remain valid.

5.4.1.2 Reference Coordinate System

A reference coordinate system is used to facilitate selection of measurement locations, and provide a method for locating the same points in the future. A reference coordinate system is a set of intersecting lines referenced to a fixed site location or benchmark. Typically, grid lines are arranged in a perpendicular pattern, dividing the survey unit into squares of equal area. However, other types of grid patterns may be used. Examples of a reference coordinate system are shown in Figures 5-4 and 5-5, located at the end of this chapter.

Scale drawings, maps, or photographs of each survey unit are prepared, along with an overlay of the reference coordinate system. While some of the maps presented with this plan have a grid overlay, the actual or final layout may be somewhat different than that provided herein. The reference coordinate system provided on attached drawings are intended primarily for illustration purposes and do not necessarily dictate final grid layouts.

Physical gridding is used only where it is useful and cost effective. Where Class 1 and Class 2 survey units are gridded, the basic grid patterns are at 1 to 2 meter intervals on structural surfaces, and at 10 to 20 meter intervals over land areas. For practical purposes, Class 3 areas may typically be gridded at larger intervals, for example, 5 to 10 meters for large surfaces and 20 to 50 meters for land areas.

The physical grid layout on a structural surface is marked by chalk line or other appropriate means along the entire reference line or only at line intersections. For land areas, the reference

coordinate system is marked by wooden stakes or metal pins driven into the soil at line intersections, or by other appropriate surface markers. The selection of an appropriate marker depends on the physical characteristics and routine use of the survey area. Examples of gridded structures are shown at the end of this chapter (Figures 5-4 & 5-5).

5.4.1.3 Background Reference Areas

The residual radioactivity of a survey unit may be compared directly to the DCGL. However, the residual radioactivity may contain radionuclides, which occur in background. To identify and evaluate those contributions attributable to licensed activities, representative background radionuclide concentrations are established using background reference areas. Background reference areas have similar physical, chemical, radiological, and biological characteristics as the areas to be surveyed. They are usually selected from near site non-impacted areas, but are not limited to natural areas undisturbed by human activities. Surveys will be conducted of one or more background reference areas (where appropriate), to determine background levels for comparison with specific survey units. However, environmental background concentrations of Cs-137 will not be subtracted from nuclide specific measurement results (e.g., gamma-ray spectroscopy measurements of sample materials etc.) for any open land area. Appendix 5-3 provides additional discussions in selecting and applying background reference areas.

Background reference areas are not necessary where:

1. The residual radioactivity does not contain radionuclides occurring in background and the detection method is radionuclide-specific.

2. The background levels are known to be an acceptably small fraction of the $DCGL_W$

5.4.2 Scan Measurements

Scan measurements are performed to locate elevated areas of residual radioactivity that will require further investigation. They are performed according to a preset pattern established for each classification. The level of scanning effort is proportional to the potential for finding elevated measurement results.

Scan measurements of Class 1 survey units are performed over 100 percent of the accessible surface area. However, personnel health and safety issues are taken into consideration when determining whether an area is accessible. Scan surveys are designed to detect small areas of elevated radioactivity that would not be detected by a select number of static measurements, using a systematic measurement pattern. If the sensitivity of the scanning method is not sufficient to detect levels of residual radioactivity below the DCGL, the number of static measurements may be adjusted appropriately. Appendix 5-2 describes how this is accomplished.

Scan measurements of Class 2 survey units are typically performed over 10 to 100 percent of the surface area. Class 2 survey units have a lower probability of elevated residual radioactivity than Class 1 survey units. Those areas with the highest potential for elevated residual radioactivity (e.g., corners, ditches, and drains) are included in the survey based on professional judgment. If the entire survey unit has an equal probability of having areas of elevated residual radioactivity, systematic scans are performed along grid intersections of randomly selected grid blocks. A 10 percent scanning coverage is appropriate if it is unlikely that any area would exceed the DCGL. Coverage of 25 to 50 percent is appropriate when there may be locations

above the DCGL. Where scanning coverage of greater than 50 percent is judged appropriate, the survey unit may be reclassified as a Class 1 survey unit.

Scan measurements of Class 3 survey units are performed typically over approximately 10 percent of the surface area. Corners, ditches, and drainage or collection areas are included in the survey based on professional judgment. Class 3 survey units have the lowest probability of containing elevated residual radioactivity. Those areas with the highest potential for elevated residual radioactivity, based on professional judgment, are selected for scanning in a Class 3 area.

Scan measurements of facility systems are performed according to required scan coverage for the classification of the survey unit. The amount of accessible surface area dictates the percentage of the surface area scanned. However, the majority of all SNEC Facility systems and components have been removed from the site. When scanning is not possible, such as in water-covered areas, outfalls etc., a sufficient number of randomly selected samples will be collected as appropriate.

5.4.3 Static Measurements

Static measurements provide a quantitative measure of the radioactivity present at the location measured. Static measurements are performed at a frequency and location throughout each survey unit, such that a statistically sound conclusion can be developed. Static measurements may be performed at locations of elevated residual radioactivity identified by scan measurements. These types of static measurements may include direct surface contamination measurements, and soil and bulk material measurements.

There are several vendors that can supply semi-automated, large area, position sensitive, radiation measurement equipment. The use of this type of instrumentation is applicable in areas where relatively flat surfaces exist (either for structures or for surface soils). Scanning results using this equipment could be acceptably substituted for static measurements when the scan MDC is well below the requirements for releasing the area (e.g., 10% of the applicable DCGL). This type of equipment can provide greater confidence in the survey results in that surveyor error is greatly reduced and typical detection sensitivities are usually higher than that obtained using hand held survey equipment. In addition, an entire area is more appropriately scanned at 100% coverage making statistical testing of survey areas unnecessary.

GPU Nuclear, Inc. is currently evaluating these types of equipment and their capabilities in an effort to expedite and possibly improve overall scanning capabilities. If instrumentation of this type is used for scan measurements and the measurements are capable of providing data of sufficient quality as that provided by static measurements, they could be used in place of static measurements. The same logic may be applied for using *in-situ* gamma spectrometry in place of sampling and analysis for soil and other volume contaminated materials in concert with appropriate surrogate radionuclides. However, GPU Nuclear, Inc. has agreed that soil samples will still be collected in open land areas additional to these semi-automated scan survey or *in-situ* gamma spectrometry special measurement techniques.

5.4.3.1 Number of Measurements

As described in Appendix 5-2, the MARSSIM process incorporates design constraints that ensure that an adequate number of sample measurements are taken per survey unit. However, a minimum number of 30 measurement points per survey unit may be collected in areas or locations where there is difficulty in applying MARSSIM fixed point requirements, because of survey unit size, access problems or obstructions. Survey units of this type will be investigated thoroughly prior to planning the FSS.

5.4.3.2 Measurement Locations

Measurements in Class 3 survey units and background reference areas are taken in random locations. Random means that each measurement location in the survey unit has an equal probability of being selected. The random selection process uses random numbers that correspond to a survey unit's reference coordinate system to establish the measurement locations within the survey unit. The random numbers are generated using a random number generator or other acceptable random selection process. Measurement locations selected that do not fall within the survey unit area, or that cannot be surveyed due to site conditions, including health and safety considerations, are replaced. A technical evaluation will be written to justify not surveying an area due to safety and health considerations.

Remaining sections of facility system piping, and embedments of all types with limited accessibility, may not allow the full number of measurement points. When this is the case, an assessment of the embedment is performed to determine if the embedment can easily be removed or can be prepared for the FSS survey. The history of the area where the embedment resides is first reviewed along with the expected radionuclide concentration present on/in the embedment. In these special cases, the same MARSSIM approach encompassing historical assessment, characterization, remediation, and post remediation survey data is used as a basis for biased scanning and sampling to the extent practicable to ensure that the release criteria are met.

For Class 1 and Class 2 survey units, a random-start systematic pattern is used in place of a random pattern. This is done to meet survey design objectives and to locate small areas of elevated residual radioactivity that may exist within the survey unit. A random selection process determines the starting point. For a survey point placement using a square grid, the physical spacing of the measurement locations, L, is determined as follows (NUREG-1575, Equation 5-6):

$$L = \sqrt{A / n_{EA}}$$

Where:

A = area of the survey unit

n_{EA} = number of measurement points

The calculated value of L is rounded down to the nearest distance easily measured in the field. Using the reference coordinates, the measurement locations are identified around the starting point in a perpendicular manner at intervals of L. This process is repeated to identify the pattern of measurement locations throughout the survey unit. Where other than a square grid system is used, the physical spacing of the measurement locations is determined such that they are

distributed around the starting point in a systematic, equidistant manner across the survey unit area.

Equation 5-5 of NUREG-1575 is used when a triangular grid sampling point layout is used.

$$L = \sqrt{A / 0.866 n_{EA}}$$

Measurement locations selected using a random selection process or a systematic pattern that do not fall within the survey unit area, or that cannot be surveyed due to site conditions, including health and safety considerations, may be replaced. A technical evaluation will be written to justify not surveying an area due to safety and health considerations. Replacements are made using other measurement locations determined by the random selection process. Supplemental measurement locations are also determined using the random selection process but may require previous survey design considerations such as those listed in Reference 5-20.

Measurement locations selected based on professional judgment violates the assumption of unbiased measurements used to develop the statistical tests and are not used in the statistical evaluation. However, special considerations are necessary for structural survey units with surface areas less than 10 m^2, land areas less than 100 m^2, and remaining site system piping and other embedments. The data generated from these smaller survey units may be obtained using professional judgment, rather than systematic or random sampling design. Once again, the number of 30 measurement points for each of these smaller survey units may be used whenever practicable. Additionally, when the random-start systematic spacing pattern cannot be applied correctly because of a survey areas limited size or accessibility, then a completely random selection process will be used for purposes of locating survey and sampling points within the survey area.

5.4.3.3 Location Identification

Measurement locations within the survey unit are clearly identified and documented to ensure that they can be found again as necessary. Actual measurement locations are marked with tags, self-adhesive labels, bar codes, permanent markings, stakes, notations on survey maps, or equivalent methods. A unique identification code or number identifies each measurement location. The number convention allows survey data to be referenced back to a specific measurement point that is identified in photographs, drawings, diagrams or maps of the survey unit.

5.4.4 Data Investigation

The data collection, investigation and evaluation process checklist is presented in Figure 5-2. Use of this checklist or a similar organizer provides a high degree of confidence that all data requirements have been met.

Figure 5-2

Example of a Data Interpretation Checklist

1. Collect Data
   - Structural
   - Open land areas
   - Embedments
2. Convert Data to Standard Units
   - Structural activity in dpm/100 cm^2
   - Solid media (soil, etc.) activity in pCi/g
3. Evaluate Elevated Measurements
   - Identify elevated data/mark location
   - Compare data with derived elevated area criteria
   - Determine need to remediate and/or reinvestigate elevated condition
   - Compare data with survey unit classification criteria
   - Determine need to investigate and/or reclassify
4. Assess Survey Data
   - Review survey planning and design data
   - Verify that data of adequate quality and quantity were obtained
   - Perform preliminary assessments (graphical methods) for unusual or suspicious trends or results – investigate further if appropriate
5. Perform Statistical Tests
   - Select appropriate tests for category of contaminant
   - Conduct tests
   - Compare test results against hypotheses
   - Confirm power level of tests
6. Compare Results to Guidelines
   - Determine average or median concentrations
   - Confirm that residual activity satisfies guidelines
7. Compare Results with survey design information (ALARA may be included in the survey design)
   - Determine if all planning and design requirements are satisfied
   - Explain/describe deviations from design-basis survey requirements

5.4.4.1 Investigation Levels

Examples of typical investigation levels are shown in Table 5-6, taken from the MARSSIM manual (NUREG-1575, Table 5.8). Investigation levels are radioactivity levels that are based on the site release criteria, which if exceeded, initiate an investigation of the survey data. Investigation levels are typically established for each class of survey unit.

Table 5-6

Typical Investigation Levels (from NUREG-1575)

Survey Unit Classification	Flag Direct Measurements or Sample Result When:	Flag Scan Measurements When:
Class I	>$DCGL_{EMC}$ or >$DCGL_W$ and > a statistical parameter based value	>$DCGL_{EMC}$
Class 2	>$DCGL_W$	>$DCGL_W$ or > MDC
Class 3	>fraction of $DCGL_W$	>$DCGL_W$ or > MDC

The principal purpose of an investigation level is to guard against the possible misclassification of the survey unit. They also serve as a QC check during the final survey process. A survey measurement that exceeds an investigation level may indicate that the survey unit has been improperly classified. It may also indicate a failing survey instrument or a localized area of elevated residual radioactivity where there was a failure in the remediation process. Large variations in background may also result in investigative surveys. Investigative surveys of this type may be performed using a variety of measurement tools including *in-situ* gamma-ray spectroscopy instrumentation.

Depending upon the results of the investigation, survey units may require no action, may be remediated, or may be reclassified and/or re-surveyed. Initial administrative action or investigation level guidelines may be found in Table 5-7. For a Class 1 survey unit, while measurements above the DCGL are not necessarily unexpected, any measurement exceeding the DCGL is investigated. The site release criteria allows individual measurements representing small areas of residual radioactivity to exceed the DCGL. However, any measurement that exceeds the DCGL is subject to the elevated measurement comparison (EMC), described in Appendix 5-1. For a Class 2 survey unit, any measurement above the DCGL is unexpected and is investigated. As there is a low expectation for residual radioactivity in a Class 3 survey unit, any above background static measurement, exceeding a small fraction of a DCGL is investigated. If the scanning MDC exceeds the DCGL, any indication of residual radioactivity during the scan is also investigated.

If a background reference area is to be applied to the survey unit, the mean of the background reference area measurements may be added to the appropriate investigation level to which the survey measurements are compared. Where an excessive number of measurements exceed the investigation level, the results are reviewed to ensure that the applied background reference area is appropriate. Additionally, multiple reference areas may be used if reference areas have significantly different background levels because of the variability in background between areas. Furthermore, when different material types are present in the survey area, the lowest background material average may be used along with the highest material sample variability to produce a conservative assessment of residual activity in the survey area.

Table 5-7

Summary of SNEC Investigation/Action Levels

	SURVEY AREA TYPE		
	CLASS 1 AREA	**CLASS 2 AREA**	**CLASS 3 AREA**
Investigation or Action Level	Flag any discrete measurement > $DCGL_W$ or > mean + 3 sigma of the survey unit. Flag any scan measurement > $DCGL_W$.	Flag any discrete measurement ≥ 50% of $DCGL_W$. Flag any scan measurement > $DCGL_W$ or > MDC.	Flag any discrete measurement ≥ 10% of $DCGL_W$. Flag any scan measurement > $DCGL_W$ or > MDC.
Reclassification Level	N/A	If scan or discrete measurement values are ≥ the $DCGL_W$, area should be reclassified as Class 1.	If scan or discrete measurement values are ≥ $DCGL_W$, area should be reclassified as Class 1. If scan or discrete measurement values exceed 50% of the $DCGL_W$, but are < the $DCGL_W$ Area should be reclassified as Class 2 area
Remediation Guideline	Consider remediation when residual activity exceeds 3 times the $DCGL_W$ for any scan or discrete measurement averaged over 100 cm^2, or when the mean for any contiguous 1-m^2 area is more than the $DCGL_W$ for discrete measurements. Soil mean is > the $DCGL_W$ for 100 m^2 area or $DCGL_{EMC}$ is greater than 3 times the $DCGL_W$	Not indicated for Class 2 or Class 3 survey areas unless above limits specified for Class 1 survey area. Re-classify and Re-survey, as necessary.	

NOTE 1: The above values are initial values that will not be changed without applicable reviews in accordance with License Condition 2.E.(d). All investigation/action levels assume background values have been subtracted (as appropriate).

5.4.4.2 Investigation

Locations identified by scan or static measurements with residual radioactivity, which exceed an investigation/action level, are marked and investigated. The elevated measurement is then confirmed to exceed the investigation/action level. The area around the elevated measurement is investigated to determine the extent of the elevated residual radioactivity, and to provide reasonable assurance that other undiscovered areas of elevated radioactivity do not exist. Scan coverage of the area being investigated is increased to 100 percent (if not already at that

level). Static measurements are also taken if scan measurements are not capable of providing sufficient data to characterize the elevated area. A posting plot, described in Section 5.6.2.1, is generated to document the area investigated and the levels of residual radioactivity found. Depending on the results of the investigation, the survey unit may require remediation, reclassification, and/or re-survey. Possible outcomes of the data investigation process are shown in Table 5-8 below.

Table 5-8

Possible Actions Resulting From Data Analysis

No.	Data Results	Class 1	Class 2	Class 3
1	One or more data points > $DCGL_{EMC}$ or $DCGL_W$	Perform statistical testing, remediate and re-survey as necessary	Re-classify & re-survey	Re-classify & re-survey
2	All data points ≤ $DCGL_{EMC}$	Survey Unit passes applicable elevated measurement comparisons	N/A	N/A
3	All data points ≤ $DCGL_W$	Survey Unit passes	Determine if re-classification is required as follows below:	Determine if re-classification is required as follows below:
4	One or more points > 50% of $DCGL_W$ but ≤ $DCGL_W$	Survey Unit passes	Increase survey coverage or review & re-classify & re-survey as necessary	Re-classify & re-survey
5	One or more points > 10% of $DCGL_W$ but ≤ 50% of $DCGL_W$	Survey Unit passes	Survey Unit Passes	Re-classify & re-survey
6	All data points ≤ 10% of DCGL	Survey Unit passes	Survey Unit passes	Survey Unit passes

Static measurements above the investigation/action level that should have been, but were not identified by scan measurements may indicate that the scanning method is inadequate. In that case, the scanning method is evaluated and appropriate corrective actions are taken. Corrective actions may include re-scanning affected survey units using other survey protocol or survey instrumentation.

5.4.4.3 Remediation

Areas of elevated residual radioactivity above the $DCGL_{EMC}$ are remediated to acceptable levels. Based on the survey data, it may be necessary to remediate all or a portion of a survey unit. Remediation activities are addressed in Chapter 4.0.

5.4.4.4 Subdividing Survey Units

Due to size restrictions and other considerations, a survey unit may need to be divided into two or more smaller survey units. Survey unit sizes may be adjusted as necessary as long as assumptions used to develop area dose models remain valid. Suggested survey unit sizes are provided in Table 5-5.

5.4.4.5 Resurvey

If a survey unit is reclassified or if remediation activities are performed, then a re-survey using the methods and frequency applicable to the new survey unit classification is performed. This includes the case where only a small fraction of the area (< 10%) of a Class 1-survey unit is remediated.

In the case where a new survey unit is separated out from an existing survey unit or an existing survey unit is subdivided, Class 3 survey units need only additional randomly located measurements to complete the survey data set. Class 1 and Class 2 survey units require a new survey design based on random-start systematic measurement locations.

When a new survey unit is separated out from an existing survey unit or is subdivided, the new survey unit will include a buffer zone that adequately bounds the area of identified contamination when it borders a non-impacted area.

5.4.5 Quality Control (QC) Measurements

QC measurements are a component of the survey quality assurance process, and include quality checking and repeat measurements. Quality checking and repeat measurements are performed to identify, assess, and monitor measurement error and uncertainty attributable to measurement methods or analytical procedures used in the data collection process. Quality checking includes direct observations of survey data and sample collections, and sample preparation and analyses. Repeat measurements are multiple measurements at the same location or from the same survey unit. Repeat measurements provide quantitative information to demonstrate that measurement results are sufficiently free of error to accurately represent the radiological condition of the SNEC Facility. Results of QC measurements are documented in accordance with approved site procedures.

5.4.5.1 Type, Number, and Scheduling

QC checks will typically be performed by randomly re-sampling and/or re-surveying 5% of all sampling and/or survey points. For a low number of points (10 or less), the number of re-survey or re-sample locations will not be less than one (1). The type, number, and scheduling of QC measurements may also be determined by a performance-based method as described in Section 4 9.2 of NUREG-1575. This method is based on the potential sources of error and uncertainty, the likelihood of occurrence, and the consequences in the context of final survey data accuracy. The primary factors considered here are 1) the number of persons or organizations involved in the data collection, 2) the number of measurement types or analytical methods used, and 3) the time interval over which the data are collected. Other factors include:

1. Number of survey measurements collected,
2. Experience of personnel involved,

3. Types of measurement methods or sampling and analytical procedures used,
4. Variability of survey instruments used,
5. Level of radioactivity in the survey unit, and
6. How close the measurement level is to the detection limit.

QC measurements will be collected throughout the data collection process to verify that sources of error and uncertainty are minimized and controlled.

5.4.5.1.1 Scan Measurements

Quality checking of surface scanning surveys is performed to evaluate the effectiveness of scanning methods for identifying areas of elevated residual radioactivity. The frequency of quality checking for scan surveys is dependent on:

1. The number of surveyors,
2. The number of scanning methods employed,
3. The time interval over which scanning data are collected,
4. The scan MDC, and
5. Professional judgment.

The ability of surveyors to identify areas of elevated residual radioactivity by scanning will be periodically tested and documented in accordance with approved SNEC Facility procedure E900-ADM-4500.42, "Training Requirements for SNEC Facility Workers".

5.4.5.1.2 Static Surface Contamination Measurements

Repeat measurements of static surface contamination are performed to assess error and uncertainty associated with field measurement methods. Measurement locations are selected based on measurement results and are representative of the entire dynamic range of residual radioactivity found. The usable range of radioactivity includes the highest measurement result and the lowest measurement result with an acceptable measurement uncertainty, compared to the desired level of accuracy. Repeat measurements with results at or near the detection limit are not used because the measurement uncertainty is usually greater than the desired level of accuracy.

5.4.5.1.3 Soil and Bulk Material Measurements

Selected soil and bulk samples are analyzed and resubmitted for analysis as to determine (measure) precision. Replicate samples may be analyzed in their entirety or split to perform multiple analyses (split sample). The original analytical results are then compared with the replicate analysis results for agreement. When possible, replicate samples are analyzed using a second (different) instrument and are submitted to the laboratory(s) as unknowns. This process is implemented at the SNEC Facility through procedure E900-QAP-4220.02, "SNEC Count Room Quality Assurance Program".

5.4.5.2 Measurement Accuracy

Measurement accuracy may be estimated using the results of QC repeat measurements and comparing the results to the original measurements. For laboratory analysis, the results of the split samples are compared to one another. The accuracy estimates based on two or more surveyors (or laboratories) refer to the agreement expected when different surveyors or laboratories perform the same measurement using the same method.

Acceptance criteria for measurement accuracy are established during the survey design process or in approved SNEC procedures. Where the acceptance criteria are not met, a documented investigation of the data collection and/or sample analysis process is initiated to assess and identify the extent of error or uncertainty and to determine corrective actions.

5.5 SURVEY DATA COLLECTION

Survey data are collected after the survey unit has been isolated and controlled to ensure that radioactive contamination has not been introduced from ongoing decommissioning activities in adjacent or nearby areas.

5.5.1 Survey Performance

Survey data are collected from the post remediation survey, the final survey, and any investigation surveys performed. The final survey uses both random and biased data collection methods and is performed using the methodology, techniques, and quality control requirements prescribed in this plan. The post remediation and investigation surveys are biased surveys performed using similar methodology, techniques, and quality control requirements as the final survey. A post remediation survey, when performed, precedes the final survey. Investigation surveys may be performed at any time.

5.5.1.1 Post Remediation Survey

A special post remediation survey is performed where extensive dismantlement activities have occurred. These surveys are based on the professional judgment of the staff, and are performed when operational radiation surveys do not provide sufficient confidence that the survey unit is ready for the FSS. These special post remediation surveys are designed to verify that residual radioactivity levels are acceptable and that no additional remediation is necessary. It is conducted using similar methodology, techniques, and quality control procedures, as those required for the final survey. Professional judgment is used to: 1) identify measurement locations most likely to have elevated levels of residual radioactivity, and 2) establish the scanning coverage and the number of static measurements to be taken.

5.5.1.2 Final Survey

The objective of the final survey is to collect data of sufficient type, quantity, and quality, supporting conclusions regarding the radiological condition of the site. The final survey uses both random and biased data collection methods. Scan measurements are taken from both biased and prescribed measurement locations. Static measurement results used in the statistical tests are obtained from randomly derived measurement locations. The FSS of each survey unit is performed to demonstrate that residual radioactivity in each survey unit satisfies the predetermined criteria for release for unrestricted use

5.5.1.3 Investigation Survey

An investigation survey is performed when one or more survey measurements exceed an investigation level as described in Section 5.4.4. The purpose of the investigation survey is to define the area and level of the elevated residual radioactivity. The data collected during investigation surveys are used to characterize the area being investigated and to provide the basis for any further actions.

5.5.2 Instrumentation

Commercially available portable and laboratory instruments and detectors are used to perform three types of measurements: 1) surface scanning, 2) direct surface contamination measurements, and 3) laboratory analysis of soil and bulk materials. Other instrumentation is also used to perform other types of measurements as dictated by data collection requirements. The issuance, control, calibration and operation of survey instrumentation is controlled by approved procedures.

5.5.2.1 Instrument Selection

Radiation detection and measurement instrumentation is selected based on reliable operation, detection sensitivity, operating characteristics, and expected performance in the field. Typically, instruments used for static measurements are capable of detecting the radiation of concern at an MDC that is less than 50 percent of the $DCGL_W$.

Generally, instruments used for scan measurements are capable of detecting the radiation of concern to a MDC less than the $DCGL_W$. Typical instrumentation that may be used is identified in Table 5-9. The detectors used for direct surface contamination measurements are usually operated with data logging survey meters.

Table 5-9

Typical Survey Instrumentation Characteristics

Measurement Type	Detector Type	Effective Detector Area and Window Density	Instrument and Model	Detector Model
Alpha Scan	Gas-flow proportional	126 cm^2 0.8 mg/cm2 Aluminized Mylar	Ludlum 2350-1	Ludlum 43-68 (or equivalent)
Beta-Gamma Scan	Gas-flow proportional	126 cm^2 0.8 mg/cm2 Aluminized Mylar	Ludlum 2350-1	Ludlum 43-68 (or equivalent)
Gamma Scan	NaI Scintillator	1" D x 1" L, also 2" D x 2" L	Ludlum 2350-1	Ludlum 44-2 or 44-10 (or equivalent)
Dose Equivalent Instrument	Plastic Scintillator	1" D x 1" Length	Bicron Micro-Rem Meter	N/A
Static Surface Contamination	Gas-flow proportional	126 cm^2 0 8 mg/cm2 Aluminized Mylar	Ludlum 2350-1	Ludlum 43-68 (or equivalent)
Soil and Bulk Material	High-purity Germanium	1.60" x 1 94", 2.16" x 2.32"	Ortec/Canberra	N/A
Exposure Rate Instrument	Pressurized Ion Chamber (PIC)	~8 Liter Sphere	Reuter-Stokes RSS-131 (or eq)	N/A

5.5.2.2 Calibration and Maintenance

Instruments and detectors are calibrated for the radiation types and energies of interest. Anticipated radionuclide mixture ratios and varying energies are accounted for during calibration by using a calibration source with a conservative and/or representative average energy as compared to the weighted average energy of the anticipated nuclide mixture. For calibration of beta detectors, detector efficiencies are determined with calibration sources consisting of Tc-99, which emits an average energy of approximately 85 keV. This average energy is conservative when compared to the expected weighted average beta emission energy of 157 keV (the predominant detectable beta emitter is Cs-137). Similarly, the use of Th-230 or Pu-239 calibration sources having weighted average alpha energies of 4.654 MeV and 5.128 MeV, respectively, is representative as compared to the alpha emission energies (approximately 5.1 MeV to 6.1 MeV) of the expected surface nuclide mix for alpha emitters (Pu-238, Pu-239 & Am-241).

Instrument calibration and maintenance are performed in accordance with approved procedures. If vendor services are used, these services are conducted in accordance with approved procedures and a vendor QA program that is subject to approval in accordance with the Quality Assurance Program For Radiological Instruments and the SNEC Decommissioning Quality Assurance Plan (DQAP). Radioactive sources used for calibration purposes are traceable to the National Institute of Standards and Technology (NIST) or equivalent standards.

Calibration source efficiency variations resulting from source back scatter is usually small compared with efficiency loss due to uneven, wetted, corroded and/or semi-dusty surfaces typically encountered during the final survey process. Additionally, the distance between the source and the detector is a primary contributor to detection efficiency variations. To account for

survey surface to detector variations in the field such as surveys over scabbled concrete, the average variation in distance from the surveyed surface to the detector is determined based on the unevenness of the survey surface. A correction or distance factor is then applied to the calibration factor of the instrument. The distance factor will be determined empirically at the SNEC Facility using an appropriate large area (150-cm^2) calibration source and an approved procedure. Efficiency correction factors for wetted, corroded or semi-dusty surfaces will be determined in accordance with NUREG-1507 (Reference 5-18) guidance. If these survey area characteristics are shown to have an impact on detection efficiency, appropriate compensatory corrections will be made in the survey design, as well as the resulting calculated survey results.

5.5.2.3 Response Checks

Instrument response checks are conducted to assure consistency in instrument response, to verify the detector is operating properly, and to demonstrate that measurement results are not the result of detector contamination or other disturbances. At a minimum, instrument response is checked before instrument use each day. Portable instruments are also checked after instrument use each day. A check source is used that emits the same type of radiation (i.e., alpha, beta, and/or gamma) as the radiation being measured and that gives a similar instrument response. However, the check source does not necessarily use the same radionuclide as the radionuclide being measured. The response check is performed using a specified source-detector alignment that can be easily repeated. If the instrument fails its response check, it is not used until the problem is resolved. Measurements made between the last acceptable response check and the failed check are evaluated and either retained or discarded, as appropriate.

5.5.2.4 Minimum Detectable Concentration (MDC)

The MDC is determined for the instruments and techniques that are used for survey data collection. The MDC is the concentration that a specific instrument and technique can be expected to detect 95 percent of the time under actual field conditions.

5.5.2.4.1 Beta-Gamma Scan MDC For Structural Surfaces

Scanning methodology implemented at the SNEC Facility will follow an approved training process. A check source assembly will be used to train personnel in detecting areas exhibiting two to three times the background count rate. The source is adjusted to a height below the source assembly surface such that two to three times the background count rate is presented to the detector while siting directly over the source. Personnel will then be asked to move the detector across the source exposure area at a fixed rate of speed, while listening for elevated counting rates. Scan speeds are adjusted according to the ability of the surveyor to discern a specified counting rate above the existing survey area background level. The check source assembly is also used to train personnel in proper scan survey techniques.

The scan MDC, or MDC_{scan}, for scanning structural surfaces for beta and gamma emitters is determined from NUREG-1727, Equation E-2:

$$MDC_{scan}\,(building\ surfaces) = \frac{270{,}000 \cdot 1.38\sqrt{B}}{\sqrt{\rho \cdot \varepsilon_i \cdot \varepsilon_s \cdot A \cdot t}}$$

Where:

MDC_{SCAN} = minimum detectable concentration for scanning building surfaces (pCi/m^2)

270,000 = conversion factor to convert to pCi/m^2 (other conversion factors may be used as well to convert to other applicable units)

1.38 = index of sensitivity d'

B = number of background counts in time interval t

ρ = surveyor efficiency, (assumed to be 0.5, Reference 5-4)

ε_i = instrument efficiency for emitted radiation (cpm/dpm)

ε_s = source efficiency for emissions/disintegration

A = probe's sensitive area (cm^2)

t = time interval of the observation while the probe passes over the source In seconds

NOTE: The correction factor (A) as defined above, is not used in the "Compass" computer program developed for purposes of survey planning by the Oak Ridge Institute for Science and Education (ORISE).

The value of ρ represents a mean value for normal field conditions and is discussed in Section 6.6 of NUREG-1507, "Minimum Detectable Concentrations With Typical Radiation Survey Instruments for Various Contaminants and Field Conditions" (Reference 5-18).

5.5.2.4.2 Alpha Scan MDC for Structural Surfaces

For most conditions found at the SNEC Facility, the presence of alpha emitting radionuclides may be predicted by use of a surrogate beta-gamma emitting radionuclide such as Cs-137. Scanning for the surrogate radionuclide is significantly easier than scanning for alpha activity. During the scanning operation, when levels of the surrogate radionuclide are found to exceed a predetermined level, both a static alpha and beta-gamma measurement may be employed to verify the elevated scan reading. Relative ratios of radionuclides will be verified using sample and analysis techniques. This method of indirectly surveying for alpha surface activity provides additional assurance that these radionuclides are at acceptable levels on structural surfaces, and is superior to scanning only for the alpha radionuclide by itself.

When only alpha scans are employed, scanning for alpha emitters will differ significantly from scanning for beta and gamma emitters, in that the expected background response of most alpha detectors is close to zero. Since the time the probe is in the area of elevated residual radioactivity varies and the background count rate may be less than 1 cpm, it is not practical to determine a fixed MDC for scanning. Instead, another approach described in Section 6.7.2.2 of NUREG-1575 (Reference 5-5) is used. Given the $DCGL_W$ and a known scan rate, the probability of detecting an area of elevated residual radioactivity is calculated. The probability of detecting given levels of alpha surface contamination can be calculated by using Poisson summation

statistics (see Section 6.7.2.2 of NUREG-1575 (Reference 5-5) for a more complete description of this method).

For alpha survey instrumentation with backgrounds ≤ 3 cpm, a single count provides a surveyor sufficient, cause to stop and investigate further. When one or more counts are registered, the surveyor pauses scanning operations and waits for a predetermined time to determine if the counts are from elevated residual radioactivity. The time interval of the pause corresponds to a 90 percent probability of detecting counts associated with elevated residual radioactivity. This time interval may be calculated in accordance with Equation 6-13 of NUREG-1575 (Reference 5-5).

5.5.2.4.3 Gamma Scan MDC for Land Areas

The MDC_{SCAN} values for the Sodium Iodide detectors and radionuclides (shown in Table 6.7 of NUREG-1575 (Reference 5-5)), are examples of typical MDC_{SCAN} values that can be calculated assuming specific background levels are present in the survey area. The method given in NUREG-1507 (Reference 5-18), provides a more detailed example of how the scan MDC for gamma emitters can be determined. This is the method that will be used by the SNEC Facility when this survey approach is used. Site specific MDCs for all survey instrumentation will be derived and incorporated into survey packages.

5.5.2.4.4 Static MDC for Structural Surfaces

For static measurements of surfaces, the MDC_{static} may be calculated using NUREG-1727, Equation E-3 (Reference 5-4). More specific values for the calibration constant K shown in that equation are shown below in numbers 1 through 3:

1. The area of the detector (A)
2. The source efficiency factor (ε_s), and
3. The instrument efficiency for the emitted radiation(s) (ε_i)

$$MDC_{static} = \frac{3 + 4.65\sqrt{B}}{(\varepsilon_i \varepsilon_s)(A/100cm^2)\cdot t}$$

Where:

MDC_{static} = minimum detectable concentration for static counting (dpm/100 cm^2)

B = background counts during measurement time interval t (counts)

t = measurement counting time interval (minutes)

ε_i = instrument efficiency for emitted radiation (counts/emission)

ε_s = source efficiency for emitted radiation (emissions/disintegration)

A = area of detector (cm^2)

4. The total efficiency (ε_T) is the product of the instrument (ε_i) and source (ε_s) efficiencies. These values will be determined during the calibration process for the specific radionuclide mix expected in each survey area/unit (as appropriate). Actual instrument efficiencies are continuously monitored by site personnel. Any information or calculations used to establish instrument efficiencies for final status survey work will be available at the site for NRC on-site inspection purposes.

5. Other correction factors may be applied to the above equation as deemed appropriate.

5.5.2.5 Detection Sensitivity

The detection sensitivity of typical detectors for surface contamination measurements is estimated and the results summarized in Table 5-10. The results are shown for the principal instruments that are expected to be used for alpha and beta-gamma direct surface contamination measurements.

Count times are selected to ensure that the measurements are sufficiently sensitive with respect to the $DCGL_W$. For example, the count times associated with measurements for surface contamination and gamma spectral analysis (soil and bulk materials) are normally set to ensure an MDC_{static} is equal to or less than 50 percent of the DCGL. The scan rate associated with surface scans is normally set to ensure an MDC_{SCAN} of no more than 75 percent of the DCGL. If the MDC_{SCAN} exceeds the DCGL, additional static measurements may be required, as discussed in Appendix 5-1.

Table 5-10

Typical Detection Sensitivities

Instrument and Detector	Radiation	BKGND Count Time (min)	BKGND (cpm)	Instrument Efficiency[a,e] (cpm/dpm)	Count Time (min)	MDC dpm/100 cm^2	Scan[b] MDC dpm/100 cm^2
Ludlum Model 2350-1, 43-68 Probe	Alpha	5	2	0.155	1	49	500[c]
Ludlum Model 2350-1, 43-68 Probe	Beta-Gamma	5	243	0.275	1	220	511
2"x2" NaI	Gamma	1	~10k-20k	900 cpm/uR/h (weighted)	1 sec (scan)	N/A	~6 4 pCi/g (Cs-137)
Bicron Micro-Rem Meter	Gamma	N/A	µrem/h (varies)	Read Out in µrem/h	N/A	N/A	N/A
Reuter-Stokes[d] Pressurized Ion Chamber	Gamma	N/A	µR/h (varies)	Read Out in µR/h	N/A	N/A	N/A

NOTES:

[a] Actual calibration sources may be Cs-137, Tc-99, Am-241 or Pu-239 The efficiency is determined by counting the source with the detector in a fixed position from the source (reproducible geometry)

[b] MDC_{scan} is calculated by assuming a scan rate of 5 cm/sec (unless otherwise marked), which is equivalent to a count time of 0.03 min, assuming an 8 9 cm detector width

[c] The alpha scan MDC is determined by the approach described in Section 6 7.2.2 of NUREG-1575 (Reference 5-5) It assumes ≥ 1 cpm is necessary for the surveyor to pause

[d] The pressurized Ion Chamber (PIC) is used for comparison only No release survey data are collected using this instrument

[e] The total efficiency (ε_t) is the product of the instrument (ε_i) and source (ε_s) efficiencies These values will be determined during the calibration process for the specific radionuclide mix expected in each survey area/unit (as appropriate) Actual instrument efficiencies are continuously monitored by site personnel Any information or calculations used to establish instrument efficiencies for final status survey work will be available at the site for NRC on-site inspection purposes

5.5.3 Survey Methods

Survey measurements are performed in accordance with specific instructions contained in approved site procedures. Measurements include surface scans, static surface contamination measurements, and laboratory analysis of soil and bulk materials. Other measurements, such as removable surface contamination and exposure rate measurements may also be obtained as required.

5.5.3.1 Scan Measurements

Scanning is performed to locate small areas of residual radioactivity above the investigation level. If an area of elevated residual radioactivity is identified during the scan of a survey unit, the area is marked for investigation.

Structure and site system surfaces are scanned for beta-gamma emitting radionuclides. Beta scintillation or thin window gas-flow proportional detectors are typically used. In general, the detector is held less than 2 cm from the surface and moved at about 1 detector width per second. The scan rate is adjusted such that residual radioactivity can be detected at or below the investigation level.

Scanning for alpha emitters and low-energy beta emitters (<100 keV) are limited to structural surfaces. Where scanning is performed, alpha scintillation or thin window gas-flow proportional detectors are typically used. The detector is kept close to the surface, usually less than 1 cm, and moved at a rate such that there is a high probability of detecting elevated residual radioactivity.

Land areas are scanned for gamma-emitting radionuclides. Sodium iodide scintillation detectors are normally used because of their high detection efficiency. The detector is held close to the ground surface (usually less than 6 cm), and moved in a serpentine (S-shaped) pattern. This is done while walking at a speed that allows the surveyor to detect the residual radioactivity at or below the investigation level. A scan rate of approximately 0.5 meters/second is typically used to start. This rate is adjusted depending on the required detector response time and the general background level encountered in the field.

5.5.3.2 Static Surface Contamination Measurements

Static measurements are taken to detect surface contamination. Static measurements are generally performed by placing the detector on, or near the surface measured. Measurements are made over a discrete measurement time interval. Results are recorded. A one minute integrated count is a practical time interval for most field survey instrumentation and provides detection sensitivities that are usually below the $DCGL_W$. Other time intervals may be used as warranted.

Static measurements are taken with 100 cm^2 detectors or are corrected to reflect a 100 cm^2 area. In the event that contamination is more than what would be acceptable for an area of 100 cm^2, an evaluation is performed to ascertain compliance with the $DCGL_W$.

Static measurements are typically restricted to relatively smooth impermeable surfaces where the radioactivity is present as surface contamination. Because the detector is used in close proximity to the potentially contaminated surface, contamination of the detector or damage to the detector caused by irregular surfaces is considered before performing direct measurements.

5.5.3.3 Soil And Bulk Material Samples

Soil and bulk material samples are routinely collected and analyzed. Soil samples are generally collected down to a depth of 15 cm at static measurement locations. Sampling at greater depths is done in areas where site characterization or other information indicates the potential exists for contamination below 15 cm. Sample preparation may include, removing extraneous material, homogenizing, splitting, drying, and compositing sample materials for counting. For QC repeat measurements, the samples are obtained from the selected measurement locations as indicated in the survey/sample design package.

Samples of paint chips, tank sediment, sewage sludge, roofing material, concrete, pavement, and other bulk materials are collected for laboratory analysis as part of the biased sampling effort. Such samples may be collected in drain receptacles, sumps, and other locations in impacted areas. Selected storm drains are sampled in accessible locations. These samples are quantitatively analyzed by gamma spectroscopy for principal gamma-emitting radionuclides and the results compared to the $DCGL_W$. If residual radioactivity can be measured at $DCGL_W$ levels by *in-situ* gamma-ray spectroscopy techniques, *in-situ* techniques may be used in place of, or supplement, sampling and analysis methods. For gamma emitting radionuclides, the above data may also be supplemented by dose rate measurements.

5.5.3.4 Special Measurements

The historical site assessment and site characterization surveys are used to indicate if residual radioactivity is present in areas or locations described below. Any of these types of areas may require the application of special measurement techniques.

5.5.3.4.1 Cracks, Crevices, And Small Holes

Surface contamination on non-planar or irregular structure surfaces, such as cracks, crevices, and small holes, may be difficult to measure directly using standard field survey detectors and established techniques. Where no remediation has occurred, and residual radioactivity is not expected above background levels, cracks, crevices, and small holes are assumed to have the same level of residual radioactivity as that found on adjacent surfaces. The accessible surfaces are surveyed the same as other structural surfaces and no special measurement methods are applied.

Where remediation has occurred on surrounding surfaces, or where residual radioactivity above background levels is expected, a representative sample of surface contamination within the crack, crevice, or small hole is obtained. The level of residual radioactivity is measured and detection sensitivities can be adjusted such that reasonable approximations may be made using indirect measurement techniques. For the most part, the areas requiring special attention with respect to this type of survey and remediation work are located in the Saxton Steam Generating Station area and its associated Discharge and Intake Tunnels. At the SNEC Facility, contaminated regions of this type are remediated when possible. Additional discussion on this subject is presented in Appendix E, Section 10 of NUREG-1727, NMSS Decommissioning Standard Review Plan (Reference 5-4).

5.5.3.4.2 Paint Covered Surfaces

Surfaces painted to fix loose contamination are remediated before FSS activities begin. For other surfaces painted after site start-up, representative samples in areas where it is suspected that elevated levels of residual radioactivity could have been covered over, are taken and analyzed. Detection sensitivities are adjusted or remediation is performed as dictated by the sample analysis results.

5.5.3.4.3 Facility Systems, Floor Drains, Embedded Piping, Unistrut

Surface contamination on internal surfaces of site systems, floor drains, embedded piping and embedded unistrut sections may be inaccessible or difficult to measure directly using typical field survey instrumentation. For surface mounted supports, fixtures, unistrut and other similar hardware, no special measurement methods are generally needed where no remediation has occurred in the area where these items reside. When this is the case, internal surfaces of embedments are assumed to have the same level of residual radioactivity as that found on accessible surfaces. Areas where no remediation has been needed generally exhibit at or near background levels of contamination.

On the other hand, contaminated or potentially contaminated hardware items may be evaluated in several ways. Where remediation has occurred, or where residual radioactivity above background levels is expected, representative samples of these embedments may be obtained using a random sampling approach. The levels of residual radioactivity are then measured and evaluated on the removed embedment sections, and the results are extrapolated to the

remaining population. As an example, several unistrut sections of known length are first removed from an area, and then are assayed in a low background area to determine the residual activity per unit length (or unit surface area). From these assay results, a mean activity per unit length (or unit surface area) is then estimated as well as a postulated standard deviation and variance for the remaining population. The required number of additional samples are then calculated for the remaining lengths of unistrut. After unit conversion, these assay results may be compared directly to the applicable DCGL values using the appropriate statistical test criteria.

For other types of facility hardware such as embedded piping, scale and sediment, samples may be obtained (as allowed by system size and accessibility), at locations along a length of pipe system. The sample results are then assayed individually and a mean and variance are established for the locations sampled. This information is then used along with the fraction of the system sampled to estimate how many additional samples are needed to bound the total activity of the piping system. In addition, calibrated detectors may be used to measure the internal activity of piping and ductwork. Appropriately sized sodium iodide detection systems attached to multi-channel analyzers may be inserted into piping systems to perform *in-situ* analysis of hard gamma emitters (gamma-ray emitters will be used as the surrogate radionuclide in this case). Other types of detection systems that can measure internal surface beta and/or alpha emissions directly (in properly prepared piping sections), may also be used when appropriate. These types of measurements are considered special measurements and as such will be performed in accordance with a fully approved site procedure. Documentation of this type of survey process will be performed using the SNEC Calculations procedure or other approved site procedure.

GPU Nuclear, Inc. has performed many internal video inspections of piping at the SNEC Facility to assist in determining the viability of internal measurements, and also to map out or diagram the extent of some system piping. This process was necessary to determine the interconnections and level of sediment in pipe sections. GPU Nuclear, Inc. will continue to use internal video inspections as necessary to bound system piping on an as needed basis.

As a final resolution, when no reasonable method exists for evaluating the residual activity remaining in/on the embedment, the embedment is conservatively assumed to be above limits and is remediated. The embedment evaluation, measurement, and assessment program will be performed concurrently with other on-going remediation efforts at the SNEC Facility in accordance with approved site procedures.

5.5.3.4.4 Volumetrically Contaminated Structural and Re-Fill Materials (Soils & Crushed Construction Debris)

It is assumed that remediation efforts will be successful in removing residual activity to meet the established limits for structural surfaces. However, continued evaluations and sampling will be used (as necessary) to determine the depth of residual activity remaining in concrete surfaces before final status survey work begins. Additionally, indirect measurement techniques such as evaluating exposure rates, gross gamma count rates and/or *in-situ* gamma-ray spectrometry measurements may be performed as needed to derive residual contamination levels on or within structural materials. Detection sensitivities will be established such that reasonable but conservative approximations of the quantity of remaining volumetric activity can be made.

To aid in simplifying dose modeling requirements for this kind of material, samples of site soils and construction debris have been sent to an off-site laboratory where K_d (distribution

coefficient) values have been developed for relevant site radionuclides. These K_d values have then been used to develop final site $DCGL_W$ values for all volumetric material types at the SNEC site. By selecting the most conservative DCGLw developed from these various material types, a universally applicable $DCGL_W$ may then be used for all SNEC Facility volumetric materials. As a result of this modeling and pathways analysis technique, SNEC site $DCGL_w$ values may be used for both surface and subsurface soil and construction debris (re-fill or otherwise). Any residual activity allowed to remain in SNEC site structures or in soil materials will meet the site dose criteria for unrestricted release based on these $DCGL_w$ values.

A sampling and measurement program will be implemented to monitor and control residual contamination levels in re-fill materials. The sampling program will be statistically based and be applied through the implementation of fully reviewed SNEC site procedures and/or work instructions. Sampling and analysis will meet requirements stated in Section 5.2.7.6 of this plan.

5.5.3.4.5 Paved Parking Lots, Roads, Sidewalks, And Other Paved Areas

Paved parking lots, roadways, concrete slabs, and other paved areas are treated as structure surfaces. Scan and static measurements are taken as prescribed by the survey design Where remediation has occurred or where residual radioactivity above background levels is suspected, direct surface contamination measurements are taken and a representative number of subsurface samples (below 15 cm) will be collected and analyzed. Depending on the size of the paved area and the distribution of the residual radioactivity, the paved area may be a separate survey unit or be included as part of a larger survey unit. Sampling of these areas is also appropriate where there is reason to believe that contamination resides in, on, or below these structures.

5.5.3.4.6 Trailers And Temporary Facilities

Trailers or other temporary facilities used to support FSS or decommissioning work are not included in this study, but instead will be released in accordance with current SNEC Facility Radiological Controls work practices and procedures. Any temporary facilities remaining at the time of FSS activities shall be classified and surveyed in accordance with the applicable area or use classification.

5.5.3.4.7 Subsurface Soil Contamination Survey

The subsurface sampling/measurement program will be controlled by site procedures and will follow a systematic process for collecting subsurface information. In this methodology, each zone (surface, subsurface and buffer zone below the potentially contaminated region) will represent a sample population. The buffer layer will extend below the depth of any formerly buried components and the suspected depth of the contamination zone. The buffer layer depth and starting point will also be adjusted as indicated by sampling. The number of cores to be taken within each zone is the number N required for the applicable statistical test applied. The core samples will be homogenized over each 1 meter of depth during the sample preparation process. The appropriate test (WRS or Sign) will be applied to the results, as applicable. If the test indicates that the layer being assessed fails, the layer or the volume will be considered for remediation. Additionally, *in-situ* measurements may be considered when any layer exhibits results approaching 50% of the release criteria.

Areas where subsurface contamination may be present at the SNEC site are identified and sampled through the following process:

- Characterization and Historical Site Assessment (HSA) information were reviewed and used to determine the appropriate area classification. The area classification chosen considers both surface and subsurface volumes below structures as well as any previous remediation or survey efforts.

- A review of any existing measurement and/or sample results in the subsurface volume is then performed to determine if sufficient sampling results are available for planning a FSS.

- These areas are then made accessible; i.e. obstacles to sampling and survey work are removed (where possible), including any structural impediments.

- Where sampling below structures is prohibitively difficult or expensive; sampling through floor/slab structures or road coverings may be the appropriate choice rather than removing the entire structure to access the subsurface volume.

- The final state subsurface regions are identified including the depth and thickness of the buffer zone.

- Each subsurface layer is sampled and surveyed IAW a survey and sampling plan.

When any sample or survey result suggests or necessitates remediation of a volume, the remediation is performed before a final round FSS design is planned.

Identified locations where subsurface sampling/measurements will be planned include:

1. The Spray Pond area (~5500 square meters)

2. The 1.148 acre SNEC Facility site. To date, a significant portion of this area has been remediated.

3. Any suspect subsurface areas identified by site management that have shown contamination levels approaching the DCGLw.

5.5.3.4.8 CV Steel Shell Activation Area Survey

The activated section of the CV steel liner is currently assumed to be a region of the CV shell that extends from about the 790' El (operational water line in the reactor cavity) up to the proposed cut off region at about the 805' El (~15 feet). Additionally, the region is assumed to extend for a full quadrant of the CV or about 39' of the circumference of this building (centered horizontally at the former location of the reactor).

When the interior surface of the CV shell is thoroughly decontaminated, from residual surface contamination, samples of the steel shell will be collected within the activation zone previously described. The analysis of these samples will provide the best average concentration for the steel shell in the activation region. Additionally, a gamma measurement of the shell in this region may be used to augment the sampling efforts. These types of gamma measurements are special measurements and are described in more detail in Section 5.5.3.4.9. The direct and indirect dose contribution will be added to the dose contribution from residual surface contamination. The sum from these two sources will be maintained below 25 mrem/y TEDE.

5.5.3.4.9 CV Steel Support Ring Surveys

During 2002, SNEC was tasked with surveying and releasing several steel surface areas of the SNEC Containment Vessel (CV) steel shell in support of installation of steel I-beams, which were designed to stabilize the shell during removal of concrete. Survey areas were first aggressively cleaned using methods such as surface grinding which removed surface oxides, paint and any residual concrete that had adhered to the SNEC CV steel surface, as well as a thickness of the steel itself. This cleaning process removed contaminants to essentially the base metal, thus ensuring that the vast majority of surface contamination had been removed before the surveys began. Pre and post cleaning surveys were performed to verify that the cleaning effort was successful.

The survey was designed using NRC screening DCGLs for surface contamination as described in Table 5-1. A conservative scanning speed was set to locate elevated areas within the survey units which when detected, were re-measured for a full one minute of count time. Elevated measurement locations were re-cleaned and re-surveyed as necessary. Randomly located static measurement points were also counted for one minute.

These areas have been surveyed "at risk" in that they have been surveyed before NRC approval of the SNEC License Termination Plan (LTP). Conservative survey planning and remediation efforts have been used to ensure that all ring installation areas were decontaminated thoroughly below potential site release limits. In addition, radiological controls remained in place throughout the survey process to prevent survey area re-contamination.

This survey information will be included in the Final Status Survey Report.

5.5.3.5 Investigation Measurements

Removable activity, dose rate, and *in-situ* gamma spectrometry measurements may be used as diagnostic tools to further characterize the radiological conditions in selected areas, and to evaluate potential response actions. Sodium iodide detectors can also be used, both for hard to reach areas e.g., embedments, piping and duct work, as well as for subsurface monitoring efforts such as gamma-logging. Sodium iodide detectors become especially useful when employed in conjunction with multi-channel analyzers that are capable of discerning between

natural occurring and site-specific radionuclides.

Gamma-logging using a multi-channel analyzer is useful in both screening surveys (to determine depth and average concentration of contamination) and in final status surveys (to provide an upper limit of the average radionuclide concentration). If no significant counts are obtained in the detection system's region of interest (ROI), within a bore hole or piping system, then a "less than" value, or minimum detectable concentration (MDC), can be quoted for the soil around the bore hole or for a measured section of system piping at a given confidence level (95%). By ensuring that the MDC is less than the release criteria, the surveyor can designate the soil in the vicinity of the detector (or section of pipe) to be below the release criteria. Additionally, this type of measurement system is sensitive to elevated materials in adjacent buried piping or elevated pockets of contamination outside of the immediate sampling zone. Therefore, GPU Nuclear, Inc. will consider using gamma-logging in conjunction with sampling in areas where volumetrically contaminated materials approach the release criteria or when contamination is thought to be present in piping systems within a survey area.

5.5.3.6 Hard-To-Detect (HTD) Radionuclides

Many radionuclides are comparatively simple to detect in the field at environmental levels using routine gamma-ray spectroscopy analysis techniques. In contrast, the "Hard-To-Detect" (HTD) radionuclides are not easily identified using any routinely applied field measurement practices. SNEC has identified H-3, C-14 and Ni-63 as being the only HTD type nuclides of significance at the SNEC Facility. A summary of the radionuclide selection process can be found in Section 6.2.2.3.

5.5.4 Sample Handling and Analysis

When sample custody is transferred (e.g., when samples are sent off-site to another lab for analysis), a chain-of-custody record accompanies the sample for tracking purposes. The sample chain of custody record documents the custody of samples from the point of measurement or collection until final results are obtained. These tracking records are controlled and maintained in accordance with approved site procedures. On-site laboratory capabilities are used to perform gamma spectroscopy of bulk sample materials, gross beta-gamma and alpha counting of smears and Tritium analysis in liquid samples. Off-site laboratory services are procured as needed for Sr-90, TRU and other Hard-To-Detect (HTD) radionuclides. Laboratory analytical methods are generally capable of measuring levels at 10 to 50 percent (or less) of applicable $DCGL_W$ values.

5.5.5 Data Management

Final survey data may be collected from post remediation surveys, final surveys, investigation surveys or special measurement evaluations such as those made to determine embedment or sub-surface activity levels.

5.5.5.1 Other Scan Measurements

When 100% of any area is scanned at a high detection efficiency, capable of discerning low levels of residual activity (well below established $DCGL_W$ levels), collected results have a greater assurance that survey areas meet the site release criteria. Therefore, the need to measure a finite number of randomly selected survey points are reduced or eliminated. Consequently, some scan survey measurement efforts performed for initial phase and/or

investigative purposes, may be accepted as final survey data provided the following conditions are met:

1. The MDA for the scan is a small fraction of the required $DCGL_W$ for the survey area, and there is sufficient sensitivity present in the survey design at an acceptable confidence level.

2. All applicable survey data collection requirements as prescribed in Section 5.5 and 5.6.1 are followed.

3. The area was isolated after the survey activity.

5.5.5.2 Other Static Measurements

Other static measurements performed during post remediation and investigation surveys are based on professional judgment. Since they are biased and not random, they may not be used in the statistical tests. However, this does not necessarily preclude their acceptance as final survey data. These measurements may be accepted as final survey data provided:

1. All applicable survey data collection requirements as prescribed in Section 5.5 and 5.6.1 are followed.

2. Thirty or more data points are collected within the survey unit. For piping and other embedments, accessibility to interior surfaces may not allow this number of measurements. In these cases, similar survey methodology encompassing historical assessment, characterization, remediation, and post remediation survey data will be used as a basis for biased measurements and sampling, to ensure that the release criteria are met.

3. None of the data points exceeds the $DCGL_W$.

4. The area was isolated after the survey activity.

5.5.5.3 Data Recording

Survey measurements will be recorded in units appropriate for comparison to the $DCGL_W$ by correcting for material specific background, efficiency, geometry, detector area, and measurement size as applicable. The recording units are dpm/100 cm^2 for surface contamination and pCi/g for volumetric radionuclide concentrations.

Records of survey data are maintained in accordance with approved site procedures. Survey data records include the identification of the surveyor, type of measurement, location, instrumentation used, results, time and date measurement was performed and the instrument calibration information.

5.6 SURVEY DATA ASSESSMENT

The data assessment process checklist is illustrated in Figure 5-2. Final survey data, described in Section 5.5, are reviewed to verify they are of adequate quantity and quality. Graphical representations and statistical comparisons of the data can be made which may provide both quantitative and qualitative information about the data. An assessment is performed to verify the data. If the quantity or quality of the data is called into question, previous survey steps are re-evaluated. The statistical tests are applied and conclusions are drawn from the data as to whether the survey unit meets the site release criteria.

5.6.1 Data Verification and Validation

The final survey data will be reviewed to verify they are authentic, appropriately documented, and technically defensible. The review criteria for data acceptability are:

1. The instruments used to collect the data are capable of detecting the radiation of interest at or below the investigation level. If not, acceptable compensatory measures have been taken.

2. The calibration of the instruments used to collect the data is current and radioactive sources used for calibration are traceable to recognized standards or calibration organizations.

3. Instrument response is checked before and, where required, after instrument use each day data are collected.

4. Survey team personnel are properly trained in the applicable survey techniques, and this training is adequately documented.

5. The MDCs and the assumptions used to develop them are appropriate for the instruments and the survey methods used to collect the data.

6. The survey methods used to collect the data are appropriate for the media and types of radiation being measured

7. Special measurement methods used to collect data are applied as warranted by survey conditions, and are properly documented in accordance with an approved site procedure or Station Work Instruction.

8. The custody of samples that are to be sent for off-site laboratory analysis, are tracked from the point of collection until the final results have been obtained, and

9. The final survey data set consists of qualified measurement results representative of current facility status are collected as prescribed by the survey design package.

If a discrepancy exists where one or more criteria are not met, the discrepancy will be reviewed and corrective actions taken (as appropriate) in accordance with site procedures.

5.6.2 Graphical Data Review

Survey data may be graphed or plotted to identify patterns and relationships in the data that might go unnoticed using purely numerical methods. When needed, a posting plot and/or a frequency plot may be used. Other graphical data representation tools can also be used as appropriate.

5.6.2.1 Posting Plot

Posting plots, generated during investigation surveys, may be used to identify spatial patterns in the data. A posting plot is simply a map of the survey unit with the data values entered at the measurement locations. The posting plot can reveal non-homogeneous spatial characteristics in the survey unit such as patches of elevated residual radioactivity or groupings of measurements that exceed the $DCGL_W$. Even in a background reference area, a posting plot may reveal spatial trends in background data that might affect the results of the statistical tests. In some cases, the trends could be due to residual radioactivity, but may also indicate non-homogeneous characteristics in the background reference area.

5.6.2.2 Frequency Plot

A frequency plot is used to examine the general shape of the data distribution. A frequency plot is a bar chart of the number of data points within a certain range of values. The frequency plot may reveal any obvious departures from symmetry, such as skewness or bi-modality (two peaks), in the data distributions for the survey unit or background reference area. When the data distribution is highly skewed, it is often because there are a few elevated areas of residual radioactivity. The presence of two peaks in the data may indicate the existence of isolated areas of residual radioactivity or a mixture of background concentration distributions due to different soil types, construction materials, etc. The greater variability in the data due to the presence of such a mixture will reduce the power of the statistical tests to detect an inadequately remediated survey unit. These situations may indicate the need to more carefully match background reference areas to the survey unit, or to divide the survey unit into survey units with more homogeneous backgrounds.

5.6.3 Basic Statistical Comparisons

Statistical quantities (range, median, mean, and standard deviation) are calculated for the final survey data set where one or more data points exceed the $DCGL_W$. The calculated quantities are compared to the values shown in Table 5-11. The statistical comparison values represent assumptions underlying the statistical test to be used. Where the statistical quantity fails the comparison, the data set and/or survey design assumptions are examined.

Table 5-11

Basic Statistical Comparisons

Statistical Quantity	Acceptable Value	Failure Response
Range (R)	$R \leq 3\sigma$	Examine data for outliers
Median (u)	$\lvert(u - \bar{X}) / \sigma\rvert \leq 0.5$	Examine data for outliers and anomalies
Mean ($\bar{X}$)	$\bar{X} \leq DCGL_W$	Apply reference area, remediate or reclassify
Sigma (σ)	$\sigma \leq 0\ 3 \times DCGL_W$	Review initial survey design parameters

5.6.3.1 Outliers

Where the range is greater than 3 standard deviations, the data are examined for outliers. Outliers are measurements that are extremely large or small relative to the rest of the data set and, therefore, are suspected of misrepresenting the population from which they were collected. Outliers may result from measurement collection and recording errors. Outliers may also represent true extreme values of a distribution, such as areas of elevated residual radioactivity, and indicate more variability in the population than was expected. Not removing true outliers and removing false outliers both lead to a distortion of estimates of population parameters. Tests developed to detect outliers in a data set may be used to identify data points that require further examination. A test alone cannot determine whether statistical outliers should be discarded or corrected. This decision is generally based on professional judgment.

5.6.4 Statistical Testing

The Sign or the Wilcoxon Rank Sum (WRS) statistical test, also known as the Mann-Whitney test, may be applied to the final survey data set where one or more measurements exceed the $DCGL_W$. The statistical test is based on the hypothesis that the level of residual radioactivity in the survey unit exceeds the $DCGL_W$. There must be sufficient survey data with levels of residual radioactivity at or below the $DCGL_W$ to reject this statistical hypothesis and to conclude the survey unit meets the site release criteria.

5.6.4.1 Application of Statistical Tests

The statistical test does not need to be performed when the survey data clearly show that the survey unit meets the site release criteria. The survey unit clearly meets the criterion if:

1. Every measurement in the survey unit is less than or equal to the $DCGL_W$, or

2. A background reference area is used and the difference between the maximum survey unit measurement and the minimum background reference area measurement is less than or equal to the DCGL. In these instances, the statistical test is not applied.

The statistical test is applied where one or more measurements exceed the $DCGL_W$. Similarly, for a survey unit where a background reference area is used, the statistical test is applied where the difference between any survey unit measurement and any background reference area measurement is greater than the DCGL. Survey results and the corresponding conclusions for when a background reference area is and is not used are shown in Tables 5-12 and 5-13, respectively.

Table 5-12

Initial Survey Results and Conclusions When A Background Reference Area Is Not Used

Survey Result (Class 1 Areas)	Conclusion: Survey unit meets site release dose criteria	
	Yes	No
All measurements less than or equal to $DCGL_W$	✓	
Mean greater than $DCGL_W$		✓
Any measurement greater than DCGL with mean less than or equal to DCGL which passes Sign test	✓	
Any measurement greater than DCGL with mean less than or equal to DCGL which fails Sign test		✓

Table 5-13

Initial Survey Results and Conclusions When a Background Reference Area is Used

Survey Result (Class 1 Areas)	Conclusion: Survey unit meets site release dose criteria	
	Yes	No
Difference between maximum survey unit measurement and minimum background reference area measurement less than or equal to $DCGL_W$	✓	
Difference between survey unit mean and background reference area mean greater than $DCGL_W$		✓
Difference between any survey unit measurement and any background reference area measurement greater than $DCGL_W$, and difference between survey unit mean and background reference area mean less than or equal to $DCGL_W$ which passes WRS test	✓	
Difference between any survey unit measurement and any background reference area measurement greater than DCGL and difference between survey unit mean and background reference area mean less than or equal to $DCGL_W$ which fails WRS test		✓

5.6.4.2 Sign Test

The one-sample Sign statistical test is used if the radionuclide of concern is not present in background and radionuclide-specific measurements are made. The Sign test may also be used if one or more radionuclides are present in background at such small fractions of the $DCGL_W$ as to be considered insignificant. In this case, background concentrations of the radionuclides are included with the residual radioactivity (in other words, the entire amount is attributed to facility operations). Thus, the total concentration of the radionuclides is compared to the site release criteria. This option is only used if it is expected that ignoring the background concentration does not affect the outcome of the statistical test. The advantage of ignoring a small background concentration is that no background reference area is necessary.

The Sign test is applied as follows:

1. List the survey unit measurements, X_i, i = 1, 2, 3,. . .,n; where: n = the number of measurements.

2. Subtract X_i from the $DCGL_W$ to obtain the difference ($DCGL_W - X_i$, i=1, 2, 3...n).

3. Discard differences where the value is exactly zero and reduce n by the number of zero measurements.

4. Count the number of positive differences. The result is the test statistic S+. Note that a positive difference corresponds to a measurement below the $DCGL_W$ and contributes evidence that the survey unit meets the site release criteria.

Compare the value of S+ to the critical values in Appendix I, Section I.3 of NUREG-1575 (Reference 5-5). The Table columns equate to the false positive decision error rate, α. The value of α is the probability of passing a survey unit which actually fails to meet the site release criteria, which is obtained from the survey design (the initial value is 0.05 – see Appendix 5-2). If S+ is greater than the critical value for the false positive decision error rate given in the Table, the survey unit meets the site release criteria. If S+ is less than the critical value, the survey unit fails to meet the criterion.

5.6.4.3 Wilcoxon Rank Sum (WRS) Test

The two-sample WRS statistical test is used when the radionuclide of concern appears in background, or when a measurement method is used that is not radionuclide-specific. Because gross activity measurements are not radionuclide-specific, they must be performed for both the survey unit being evaluated by the WRS test and for the corresponding background reference area.

The WRS test is applied as follows:

1. Adjust the background reference area measurements by adding the $DCGL_W$ to each background reference area measurement, X_i ($X_i + DCGL_W$).

2. Sum the number of adjusted background reference area measurements, m, and the number of survey unit measurements, n, to obtain N (N = m + n).

3. Pool and rank the measurements in order of increasing size from 1 to N. If several

measurements have the same value, they are all assigned the average rank of that group of measurements.

4. Sum the ranks of the adjusted background reference area measurements to obtain Wr.

5. Calculate the critical value using equation I.1, NUREG-1575 (Reference 5-5). This equation is used when there are several measurements that have the same value.

$$Critical\ Value = ((m(n+m+1))/2) + (z\sqrt{nm(n+m+1)/12})$$

Where:

z = The (1 - α) percentile of a standard normal distribution, which can be found in the Table 5-14 below.

Table 5-14

Values For α and z

α	z
0.001	3.090
0.005	2.575
0.01	2.326
0.025	1.960
0.05	1.645
0.1	1.282

NOTE: The value of α is obtained from the survey design (initial value is 0 05 - see Appendix 5-2). NRC approval is required to increase the α (type 1 decision error) >0 05 in accordance with License Condition 2 E.(h) Where m and n are less than 20, the critical value is given in Table I 4 of NUREG-1575 (Reference 5-5)

6. Compare the value of Wr with the critical value calculated above. If Wr is greater than the critical value, the survey unit meets the site release criteria. If Wr is less than the critical value, the survey unit fails to meet the criterion.

5.6.5 Data Conclusions

The results of the statistical test allow one of two conclusions to be drawn. The first conclusion is the survey unit meets the site release criteria. The data have provided statistically significant evidence that the level of residual radioactivity in the survey unit does not exceed the site release criteria. The decision that the survey unit is acceptable for unrestricted release can be made with sufficient confidence and without further analysis.

The second conclusion that is that the survey unit fails to meet the site release criteria. The data does not provide sufficient statistically significant evidence that the level of residual radioactivity in the survey unit does not exceed the site release criteria. The data is analyzed further to determine why the statistical test result led to this conclusion.

Possible reasons the survey unit fails to meet the site release criteria are:

1. It is in fact true,

2. It is a random statistical fluctuation, or

3. The test did not have sufficient power to detect that it is not true. The power of the test is primarily based on the actual number of measurements obtained and their standard deviation. A retrospective power analysis for the test may be performed as described in Appendices I.9 and I.10 of NUREG-1575 (Reference 5-5). If the power of the test is insufficient due to the number of measurements, additional data may be collected. If it appears that the failure may be due to statistical fluctuations, the survey unit may be resurveyed and another set of discrete measurements collected for statistical analysis. A larger number of measurements increases the probability of passing if the survey unit actually meets the site release criteria. If it appears that the failure was caused by the presence of residual radioactivity in excess of the site release criteria, the survey unit is remediated and resurveyed.

5.7 SURVEY RESULTS

Survey results are documented in history files, survey unit release records, and are summarized in the final survey report. Other detailed and summary data reports may be generated as requested by the NRC or SNEC Management.

5.7.1 Survey Unit Release Record

The survey unit release record is the complete release record in a standardized format prepared for each survey unit or group of survey units with similar histories. The survey unit release record is a collection of information necessary to demonstrate compliance with the site release criteria. This record includes:

1. A history file checklist:

   The history file checklist references relevant operational and decommissioning data. The purpose of this checklist is to provide a basis for the survey unit classification. The history file will reference relevant sections of the Historical Site Assessment (Reference 5-19) and other compiled records including:

   - History of remediation
   - The survey unit operating history affecting radiological status
   - Scoping, site characterization and post remediation survey data
   - Other relevant information.

2. Description of the survey unit

3. Survey design information for the survey unit

4. Survey unit ALARA analysis, if performed

5. Survey measurement locations and corresponding survey data

6. Survey unit investigations performed with documented results, as applicable

7. Any survey unit data assessment results

8. Results of any special measurements performed for the survey unit

5.7.2 Final Survey Report

A final survey report will be prepared and submitted to the NRC. The report will provide a summary of any ALARA analysis, survey data results, and overall conclusions, which demonstrate that the SNEC Facility and site meet the radiological criteria for unrestricted use. Information such as the number and type of measurements, basic statistical quantities, and statistical test results will be included in the report.

The following outline illustrates a general format that may be used for the final status survey report. The outline below may be adjusted to provide a clearer presentation of the information. The level of detail will be sufficient to clearly describe the final status survey program and certify the results.

Information to be submitted (Reference 5-4, Section 14.5):

1. A summary of the results of the final status survey.

2. A discussion of any changes that were made in the final status survey from what was proposed in the LTP or other prior submittals.

3. A description of the method by which the number of samples were determined for each survey unit (see Reference 5-5, Section 5.5.2).

4. A summary of the values used to determine the numbers of samples and a justification for these values (see Reference 5-5, Section 5.5.2).

5. Survey results for each survey unit including:

   - Number of samples taken for the survey unit.

   - A map or drawing of the survey unit showing the reference system and random start systematic sample locations for Class 1 and 2 survey units, and random locations shown for Class 5 survey units and reference areas.

   - Measured sample concentrations.

   - Statistical evaluation of the measured concentrations (see Reference 5-5, Section 8.3, 8.4 and 8.5).

   - Judgmental and miscellaneous sample data sets reported separately from those samples collected for performing the statistical evaluation.

- Discussion of anomalous data including any areas of elevated direct radiation detected during scanning that exceeded the investigation level or measurement locations in excess of the DCGLw.

- A statement that a given survey unit satisfied the DCGLw and the elevated measurement comparison if any sample points exceeded the DCGLw.

6. A description of any changes in initial survey unit assumptions relative to the extent of residual radioactivity.

7. When a survey unit failed, a description of the investigation conducted to ascertain the reason for the failure and a discussion of the impact that the failure has on the conclusion that the facility was ready for final radiological surveys.

8. If a survey unit failed, a description of the impact that the reason for the failure has on other survey unit information.

5.7.3 Other Reports

If requested by the NRC, computer-generated and/or summary data reports will be provided in hard copy or electronic form. Survey data include date, instrument, location, type of measurement, and mode of instrument operation. Other data, such as conversion factors, background reference areas, and the MDCs used, are available which will allow independent verification of the results. Measurement results will also be presented graphically. The FSS report will be independently reviewed.

Any independent verification survey performed will be performed by an organization outside the SNEC Facility staff and management organization. Reports generated as a result of any independent verification survey process initiated by the SNEC Facility, will be available upon request.

5.8 DEFINITIONS

1. Accessible Surface Area - An area available to a radiation detector for direct scanning or fixed-point measurements.

2. Area Factor (A_{EMC}) - A factor used to adjust the $DCGL_W$ to estimate $DCGL_{EMC}$ and the minimum detectable concentration for scanning surveys in Class 1 survey units ($DCGL_{EMC} = DCGL_W \times A_{EMC}$. The area factor ($A_{EMC}$) is the magnitude by which the residual radioactivity in a small area of elevated activity can exceed the $DCGL_W$, while maintaining compliance with the release criterion. SNEC Facility area factors are listed in Table 5-15 of Appendix 5-1.

3. Background Radiation - Naturally occurring radiation which may include cosmic, terrestrial (radiation from the naturally radioactive elements) and man-made radiation from global fallout.

4. Characterization Survey - A radiological survey and its supporting evaluations performed to establish the SNEC Facility radiological condition for planning decommissioning activities.

5. Confidence Level - The probability associated with a confidence interval which expresses the probability that the confidence interval contains the population parameter value being estimated.

6. Derived Concentration Guideline Level (DCGL) - Residual radioactivity levels that equate to the site release criteria for that particular pathway or measurement. The two (2) basic DCGLs defined in this plan are 1) the $DCGL_W$ and, 2) the $DCGL_{EMC}$. The $DCGL_W$ is the average concentration limit for the standard size survey area. The $DCGL_{EMC}$ is the elevated measurement area DCGL, which is used for small areas of elevated activity (above the $DCGL_W$). When not defined, DCGL refers to the $DCGL_W$. Other DCGLs discussed in this plan (e.g , $DCGL_{GA}$ etc.) are derived from these two basic definitions and are sometimes referred to as an "effective DCGL".

7. Elevated Area - Areas of residual contamination exceeding the guideline value.

8. Final Status Survey (FSS) - Radiological measurements, evaluations and supporting activities undertaken to demonstrate that the SNEC Facility satisfies the criteria for unrestricted use.

9. Hard-to-Detect Nuclide (HTD) - A radionuclide emitting radiation(s) that are difficult to detect with field or laboratory based instrumentation.

10. History File - A compilation of information used to justify the classification and survey design for the survey unit. It should reference sections of the Historical Site Assessment, characterization survey data, remediation surveys and other information used to establish the basis for the design of the final status survey.

11. Independent Verification Survey - An information only radiological survey, performed by an organization independent of the SNEC Facility staff and

management, which will provide SNEC Facility management with an additional level of confidence concerning the validity of the Final Survey results.

12. Minimum Detectable Activity (MDA) - The minimum level of radiation or radioactivity that can be measured by a specific instrument and technique. The MDA is usually established on the basis of assuring false positive and false negative rates of less than 5%.

13. Minimum Detectable Concentration (MDC) - The minimum activity concentration on a surface or material volume that can be statistically detected above background. This is usually set at the 95 % confidence level.

14. Multiple Source Terms - Generic term used when more then one source term element is encountered (e.g., a remaining site structure with surface contamination and embedments).

15. Operational Survey - A radiological survey performed in accordance with SNEC procedures in support of routine site operations.

16. Quality Control Survey - A survey that consists of repeat measurements on a specified fraction of the survey areas. The survey areas are usually selected at random to provide an additional check of final status survey measurements.

17. Release Criteria - A term used to identify the radiological requirements for release of the SNEC Facility for unrestricted use.

18. Remediation Survey - Any survey performed that is used to determine the effectiveness of remediation activities. The final post remediation survey is a special remediation effectiveness survey performed with instrumentation similar to the type used for the FSS. The survey methodology is also similar to actual FSS methodology.

19. Scan Survey - A qualitative radiological monitoring technique that is performed by moving a detector over a surface at a specified speed and distance to detect elevated activity areas or locations. Also called a "Surface Scan".

20. Scoping Surveys - Surveys such as investigative surveys used to provide a quick look at conditions before or during FSS work. These surveys are not necessarily documented.

21. Structures - All SNEC Facility site buildings and their surfaces. In addition, platforms, restraints and supports, and external surfaces of piping systems, heating and ventilation systems, tanks, stacks, etc., are also treated as structures in the final status survey if they exist beyond remediation efforts.

22. Surface Contamination - The total of both fixed and removable contamination. For the purposes of this plan, this would also include any remaining neutron-activated material near the surface. Also called total surface contamination.

23. Survey Area - The basic survey entity for the management of the Final Status Survey. It is comprised of one or more survey units, the bounds of which are

defined by existing facility physical features, such as a room, intersection of walls, column-and-row layout of a floor elevation, or structural I-beams.

24. Survey Location - In a structural or open land survey area, a survey location is usually represented by a single grid block. In a system survey area, a specified length of piping or a component such as a valve or tank is referred to as a survey location. A survey location can contain one or more survey points. Also referred to as measurement locations.

25. Survey Unit Release Record - A collection of information in a standardized format for controlling and documenting field measurements taken for the Final Status Survey. A survey unit release record is prepared for each survey area. The survey unit release record may include the survey instructions, a control form, grid map(s), survey measurement data sheets and survey maps. It may also be called a survey package.

26. Survey Point - A smaller subdivision within a survey location (grid block, system, component) where local measurements are taken. For structures and systems, a survey point generally refers to an area covered by a detector, or an area of 100 cm2 when a smear is taken. For open land areas, a survey point refers to the area covered by a detector (for paved surfaces), the point at which a dose rate measurement is taken, or the point at which a soil or pavement sample is collected.

27. Survey Unit - A geographical area consisting of structures or land areas of specified size and shape at a remediated site for which a separate decision will be made whether the unit attains the site-specific reference-based cleanup standard for the designated pollution parameter. Survey units are generally formed by grouping contiguous site areas with a similar use history and the same classification of contamination potential. Survey units are established to facilitate the survey process and the statistical analysis of survey data.

28. Total Effective Dose Equivalent (TEDE) - The sum of the deep dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).

29. Unity Rule - Where more than one radionuclide is present, the sum of the ratios of each radionuclide concentration to its respective DCGL should not exceed unity. When this method is used, the effective DCGL is equal to one (1).

5.9 REFERENCES

5-1 Code of Federal Regulations, Title 10, Part 50.82, "Termination of License"

5-2 Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors," January 1999

5-3 Code of Federal Regulations, Title 10, Part 20.1402, "Radiological Criteria for Unrestricted Use"

5-4 NUREG-1727, "NMSS Decommissioning Standard Review Plan", September 2000.

5-5 NUREG-1575/EPA 402-R-97-016, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," August 2000

5-6 NUREG-1505, "A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys"

5-7 SNEC Facility Site Characterization Report, May, 1996

5-8 NUREG/CR-5512, "Residual Radioactive Contamination From Decommissioning, Final Report," Volume 1, October 1992

5-9 Draft NUREG-1549, "Using Decision Methods for Dose Assessment to Comply With Radiological Criteria for License Termination," July 1998

5-10 Yu, C. F. et al., Manual for Implementing Residual Radioactivity Materials Guidelines Using RESRAD, Environmental Assessment Division, Argonne National Laboratory

5-11 Yu, C. F. et al., RESRAD-Build, A Computer Model for Analyzing the Radiological Doses Resulting from the Remediation and Occupancy of Buildings Contaminated with Radioactive Material. Environmental Assessment Division, Argonne National Laboratory

5-12 Regulatory Guide 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operations) – Effluent Streams and the Environment"

5-13 SNEC Procedure, 1000-PLN-3000.05, "SNEC Facility Decommissioning Quality Assurance Plan"

5-14 SNEC Procedure, E900-PLN-4542.01, "SNEC Radiation Protection Plan"

5-15 SNEC Procedure, E900-ADM-4500.44, "SNEC Facility Calculations"

5-16 SNEC Procedure, E900-ADM-4500.04, "SNEC Records Retention Procedure"

5-17 SNEC Procedure, E900-ADM-4500.12, "Radiological Surveys: Requirements & Documentation Procedure"

5-18 NUREG-1507, "Minimum Detectable Concentrations With Typical Radiation Survey Instruments for Various Contaminants and Field Conditions," June 1998

5-19 SNEC Facility Historical Site Assessment Report, January 2000

5-20 Letter Report, "Use of Two-Stage or Double Sampling in Final Status Decommissioning Surveys", Prepared by C. Gogolak for NRC, February, 2000.

APPENDIX 5-1

ELEVATED MEASUREMENT COMPARISON (EMC)

The EMC, sometimes called a "hot spot test," is a simple comparison of measured values against a limit. There are two applications of this comparison in the final survey process. It is used when the sensitivity of the scanning technique is not sufficient to detect levels of residual radioactivity below the DCGL (i.e., where the MDC_{scan} is greater than the DCGL). In this application, the number of static measurements may need to be adjusted. Appendix 5-2 describes how this is done. The second application in this appendix, is when one or more scan or static measurement data points exceed the DCGL. The use of the EMC for measurements above the DCGL provides assurance that unusually large measurements receive the proper attention and that any area having the potential for significant dose contributions is identified. The EMC is intended to flag potential failures in the remediation process.

Locations, identified by scan or static measurements, with levels of residual radioactivity, which exceed the DCGL, are investigated (see Section 5.4.4). The size of the area where the elevated residual radioactivity exceeds the DCGL and the level of the residual radioactivity within the area are determined. The average level of residual radioactivity is then compared to the $DCGL_{EMC}$. If a background reference area is to be applied to the survey unit, the mean of the background reference area measurements may be added to the DCGL or the $DCGL_{EMC}$ to which the average level of residual radioactivity is compared.

The $DCGL_{EMC}$ is calculated using the following equation (NUREG-1575, Equation 8-1):

$$DCGL_{EMC} = Area\ Factor \times DCGL$$

The area factor is the multiple of the DCGL that is permitted in the area of elevated residual radioactivity without requiring remediation. The area factor is related to the size of the area over which the elevated residual radioactivity is distributed. That area, denoted A_{EMC}, is generally bordered by levels of residual radioactivity below the DCGL, and is determined by the investigation. The area factor is the ratio of dose per unit area or volume for the default surface area for the applicable dose modeling scenario to that generated using the area of elevated residual radioactivity, A_{EMC}. It is calculated based on the methodology given in chapter 8 of NUREG-1505 (Reference 5-6).

If the average level of the elevated residual radioactivity is less than the $DCGL_{EMC}$, there is reasonable assurance the site release criteria is still satisfied and the area does not require remediation. Radioactivity at the $DCGL_{EMC}$ distributed over the area A_{EMC} delivers the same calculated dose as does residual radioactivity at the DCGL distributed over the default surface area. If the $DCGL_{EMC}$ is exceeded, the area is remediated and resurveyed. Area factors for open land areas at the SNEC Facility are provided in Table 5-15. Area factors for surface area DCGLs supplied by the NRC are provided in Table 5-15A

Table 5-15

Area Factors (AF) For Open Land Areas

Based on 25 mrem/y TEDE and Upper 1 Meter Volumetric Surface Modeling

File Names ⇒	NEW XXXXX.RAD*		NEW XXXXXA.RAD		NEW XXXXXB.RAD		NEW XXXXXC.RAD		NEW XXXXXD.RAD		NEW XXXXXE.RAD	
AREA ⇒	10000 m^2		2500 m^2		400 m^2		100 m^2		25 m^2		1 m^2	
Radionuclides	Base DCGL	AF	Implied DCGL EMC	AF	Implied DCGL EMC	AF	Implied DCGL EMC	AF	Implied DCGL EMC	AF	Implied DCGL EMC	AF
Am-241	25.7	1.0	47.7	1.9	110.1	4.3	321.7	12.5	699.1	27.2	3005	116.9
C-14	26.8	1.0	151.1	5.6	984.8	36.7	2 69E+03	100.2	7206	268.9	1.79E+05	6682.8
Co-60	3.5	1.0	4.4	1.3	4.9	1.4	5.4	1.6	7.0	2.0	43.4	12.4
Cs-137	6.6	1.0	14.9	2.3	19.9	3.0	23 8	3.6	31.1	4.7	189.3	28.7
Eu-152	10.1	1.0	10.5	1.0	11.1	1.1	12.1	1.2	15.5	1.5	94.3	9.3
H-3	645	1.0	1.47E+03	2.3	3.23E+03	5.0	7 87E+03	12.2	1.78E+04	27.6	3.55E+05	550.2
Ni-63	747	1.0	3.66E+03	4.9	1.29E+04	17.2	5.14E+04	68.8	2.05E+05	275	5.07E+06	6789.8
Pu-238	30.1	1.0	57.7	1.9	142.9	4.7	408.2	13.6	694.4	23.1	1.08E+04	358.8
Pu-239	6.8	1.0	11.9	1.7	26 9	4.0	56.4	8.3	114.8	16.9	1374	202.1
Pu-241	866	1.0	1607	1.9	3713	4.3	1.09E+04	12.6	2.39E+04	27.6	1.02E+05	118.1
Sr-90	1.2	1.0	3 6	3.0	9.8	8.1	38.5	32.1	146.7	122.3	2826	2355

* Where "XXXXX" is the radionuclide computer file name, as an example "Am241".

NOTE 1: Base case DCGLs (in pCi/g) are for 10,000 square meter surface model only.

NOTE 2: The above set of DCGL values are used only to determine the Area Factors (AF) that will then be applied to the values listed in Table 5-1 (surface materials only).

NOTE 3: When AF values are calculated in the RESRAD computer code, the settings for contaminated fractions for plant food, meat and milk must be re-set to their default condition (-1) in order to allow the computer code to scale the food supply for the size of the areas appropriately.

Table 5-15A

Area Factors For Structural Surfaces

(Based on NRC Screening Values - see Table 5-1)

Nuclide	36 m^2	25 m^2	16 m^2	9 m^2	4 m^2	1 m^2
Am-241	1	1.5	2.3	4.1	9.2	36.2
C-14	1	1.4	2.2	4.0	8.9	35.9
Co-60	1	1.2	1.5	2.0	3.4	10.1
Cs-137	1	1.2	1.5	2.2	3.7	11.2
Eu-152	1	1.2	1.5	2.1	3.5	10.7
H-3	1	1.4	2.2	4.0	8.9	35.8
Ni-63	1	1.4	2.2	4.0	9.0	35.3
Pu-238	1	1.4	2.3	4.0	9.1	36.9
Pu-239	1	1.4	2.2	4.0	9.0	35.4
Pu-241	1	1.4	2.2	4.0	9.0	34.8
Sr-90	1	1.4	2.2	1.5	8.8	34.7

NOTE: DCGL is in dpm/100 cm^2

APPENDIX 5-2

STATISTICAL INFORMATION

The method described in NUREG-1575 (Reference 5-5) and NUREG-1505 (Reference 5-6) for determining the number of survey measurements necessary to assure a population set sufficient for statistical analysis is summarized here in the manner it is applied at the SNEC Facility. An effective survey design slightly overestimates both the number of survey measurements and the standard deviation to ensure adequate power of the statistical test. This ensures that a survey unit is not subjected to additional remediation simply because the final survey is not sensitive enough to detect that residual radioactivity is below the DCGL.

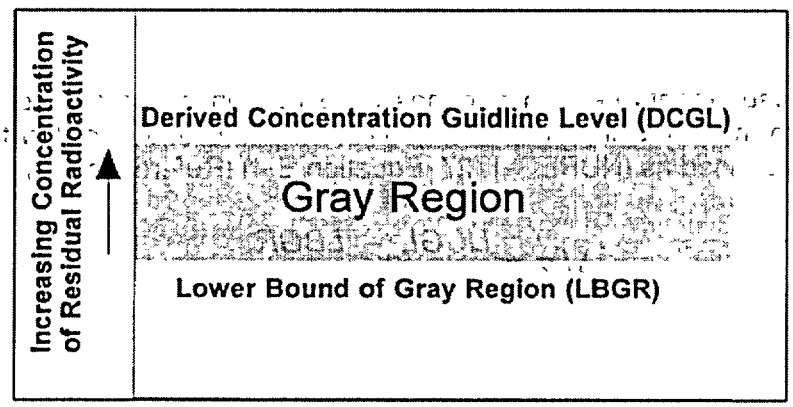
TERMS AND STATISTICAL PARAMETERS

A minimum number of measurements are needed to obtain sufficient statistical confidence that the conclusions drawn from the survey data are correct. Several terms and statistical parameters are described in this chapter that are used in determining the number of measurements needed to apply the statistical tests.

Lower Bound of Gray Region (LBGR)

The LBGR is the concentration to which the survey unit must be remediated in order to have an acceptable probability of passing the statistical test for meeting the site release criteria. It represents the lower bound of the area of uncertainty regarding the concentration of residual radioactivity in the survey unit. The DCGL represents the upper bound. The width of the gray region is equal to the difference between the DCGL and the LBGR, as illustrated in the Figure 5-3 below.

Figure 5-3
Gray Region Diagram



Initially, the LBGR is arbitrarily set at 0.5 times the DCGL. The LBGR is adjusted as needed to result in a relative shift of 1 to 3 (see NUREG-1575 and NUREG-1505 documents for full description of terms (Reference 5-5 and 5-6)).

The survey design goal is to achieve an LBGR between 0.5 and 1 times the DCGL. Since the LBGR serves as the remediation goal for decommissioning activities, generally the smaller the LBGR, the more rigorous are the requirements for dismantlement, decontamination, and post

remediation surveys. As the LBGR approaches the value of the DCGL and the band of uncertainty narrows, the number of samples needed to demonstrate compliance with the site release criterion rises dramatically.

Standard Deviation (σ)

The standard deviation, denoted by σ, represents the spatial variability in the concentration of the residual radioactivity in the survey unit. The mean and standard deviation are calculated using standard equations.

An estimated value of σ may be calculated either from existing data or by taking limited preliminary measurements of the concentration of the residual radioactivity in the survey unit (sometimes called a pilot or scoping study). It is also acceptable to assume that σ equals 0.3 times the mean concentration. Alternatively, a reasonable estimate based on available site knowledge may be used.

The value selected as an estimate of σ for the survey unit may be based on data collected only from within that survey unit or from data collected from a much larger area of the site, since there may be some difficulty in determining which individual measurements from a preliminary survey may later represent a particular survey unit. The most practical solution may be to estimate σ for each area classification (Class 1, Class 2 and Class 3). If there are multiple types of materials within an area classification, additional estimates of σ may be required.

A separate estimate of σ may be obtained for every background reference area. If the σ in the background reference area is larger than the σ in the survey unit, the larger value may be used.

The survey design goal is to avoid an estimated σ that is overly optimistic or conservative. If the value is grossly underestimated, the number of measurements will be too few and may result in unnecessary remediation or re-survey. If the value is grossly overestimated, the number of measurements will be unnecessarily large.

Relative Shift (Δ/σ)

The number of measurements needed depends on a ratio involving the concentration to be measured relative to the variability in the concentration. The ratio is called the relative shift, denoted by Δ/σ. It is defined as (NUREG-1727, Equation E-4) (Reference 5-4):

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

Where:

The variables have been previously defined.

The survey design goal is to achieve Δ/σ values between 1 and 3. The number of measurements needed rises dramatically when Δ/σ is smaller than one. Conversely, little is usually gained by making Δ/σ larger than about three. If Δ/σ is greater than three, the LBGR should be increased until the relative shift is equal to three.

DECISION ERRORS

The principal study question or statement is, **"are the levels of residual radioactivity in all survey units below applicable release criterion and can the site be released?"** Results from surveys and other environmental testing will be used to determine the answer to this question.

A decision error is the probability of making an error in the decision on a survey unit, either passing a survey unit that should fail or failing a survey unit that should pass. The first decision error, passing a survey unit that should fail, is referred to as a false positive or TYPE I decision error. The probability of making this error is denoted by α. Setting high value for α results in a higher risk of passing a survey unit that should fail. Setting low value of α lowers the risk of passing a survey unit that should fail.

The second decision error, failing a survey unit that should pass, is referred to as a false negative or TYPE II decision error and is denoted by β. Selecting a high value for β results in a higher risk of failing a survey unit that should pass and subjecting it to further investigation. Selecting a low value for β lowers the risk and minimizes these investigations. The cost of setting a low value for either α or β is a higher value for the other or an increased number of measurements to demonstrate compliance with the release criteria.

When using the statistical testing procedures as described in NUREG-1575 and NUREG-1505 (Reference 5-5 and 5-6) documents i.e., the Sign Test or the Wilcoxon Rank Sum (WRS), larger decision errors may be unavoidable when encountering difficult or adverse conditions. This is particularly true when trying to measure residual radioactivity concentrations close to the variability in the concentration of those materials in natural background. In order to avoid an unreasonable number of samples when Δ/σ is very small, larger values of α may be considered as shown in Table 5-16 below.

Table 5-16

Acceptable Decision Error α as a Function of DCGL

DCGL/σ	α
>3	0.05
1.2 to 3	0.10
0.6 to 1.2	0.25
<0.6	0.30

Table 5-16 values are based on the assumption that the LBGR should not have to be set to less than 0.5 times the DCGL, and that if α is allowed to increase, β will also be allowed to increase.

There are no constraints on the value of β. However, decreasing β increases the number of samples needed, making vary small values of β unattractive.

The survey design objective is then to establish the value of α equal to or less than 0.05 and to minimize the value of β while maintaining the minimum number of measurements at an optimal number. NRC approval is required to increase the α (type 1 decision error) >0.05, in accordance with License Condition 2.E.(h).

NUMBER OF MEASUREMENTS

The statistical parameters α, β and Δ/σ are used to estimate the number of measurements that will produce the desired values of α and β. The number of measurements are based on the statistical test which is applied to the survey unit. The two statistical tests used in the final survey data analysis process are the Sign Test and the Wilcoxon Rank Sum (WRS) Test. The criteria for using these testing procedures are summarized in Table 5-17.

Table 5-17

Statistical Tests and Criteria For Their Use

Statistical Test	Criteria for Use
WRS Test	Radionuclide of concern appears in background, or measurements are used that are not radionuclide-specific.
Sign Test	Radionuclide of concern is not present in background and radionuclide-specific measurements are made, or radionuclides are present in background at such small fractions of the DCGL as to be considered insignificant.

NOTE: For specific information on statistical testing procedures, see Table 2 3 of NUREG-1505 (Reference 5-6).

The number of measurements is determined by rounding up the number calculated using the appropriate statistical test and adding 20% more measurements. Additional measurements are added to protect against the possibility of lost or unusable data.

Wilcoxon Rank Sum (WRS) Test

The two-sample WRS test is used when the radionuclide of concern appears in background or if measurements are used that are not radionuclide specific. Because gross activity measurements are not radionuclide specific, they must be performed for both the survey unit(s) being evaluated by the WRS test and for corresponding reference area(s). The number of measurements needed for the WRS test is determined from the following equation (NUREG - 1727, Equation E-5) (Reference 5-4):

$$n = (1/2)\frac{(Z_{1-\alpha} + Z_{1-\beta})^2}{(3)(P_r - 0.5)^2}$$

Where:

n = number of measurements in survey unit

$Z_{1-\alpha}$ = percentile represented by decision error α (NUREG-1575, Table 5.2)

$Z_{1-\beta}$ = percentile represented by decision error β (NUREG-1575, Table 5.2)

P_r = probability that a random measurement from survey unit exceeds random measurement from background reference area by less than DCGL when

survey unit median is equal to LBGR concentration above background (NUREG-1575, Table 5.1)

½ = factor included in Equation 5-1 of NUREG-1575 to define *n* as the number of measurements in a survey unit

Additional n measurements are also needed in the background reference area.

Sign Test

The one sample Sign test is used if the radionuclide of concern is not present in background and radionuclide specific measurements are made. The Sign test may also be used if one or more radionuclides are present in background at such small fractions of the DCGL as to be considered insignificant. In this case, background concentrations of the radionuclides are included with the residual radioactivity (in other words, the entire amount is attributed to facility operations). Thus, the total concentration of radionuclides are compared to the release criteria. This option is only used if it is expected that ignoring the background concentration will not affect the outcome of the statistical tests. The advantage of ignoring a small background concentration is that no background reference area is needed, which simplifies the final survey considerably.

Since SNEC Facility radionuclides outside the SNEC CV are largely composed of Cs-137 and Co-60, which are relatively easy to measure, the Sign test is the most likely statistical testing procedure to be employed. Inclusion of Cs-137 and Co-60 background values into the DCGLs should not significantly effect meeting site release criteria.

The number of measurements needed for the Sign test is determined from the following equation (NUREG-1727, Equation E-6) (Reference 5-4):

$$n = \frac{(Z_{1-\alpha} + Z_{1-\beta})^2}{4(\text{Sign } \rho - 0.5)^2}$$

Where:

Sign ρ = estimated probability that random measurement for a survey unit is less than the DCGL when the survey unit median concentration is actually at the LBGR (Table 5.4 of NUREG-1575 (Reference 5-5))

All other variables have been previously defined.

Elevated Measurement Comparison (EMC)

The EMC is used to determine if additional measurements may be needed when the level of residual radioactivity that can be detected by scanning (MDC_{scan}, see Section 5.5.2.5) is larger than the DCGL. The WRS and Sign tests evaluate whether or not the residual radioactivity exceeds the DCGL for contamination that is approximately uniform across the survey unit. These tests may not successfully detect small areas of elevated contamination. Instead, systematic measurements, in conjunction with surface scanning, are used to obtain adequate assurance that small areas of elevated radioactivity will meet the DCGL. When the MDC_{scan}

exceeds the DCGL, the EMC provides the reasonable level of assurance that any small areas of elevated residual radioactivity that could be significant are not missed during the final survey.

The number of measurements needed to detect an elevated concentration, n_{EMC}, in a survey unit area, A, is (NUREG-1727, Equation E-8) (Reference 5-4):

$$Area\ \ Factor = \frac{MDC_{scan}}{DCGL}$$

Once the area factor is determined, the corresponding value for A_{EMC} can be obtained (see Appendix 5-1) and the value of n_{EMC} calculated.

If n_{EMC} is larger than n, additional samples (up to a total in the survey unit of as many as n_{EMC}) may be needed to demonstrate that areas of elevated residual radioactivity meet the site release criteria. In cases where n_{EMC} is larger than n, the site characterization should be considered and, based on what is known about the site, it may be possible to estimate a concentration that is unlikely to be exceeded. This maximum concentration may be converted into an area factor (multiple of the DCGL), and then the corresponding A_{EMC} value obtained (see Appendix 5-1) and used in the above equation. Similarly, based on knowledge of the site it may be possible to estimate the smallest area likely to have elevated levels of residual radioactivity. If this is so, that area can be used in the above equation. Likewise, knowledge of how the residual radioactivity would be likely to spread or diffuse could determine an area to be used in this equation.

The EMC is only applied to Class 1 survey units, since areas of elevated residual radioactivity should not be present in Class 2 or Class 3 survey units.

Optimizing the Number of Measurements

Once the acceptable design values have been established, survey design constraints may be changed to evaluate how these changes affect the number of measurements for several basic measurement designs. The survey design constraints are:

- Limits on the decision error probabilities α and β;
- Width of the band of uncertainty or gray region (by adjusting the LBGR); and
- Survey unit boundaries (it may be possible to reduce the number of measurements by changing or eliminating survey units that may require different decisions).

The process may be iterative in that the initial values selected must be modified to allow dependent variables to fall within the survey design constraints.

Selecting a Minimum Number of Measurements

As discussed above, the MARSSIM process incorporates design constraints that ensure that an adequate number of sample measurements are taken per survey unit. However, to simplify the final survey process and to ensure conservatism without an associated unreasonable expenditure of resources, a minimum number of 30 sample measurements per survey unit may be collected for survey units less than 10 m^2 whenever possible.

GRID SIZE DETERMINATION AND DESIGN

The sample size, determined in previous sections, provides the necessary input on setting sampling locations and survey patterns. The method is outlined in section 5.5.2.4 of NUREG-1575 (Reference 5-5). The reference grid system on which sampling patterns are superimposed, is a basic alpha numeric x-y layout unless otherwise determined to be impractical because of actual field conditions. Table 5-16 presents a summarized presentation of typical survey design parameters.

APPENDIX 5-3

BACKGROUND REFERENCE AREAS

Background reference areas are used if: (1) the residual radioactivity contains a radionuclide that occurs in background, or (2) the measurements to be made are not radionuclide-specific. Background reference areas are not needed when radionuclide-specific measurements are used to measure concentrations of a radionuclide that is not present in background. They also are not needed when one or more radionuclides are present in background at such small fractions of the DCGL as to be considered insignificant. Surveys are conducted in one or more background reference areas to determine background levels for comparison with the conditions determined in specific survey units.

SELECTION

Background reference areas are selected which have similar physical, chemical, radiological, and biological characteristics as the survey unit being evaluated. They are usually selected from non-impacted areas, but are not limited to natural areas undisturbed by human activities. Generally, background reference areas are not part of the survey unit being evaluated. However, where necessary, they may be associated with the survey unit being evaluated, but cannot be potentially contaminated by site activities. For example, background measurements may be taken from core samples of concrete, pavement, or other types of surface materials. Occasionally, the survey unit itself may serve as the background reference area when a surrogate radionuclide in the survey unit can be used to determine the background. For example, it may be possible to use the measured alpha- or beta-emitting fraction of an established radionuclide distribution in embedded piping to calculate a net activity and subtract it from a gross activity to determine background levels.

For materials present on-site, either in buildings or as non-soil materials present in outdoor survey units (e.g., concrete, brick, drywall, fly ash, petroleum product wastes, etc), background reference areas of non-impacted materials that are as similar as possible to the materials on-site are used. Sometimes such materials are not available. In those situations, a good faith effort is made to find the most similar materials readily available or use appropriate published estimates.

Each background reference area should have an area at least as large as the survey unit, if practical, in order to include the full potential spatial variability in background concentrations.

MARKEDLY DIFFERENT SURVEY UNIT MATERIAL BACKGROUNDS

Survey units may sometimes contain a variety of materials with markedly different backgrounds. An example might be a room with drywall walls, concrete floor, glass windows, metal doors, wood trim, and plastic fixtures. A separate survey unit is not made for each material. If one material is predominant or if there is not too great a variation in background among materials, a background reference area comprised of the predominant material may be appropriate. For example, a room may be mostly concrete but with some metal beams, and the residual radioactivity may be mostly on the concrete. In this situation where the presence of concrete predominates, the background for concrete is used for the room. If a measurement location on the random-start grid falls on the metal beam, the measurement location may be moved to be on the nearest piece of concrete. Alternately, the measurement could be taken on the metal, but this is then noted on the survey record so that an unusual reading can be explained.

When there are different materials with substantially different backgrounds in a survey unit, a non-impacted room with roughly the same mix of materials may be used as a background reference area. Alternately, measured backgrounds for the different materials or for groups of similar materials may be used. In this case, it is acceptable to perform the Sign test on the difference between the paired measurements from the survey unit and from the appropriate background material.

MARKEDLY DIFFERENT BACKGROUND REFERENCE AREAS

If significant differences in backgrounds among background reference areas are found, a value of three times the standard deviation of the mean between the background reference areas is added to the mean background for all background reference areas to define a background concentration. The value of three times the standard deviation of the mean is chosen to minimize the likelihood that a survey unit that contains only background would fail the statistical test for release. The WRS test is then used to test whether the survey unit meets the radiological criteria for license termination.

BACKGROUND LEVEL DETERMINATIONS

In general, background levels will be established (as needed) for each type of instrument used for total surface contamination measurements, removable contamination measurements, and gamma exposure rate measurements. In addition, backgrounds will be determined for specialized detectors and detector systems on an as needed basis. These background measurements may include large area detector backgrounds for floor monitoring instrumentation, background determinations for detectors used for surveying piping interiors and background values for gamma-ray spectroscopy systems.

For soil and sediment samples, background levels will be determined for man-made radionuclides not resulting from plant activities.

The objectives of background determinations for the SNEC Facility final status survey measurements will be to:

- Establish reference background values for each type of detector system used in the SNEC Facility FSS survey.
- Assess the variability in background responses for principal detectors under different applications and conditions of use.
- Determine the need for correction factors or special measurements to establish the background for final status survey measurements in specific locations.
- Account for man-made radioactivity not resulting from plant operations.
- Measure reference values at different directions and distances from the center of the SNEC Facility outward, including down wind directions.

Background determinations will be performed in accordance with approved procedures and survey request work package guidance. Background values will be evaluated in accordance with NUREG-1575 (Reference 5-5) guidance. Methods used to determine background values for each type of final status survey measurement method are summarized below.

Background measurements will be performed on surfaces unaffected by licensed activities. The principal criteria for selection of measurement locations for building surfaces will be the location's similarity to SNEC Facility construction and freedom from radioactivity attributable to SNEC Facility operations. Instrumentation will be of the same type to be used for the SNEC Facility FSS process.

Soil and sediment samples will be collected from areas unaffected by licensed operations in order to establish the background levels of man-made radionuclides not resulting from plant operations. The background samples will be collected at locations similar to their respective on-site sampling locations whenever possible. Gamma-ray spectroscopy will be used to determine individual concentrations of radionuclides. Additionally, laboratory analysis for other non-gamma emitting radionuclides will be performed as necessary.

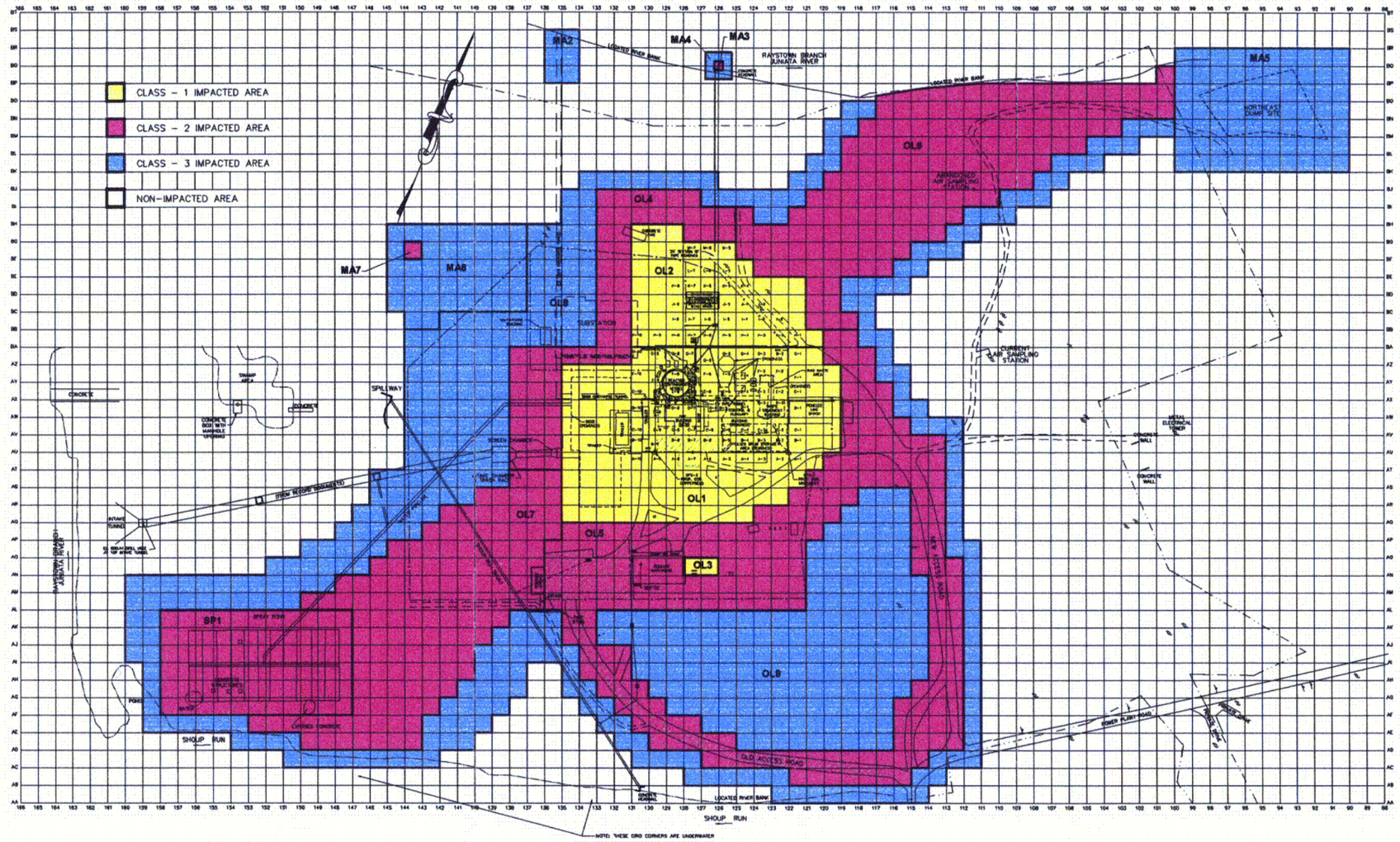
A pressurized ion chamber (PIC) will be used to establish the background gamma exposure rate for the site. The pressurized ion chamber will be used as the reference instrument for establishing gamma-ray exposure rate background and the true μR/h equivalent response of portable survey meters. Additionally, a Micro-Rem[a] Meter (or equivalent) will be used to evaluate site dose rates. Measurements obtained with this instrument will be compared to PIC measurements at selected locations. The Micro-Rem Meter will then be used for the bulk of the final status survey gamma-ray exposure rate measurements at the site.

Compiled background information will be used as the reference areas for most survey units as necessary. Where actual background conditions are shown to vary significantly, area-specific background values will be used whenever possible.

Alternatively, *in-situ* gamma spectroscopy may be used to identify those components of the background gamma radioactivity that are possibly due to plant contamination.

Background values of direct beta-gamma measurement instruments are affected by conditions in the immediate vicinity of the detector. Significant variations from background reference values may be observed. These variations are caused by the natural radioactivity composition in surrounding materials and by shielding effects in some cases. Consequently, background measurements for special conditions will be compiled for use in calculating values (net dpm/100 cm^2, etc.) to reduce the bias in survey unit population statistics. Special condition backgrounds may be compiled for poured concrete, steel and iron, aluminum and any other material as deemed appropriate.

[a] Micro-Rem Meter is product of the Bicron Instrument Company.



GRAPHIC SCALE

(IN METERS)

1 : 800

LEGEND

- IP & ALLOY CAP SET FOR SAMPLING GRID OR AT STRUCTURE CORNERS
- P.K. NAIL SET FOR SAMPLING GRID
- PROPERTY LINE
- FENCE
- EXISTING STRUCTURE
- EDGE WATER
- STORM MANHOLE
- TOILET
- DEMOLISHED STRUCTURE (FIELD LOCATED W/W IP AT CORNERS)
- DEMOLISHED STRUCTURE (NOT FIELD LOCATED)
- PAVEMENT
- DROP INLET
- SIGN
- UTILITY POLE

NOTE: BLOCKS A-1 THRU M-7 PINS SURVEYED OCTOBER 1983

NO.	DWG. NO.	TITLE
REFERENCES		

REVISIONS

REV	ZONE	DESCRIPTION

C01

Cad File Name : SNECRM-020_S1_R1

FIGURE 5-1

1 GRID BLOCK = 10 SQUARE METERS

THIS IS A COMPUTER GENERATED DRAWING DO NOT REVISE IT MANUALLY

GPU NUCLEAR

DRAWN: B. SCHABLE	DATE: 08/23/02	
DESIGNED	DATE	
TECHNICAL MANAGER	DATE	
THERMOST	DATE	
ENGINEERING MANAGER	DATE	

SNEC FACILITY

SITE AREA GRID MAP

GRADE ELEV. 812'

TMI-1

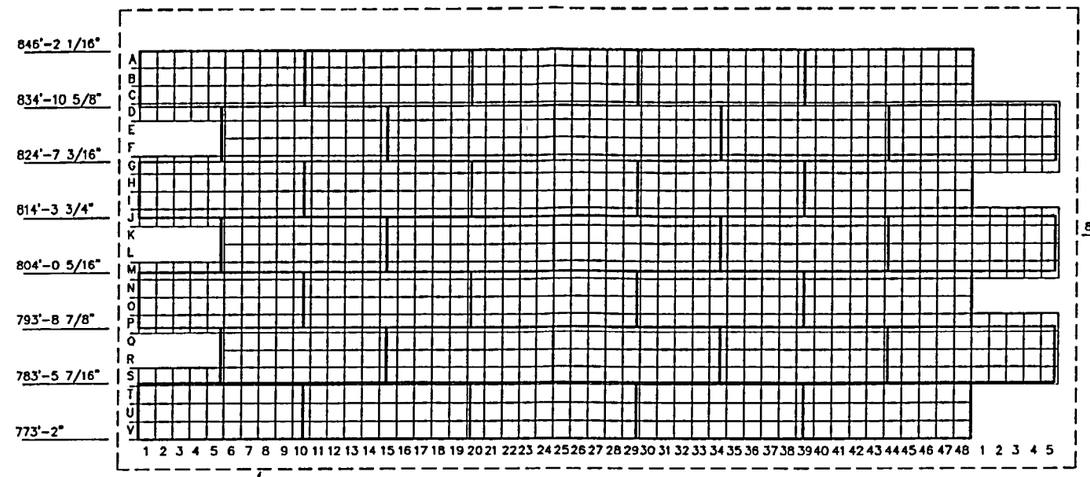
DWG. NO.	SA	REV.
SNECRM-020	1	1

SCALE:

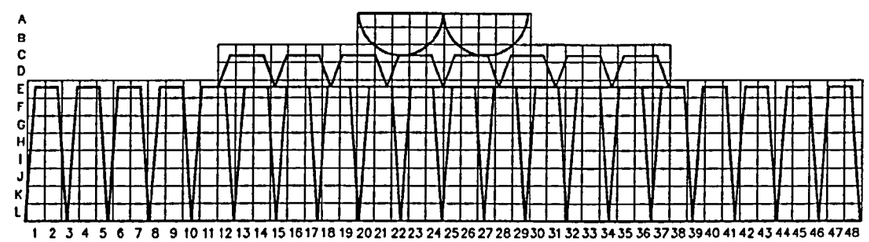
INTERFACING CONCURRENCE

SA NO. --

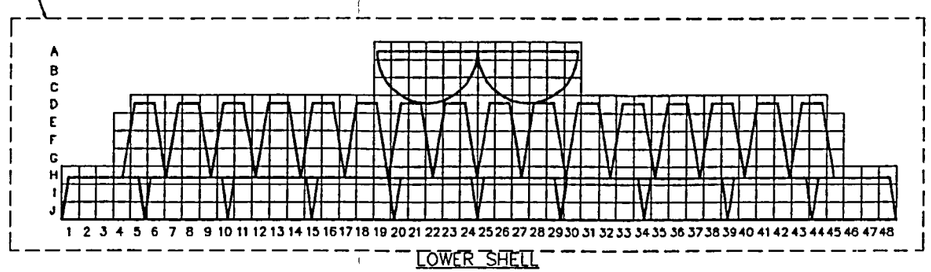
SNECRM-020



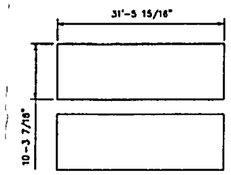
DOTTED LINE DENOTES SECTIONS OF CV SHELL THAT WILL BE SURVEYED

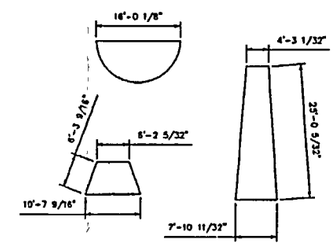


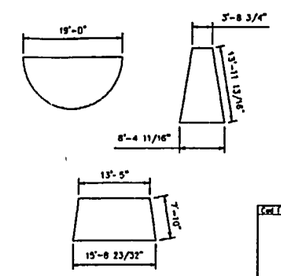
DOME



LOWER SHELL







REV	ZONE	REVISIONS DESCRIPTION

Cad File Name: SNECRM-019_S1_R0

FIGURE 5-4
EXAMPLE OF STRUCTURAL GRIDING
(1 GRID = 1 sq. METER)

THIS IS A COMPUTER GENERATED DRAWING DO NOT REVISE IT MANUALLY

GPU NUCLEAR

R.A.FOCH 01/18/02

SNEC FACILITY
CONTAINMENT VESSEL SURVEY
INTERNAL / EXTERNAL GRID

TMI-1 | DWG. NO. SNECRM-019 | SH. 1 | REV. 0

SCALE

INTERFACING CONCURRENCE | SA NO. -

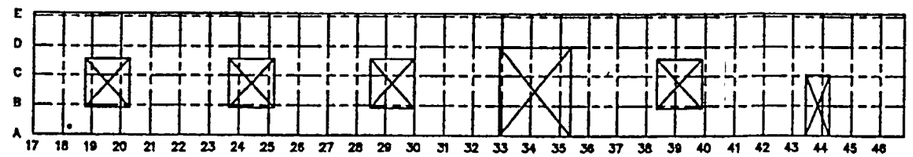
NO.	DWG. NO.	TITLE

REFERENCES

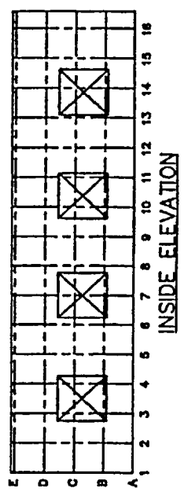
PROJECT SNECRM-019 SH. 1

DWG.NO. SNECRM-013 SH. 1

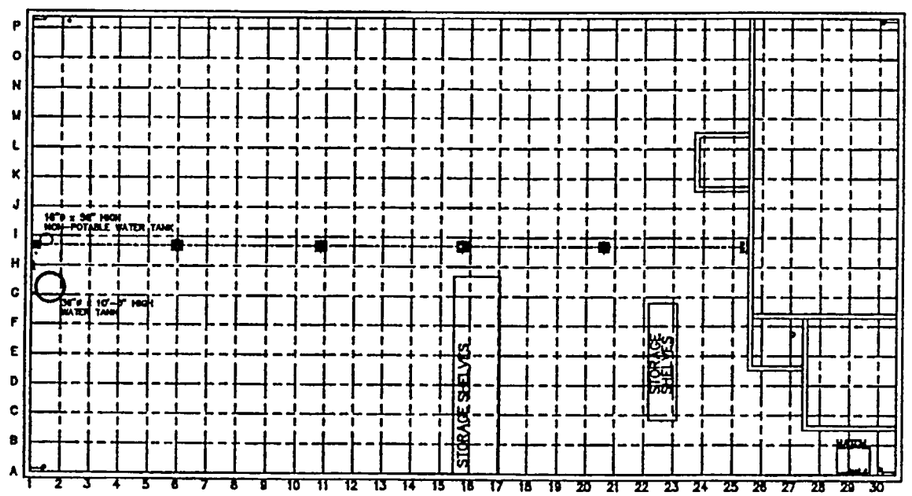
REVISIONS		
REV	ZONE	DESCRIPTION



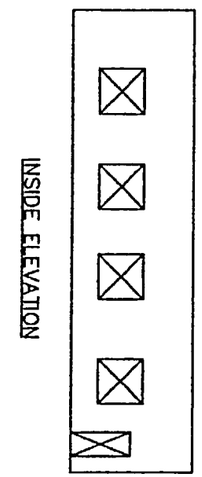
INSIDE ELEVATION



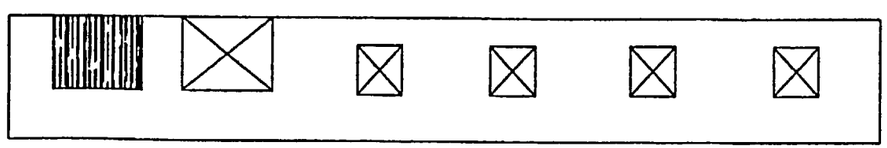
INSIDE ELEVATION



FLOOR PLAN



INSIDE ELEVATION



INSIDE ELEVATION

NOTES:

1. SEE SNECRM-013 SH. 3 FOR ELEVATIONS DENOTED BY ▸.

NO.	DWG. NO.	TITLE
	REFERENCES	

Cad File Name : SNECRM-013_S1_R0

FIGURE 5-5
EXAMPLE OF STRUCTURAL GRIDING
(1 GRID = 1 sq. meter)

THIS IS A COMPUTER GENERATED DRAWING DO NOT REVISE IT MANUALLY

GPU NUCLEAR

DRAWN	R. SCHABLE	07/15/99
CHECKED	R.A.KOCH	01/17/00
ENGINEER		
ENGINEERING MANAGER		

SNEC FACILITY
PENELEC WAREHOUSE
FLOOR PLAN AND EXTERIOR WALL ELEVATIONS

TMI-1	DWG. NO. SNECRM-013	SH. 1	REV. 0
	SCALE: 1/8"=1'-0"		

INTERFACING CONCURRENCE

BA NO.

6.0 COMPLIANCE WITH RADIOLOGICAL CRITERIA FOR LICENSE TERMINATION

6.1 DISCUSSION

The SNEC decommissioning objective is to release the site for unrestricted use per 10 CFR 50 and Subpart E of 10 CFR Part 20.1402. This section of the SNEC LTP will briefly discuss the methods and assumptions used to demonstrate that the total effective dose equivalent (TEDE) from residual radioactivity is less than 25 mrem per year (0.25 mSv) to an average member of the critical group. References 6-5, 6-6 and 6-7 have been used to determine the appropriate methodologies for meeting site release criteria. These methodologies are also described in Chapter 5 of this plan.

6.2 DOSE MODELING

The distribution and variability of radioactivity in the environment, and dose rates from natural sources, provide the framework for developing models in determining compliance with criteria for site cleanup and restoration. The dose limit of 25 mrem TEDE in any year is of approximately the same magnitude as the geographic variability of doses from natural background. It is comparable to the difference in annual doses likely to be experienced by a person who moves from one location to another. The TEDE, adopted in1991 by the NRC, is the sum of the deep dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures). For site cleanup and restoration standards, the dose limit is applied to all site-specific sources, including natural radionuclides whose concentrations have been enhanced by site activities.

Dose modeling is used to estimate the TEDE to an average member of the critical group from residual radioactivity. The critical group means the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity for any applicable set of circumstances.

The surface contamination and radionuclide concentrations levels of structures, land areas, and plant systems remaining at the time of the final site survey are compared to derived concentration guideline levels (DCGL) calculated using applicable dose models. A DCGL is defined as the concentration of residual radioactivity distinguishable from background radiation which, if distributed uniformly throughout a survey unit, would result in a TEDE of 25 mrem/yr to an average member of the critical group. DCGLs are presented in terms of surface and volumetric radioactivity concentrations and are expressed in units of dpm/100 cm^2 or pCi/g.

Data and information related to types and amounts of radioactive material on the SNEC site have been documented in the Site Characterization Report (Reference 6-18) and Chapter 2 of this plan. Prior to site release consideration, all radioactive waste will have been disposed, sources will have been disposed or scheduled to be transferred to another licensee, all buildings and open land areas remediated, and minor amounts of contamination will have been detected and documented via surveys.

Dose models are based on two types of conceptualizations, i.e. two-dimensional surface area geometry and a three-dimensional volumetric geometry. In certain cases, e.g. embedded pipes, a combination of dose models will be employed to calculate the radiation dose from surface contamination and volumetric sources. Dose model scenarios are discussed below.

6.2.1 Building Occupancy Scenario

For the two-dimensional surface area model, the building occupancy scenario is used as the best representation. This model applies to the SNEC Containment Vessel (CV), Saxton Steam Generating Station (SSGS), tunnels and/or other impacted site structures. Since these structures will either be demolished or left behind, the building occupancy scenario is most representative. Other buildings (garages and support structures) left behind will also be surveyed and released using the building occupancy scenario. DandD Version 1.0 has been used as the preferred modeling software for the building occupancy scenario.

The exposure pathways selected for analysis in the building occupancy scenario include external exposure to penetrating radiation from surface sources, inhalation of resuspended surface contamination, and inadvertent ingestion of surface contamination. The selection of these pathways provides a balanced analysis for:

a) External exposure to penetrating radiation from surface sources;

b) Inhalation of resuspended surface contamination;

c) Inadvertent ingestion of surface contamination; and

d) External exposure to penetrating radiation from embedded sources.

This scenario accounts for exposure to both fixed and removable thin-layer or surface radioactivity within a structure. This scenario assumes individuals (critical group) occupy the building in a passive mode without deliberately disturbing the residual radioactivity on building surfaces. Occupancy of the building is assumed to begin immediately after license termination. The exposure duration is assumed to be a full work year (approximately 2000 hours). The selection of this scenario assumes the following site conditions will exist to qualify:

1. The contamination on building surfaces (e.g. walls, floors, and ceilings) should be surficial and non-volumetric.

2. Contamination on surfaces is mostly fixed (not loose) with the fraction of loose contamination not to exceed 10% of the total surface activity.

3. This screening criteria will not apply to contaminated surfaces on buried components or conduit (e.g. drainage or sewer pipes). Such component and conduit surfaces will be treated on a case by case basis

Surface contamination DCGLs were developed through the use of the DandD computer code using the standard default parameters. The results from the DandD runs for each of the 11 site related radionuclides are in units of mrem TEDE per dpm/100 cm^2. Default surface area DCGLs (Reference 6-8) are listed in Table 6-2. These default values were verified using DandD. The DandD output for the building occupancy scenario model is contained in Appendix 6-1.

Building and component surfaces will be decontaminated to as low as reasonably achievable to meet the release criteria building occupancy screening levels. For building debris that cannot be assessed using the surface screening criteria, a volumetric assessment of the materials will be made. These materials will be characterized (if not already known) as to radionuclide contaminant(s), type of material, corrosivity rates (metals), and leach rates (concrete/debris) for

input. This switch to volumetric consideration brings the resident farmer scenario back as the release scenario. Since some of the material will be buried 3 feet below grade, the contamination zone may be in the saturated zone. A subsurface volumetric dose model has been developed to evaluate this condition.

Exposure pathway (d) listed above applies to areas where there is penetrating radiation from embedded sources of radioactivity, such as embedded piping. To the extent practical embedded sources will be filled with grout or concrete. For modeling this scenario a bounding calculation using sum of the fractions method will be employed. This method will combine applicable surface and volumetric DCGLs along with a shielding code (e.g. MicroShield) to calculate the respective dose from residual activity remaining on structural surfaces, within residual piping, walls and floors or within activated metal (e.g. CV steel liner). Use of Equation 6-1 will ensure the combined exposure is bounded for all applicable sources and result in less than the 25 mrem/yr limit.

Equation 6-1

$$\sum_{i=1}^{n}\left(\frac{C_{si}}{DCGL_{si}}+\frac{C_{vi}}{DCGL_{vi}}\right)+\left[\frac{Direct\ \gamma\ Dose}{25}\right]\leq 1$$

Where: C_{si} = Surface contamination of radionuclide i (dpm/100 cm^2).

C_{vi} = Specific volume concentration of radionuclide i (pCi/g).

$DCGL_{si}$ = Surface contamination DCGL of radionuclide i from Table 6-2.

$DCGL_{vi}$ = Volumetric DCGL (25 mrem/yr) of radionuclide i from Table 6-2.

Direct γ Dose = Microshield (or equivalent) shielding code calculation (mrem/yr).

6.2.1.1 Surface Area Factors

Surface area factors have been developed using comparative analyses between DandD, 1.0 and RESRAD-BUILD, 3.0. Derivation of these area factors has been documented in Reference 6-10. These area factors have been used to develop $DCGL_{EMC}$ screening values for residual radioactivity on building surfaces. Default surface area screening values (Reference 6-8) were used as inputs into the RESRAD-BUILD, 3.0 program to determine the annual default dose at 36 m^2. This dose was then used to ratio against doses calculated for 25, 16, 9, 4, and 1-m^2 areas. The calculated ratio is equal to the area factor value for the respective area sizes. The surface area DCGL can be multiplied by the derived area factor to determine the $DCGL_{EMC}$. Surface area factors for SNEC are listed in Chapter 5, Table 5-15A.

6.2.2 Resident Farmer Scenario

For this scenario the assumption is that residual radioactivity is distributed in a surface soil layer covering the plant site (surface model) or in subsurface fill materials (subsurface model). The receptor is considered to reside in a home in or near any of the areas of concern. Use of the site is for residential or light farming activities. The scenario assumes continuous exposure via multiple exposure pathways to the critical group. The critical group is the resident farming family who lives on the plant site following site remediation, grows some portion of their diet on

the site, and drinks water from a source at the site. The most conservative parameters are selected from each of the areas of concern to identify a site-wide residential scenario, which results in the highest exposure. This site-wide exposure is then used to determine nuclide-specific DCGLs for each surface and subsurface layer. The pathways that apply to the residential farming scenario include:

a) External exposure (while indoors and outdoors) to penetrating radiation from volume sources in the contamination layer;

b) Inhalation of resuspended surface sources

- through wind erosion while indoors or outdoors,
- tracked indoors,
- while excavating and spreading contaminated overburden material during home construction and yard leveling;

c) Ingestion of drinking water from a groundwater source (e.g. bedrock well);

d) Ingestion of plant products grown in contaminated soil and/or irrigated with contaminated groundwater;

e) Ingestion of animal products (e.g. beef and milk from cattle raised onsite that ingested contaminated drinking water, plant products and soil);

f) Direct soil ingestion;

g) Ingestion of fish from a contaminated surface water source; and

h) Direct exposure from re-excavated volume sources.

At SNEC, the shallow water table and boulders in the overburden layer discourage construction of a basement for an on-site residence. However, excavation and spreading of fill material from beneath the top meter and into the upper overburden layer could occur in leveling sloped areas for a home site. This scenario was analyzed as part of the subsurface modeling.

Two models have been developed covering surface (Reference 6-9) and sub-surface (Reference 6-11) open land areas for the Resident Farmer scenario. Both models were developed using the RESRAD Version 6.1 computer code using the deterministic and probabilistic options. GPU Nuclear, Inc. developed the surface model while URS Corporation developed a sub-surface model, incorporating many of the same input parameters used in the surface model. Due to the voluminous nature of the dose modeling results documentation has been included in electronic media (CD-ROM) and submitted to the NRC for review (Reference 6-12). The dose modeling approach and input parameter selection are illustrated in Figure 6-1. General approaches and selection of key input parameters are discussed in the following sub-sections.

DCGL results were compared between the two models. **The most conservative DCGL values were combined to form a single list for the 25 mrem/yr release limit. The most conservative DCGLs to implement SNEC's 4 mrem/yr drinking water dose goal were similarly derived.** These DCGL values are listed in Table 6-2.

6.2.2.1 Probabilistic Approach

For each radionuclide RESRAD 6.1 (in the probabilistic mode) was used to perform uncertainties analyses and determine the sensitive parameters. The appropriate input file containing all physical, behavior and metabolic parameters was generated. This file included Haley & Aldrich hydrogeology values (Reference 6-17), K_ds developed by Argonne National Lab (Reference 6-15), and contaminated zone dimensions. DandD default values were used for metabolic and behavior inputs. RESRAD default values and distributions were used for physical parameters that could not be empirically tested or where no site-specific data existed.

A random seed of 1000 was used for uncertainty sampling. The Latin Hypercube Sample (LHS) method was used to generate samples of input values for the probabilistic analysis. Uncertainty correlations were established between density and total porosity, density and effective porosity, and total porosity and effective porosity with a correlation value specified as 0.99 for all three zones (i.e. contaminated, saturated and unsaturated).

The first 6 correlation tables (coefficients for 'peak of mean dose time dose' and 'peak all pathways dose') of the MCSUMMAR.REP computer file were extracted. Within these tables, the higher correlation coefficient (r^2 value) between the PRCC and PCC columns was selected. These values determine the sensitive nature of the parameter. Sensitive parameters were identified with correlation values greater than or equal to 0.25 or less than or equal to –0.25.

A default case of RESRAD was run in the probabilistic mode with only the sensitive parameters varying. An LHSBIN.DAT report was then generated and imported into an EXCEL spreadsheet to identify the means and 25th and 75th percentile values for the sensitive parameter distributions. Applicable values were then used as base deterministic inputs.

With the exception of C-14 and H-3, K_d values were developed for each SNEC related radionuclide by Argonne National Laboratories (ANL) from analysis of a group of samples collected at the SNEC site that included materials such as soils and fly ash, and building construction materials such as pulverized concrete, brick and block, etc. These values were then reviewed to determine their impact on dose. In all cases the lowest K_d developed for each radionuclide from each sample type produced the highest site dose. GPU Nuclear then selected the most conservative K_d value for each radionuclide to represent all material types at the site, thus site soils and re-fill materials may be placed in any location at the site without exceeding site dose limits.

For C-14 and H-3, ANL recommended a value near 1 as the appropriate K_d to be used at the site based on the type of volumetric materials present. Since these values were recommended and not empirically derived, a review of the impact on dose at K_d values within a range of possible K_d values near 1 was conducted by GPU Nuclear, Inc. The results indicated that a default value of 0.25 for H-3 and 1 for C-14 would provide the greater impact on dose and therefore these values were selected for use when the probabilistic analysis indicated K_d was a non-sensitive parameter. When sensitive, the approach previously described using the 25th or 75th percentile of the RESRAD K_d default parameter set was selected.

6.2.2.2 Deterministic Approach

Prior to running RESRAD in the deterministic mode, a new input file containing information from probabilistic mode runs, was created as follows:

- Suppression of the uncertainty analysis.
- The 75th percentile value was used to replace the base-deterministic input value for those sensitive parameters with sensitivity coefficients greater than or equal to 0.25.
- The 25th percentile value was used to replace the base-deterministic input value for those sensitive parameters with sensitivity coefficients less than or equal to -0.25.
- The mean value was used to replace the base-deterministic input value for those sensitive parameters not bounded by the 25th and 75th percentile values.
- Except when the coefficients of sensitivity for the distribution coefficients (K_d) are greater than or equal to 0.25, the minimum Argonne developed K_d was used.

To determine the applicable DCGL values for each radionuclide, RESRAD was run in the deterministic mode with the revised input file. The summary report provided the peak dose, year of occurrence and pathway breakdown for each peak dose. The 25 mrem/yr dose limit was divided by the peak dose to determine a DCGL representing exposure from all pathways. This process was used for each radionuclide, soil region and SNEC area of concern. For 4 mrem/yr drinking water dose goal, the above process was repeated with all pathways turned off except for the drinking water pathway. Files generated for drinking water dose analysis were appended with DW.

6.2.2.3 Radionuclide Selection

To date, eleven (11) radionuclides have been identified as being significant dose contributors for the SNEC site with Cs-137 being identified as the most predominant. Reference 6-13 provides the analysis for determining site-related radionuclides. These radionuclides have been loaded into both RESRAD and DandD software codes to determine applicable DCGLs for each respective model. Guidance from NUREG/CR-3474 and NUREG/CR-0130 was used to first develop a comprehensive list of radionuclides that could potentially be found in media at the SNEC site, during its operation and post shutdown periods. From this list various criteria was used to deselect radionuclides. Information on site-specific radionuclides was also determined using results of characterization surveys, waste stream analyses and historical site assessments that are appropriate for each medium. Once a list was developed a 4-step process was used to deselect radionuclides that are not applicable to SNEC.

Step 1 - SNEC has been shut down for almost 30 years. All radionuclides with half lives less than 3 years have been deselected since they have decayed 10 half lives.

Step 2 - Over 500 samples in various media have been analyzed as part of the characterization process. Radionuclide results below minimum detectable activity (MDA) levels were deselected.

Step 3 - Radionuclides in media that were < 1% of the total mix activity and < 10% of the dose limit were also deselected. Per Appendix E of NUREG-1727 (Reference 6.5), radionuclides contributing < 10% of the dose limit can be screened out.

Step 4 - Evaluate which sample media contain certain radionuclides.

From this analysis, seven (7) nuclides were deselected for meeting the <1% of the mix and <10% of the dose limit criteria. Together, all these nuclides contributed 3.45% of the total dose limit (25 mrem/yr). DCGLs will be adjusted in the final site design process to take into account this small fraction of the dose limit. As a result of the deselection process and most recent characterization data, Table 6-1 has been developed listing radionuclides present at the SNEC site. This table represents the list of radionuclides potentially found in volumetric media and on structural surface areas.

Table 6-1

SNEC Radionuclide List

H-3	Eu-152
C-14	Pu-238
Co-60	Pu-239
Ni-63	Pu-241
Sr-90	Am-241
Cs-137	

To date the results of sample analyses at the SNEC site have provided no valid confirmation for the presence of Np-237 above minimum detectable activity (MDA). Since this radionuclide is a daughter of Am-241 there is a minimal possibility of it showing up as a positively identified radionuclide. In the DandD and RESRAD codes the computer analysis takes into account the dose of the parent and all the daughters in the decay chain. Therefore, Np-237 is accounted for in the dose analyses for Am-241 and not included in the list of radionuclides of concern for the SNEC site. This is similar to how Cs-137 (parent) and its daughter, Ba-137m, are treated in the dose analysis. Laboratory analyses are reviewed to ensure radionuclides in Table 6-1 continue to be representative of the site. Should a radionuclide appear which is not on Table 6-1, a technical analysis will be performed to determine its validity.

6.2.2.4 Contaminated Zone Description

The soil guideline (DCGL) is defined as the radiological concentration in soil that is acceptable if the site is to be used without radiological restrictions. The SNEC surface model is based on a maximum sized 10,000 m^2-contaminated area, one meter thick with no cover material. The concentration of a radionuclide is considered to exceed background concentrations if it is greater than the mean background plus twice the standard deviation of the background measurements. Based on years of radiological surveys at the site the 10,000 m^2 contaminated area dimension was selected as a dose model default parameter and is considered bounding. The one-meter thickness was selected based on remediation work conducted in 1994 at the site

(Reference 6-14) and the average below grade groundwater level. For areas less than 10,000 m^2, area factors have been developed and listed in Chapter 5, Table 5-15. Soil at the SNEC site is defined as unconsolidated earth materials, including concrete and other structural debris that might be present.

The subsurface model calculates the dose from contaminants that may be in the saturated zone as a result of reuse of fill and debris materials. Subsurface materials for the Spray Pond and general site areas are very similar, consisting of approximately two meters of overburden and a greater thickness of underlying bedrock. The subsurface material in the SSGS consists of crushed, homogenized site construction debris that is covered with one meter of clean fill. Because of these differences, DCGLs were developed for only one material (homogenized debris) in the SSGS and for two materials (overburden and bedrock) in the Spray Pond and general areas.

6.2.2.5 Dose Calculation Times (years)

Radiation doses, health risks, soil guidelines and media concentrations are calculated over user-specified time intervals. The source is adjusted over time to account for radioactive decay and ingrowth, leaching, erosion and mixing. Although the regulatory recommendation is to use a 1000-year period, a 10,000-year period (more conservative assumption) was used to account for in-growth and decay of long lived transuranic nuclides that have a potential impact on the ground water pathway dose. RESRAD uses a one-dimensional groundwater model that accounts for different transport of parent and daughter radionuclides with different distribution coefficients (K_d).

6.2.2.6 Site Geology and Hydrology

Subsurface investigations have been conducted at the SNEC Facility since 1981. The purpose of the investigations was to define the geologic and hydrogeologic characteristics at the site. Several of the early investigations focused on monitor well installations at key plant locations. Recent investigations examine groundwater trends beyond the immediate plant area at more distant locations in order to characterize a broader aspect of the geologic conditions, groundwater flow and hydraulic conductivity.

There is reportedly approximately 7 to 18 feet of overburden material overlying bedrock (a fractured siltstone). The overburden materials generally consist of a fill overlying a natural boulder layer in a dense sandy, silty, clay matrix. Groundwater occurs in both the overburden/bedrock interface and bedrock.

Groundwater flow is toward the northwest from the Facility in both the overburden/bedrock interface and bedrock. The direction of flow is not effected by seasonal water level changes. The groundwater data indicates that the Raystown Branch of the Juniata River is a groundwater discharge feature. A subsurface discharge tunnel of a former coal fired generating station affects groundwater flow at the overburden/bedrock interface, acting as both a barrier and a drain. Groundwater flow in bedrock is controlled by northwest trending fractures.

Site-specific geometry (cross-section view) and hydrology data were used for input into the RESRAD code. This input data was based on studies conducted by a contracted geology firm (Reference 6-17) or default parameters determined by the RESRAD code, whichever was more conservative.

6.2.2.7 Chemical Form and K_ds

The chemical form of the SNEC residual radioactivity is bounded by the use of the default dose conversion factors (DCFs) in the RESRAD 6.1 code. These DCF values are based on chemical form information in Federal Guidance Report # 11 that give the individual the highest dose per unit intake.

Distribution coefficient (K_d) values are used in the RESRAD 6.1 code to predict the behavior of radionuclides in soil. Argonne National Laboratory has conducted tests and provided K_d measurements on SNEC soils and fill materials. Results of these tests are contained in Reference 6-15.

6.2.2.8 Water Transport Parameters

The well from which water is withdrawn for domestic use or irrigation is conservatively assumed to be located either in the center of the contamination zone (in the mass balance, MB, model) or at the downgradient edge of the contaminated zone (in the nondispersion, ND, model). For either location, radionuclides are assumed to enter the well as soon as they reach the water table. Usually, the MB model is used for smaller contaminated areas (e.g. 1000 m^2 or less) and the ND model is used for larger areas. For the SNEC surface model the ND input was used as the RESRAD input. For the SNEC subsurface model the MB input was used.

6.2.2.9 Volumetric Area Factors

Volumetric area factors were developed using RESRAD 6.1 and SNEC inputs for the surface modeling parameters (Reference 6-9). In the base-case surface model the contaminated fraction of plant, meat and milk products was assumed to equal one (1) using the resident farmer scenario. Default values of –1 were substituted for these three input parameters for areas less than 10,000 m^2. This was done so RESRAD could also scale smaller contaminated areas (2500 m^2, 400 m^2, 100 m^2, 25 m^2, and 1 m^2). The three default parameter values (-1) appropriately size the contaminated fractions of plants, meat and milk obtained from the site when smaller and smaller area sizes are input into the RESRAD computer code. Volumetric area factors for SNEC are listed in Table 5-15.

6.3 DCGL SUMMARY & DOSE ASSESSMENT

The DandD and RESRAD codes were run to determine compliance with 10CFR20.1402. DCGL results are listed in Table 6-2. Detailed information from dose modeling computer runs is contained on electronic media (CD-ROM) that has been submitted to the NRC (Reference 6-12).

Table 6-2

SNEC Facility DCGL Values [a]

Radionuclide	25 mrem/y Limit Surface Area (dpm/100cm^2)	25 mrem/y Limit (All Pathways) Open Land Areas (Surface & Subsurface) (pCi/g)	4 mrem/y Goal (Drinking Water) Open Land Areas [b] (Surface & Subsurface) (pCi/g)
Am-241	2.7E+01	9.9	2.3
C-14	3.7E+06	2	5 4
Co-60	7.1E+03	3.5	67
Cs-137	2.8E+04	6 6	397
Eu-152	1.3E+04	10 1	1440
H-3	1.2E+08	132	31.1
Ni-63	1.8E+06	747	1.9E+04
Pu-238	3.0E+01	1 8	0.41
Pu-239	2 8E+01	1.6	0.37
Pu-241	8.8E+02	86	19.8
Sr-90	8.7E+03	1.2	0 61

Footnotes:

a) While drinking water DCGLs will be used by SNEC to meet the drinking water 4 mrem/yr goal, only the DCGL values that constitute the 25 mrem/yr regulatory limit will be controlled under this LTP and the NRC's approving license amendment.

b) Listed values are from the subsurface model. These values are most conservative between the two models (i e. surface & subsurface).

The dose assessment using these values indicates that the dose will be below 25 mrem/year TEDE release limit and the 4 mrem/year groundwater dose goal. Therefore, there is a high degree of confidence that additional refinement of the source terms and modeling assumptions are unnecessary and the site can be released for unrestricted use.

6.4 ALARA ANALYSIS

Subpart E of 10 CFR 20 contains specific requirements for a demonstration that residual radioactivity has been reduced to a level that is ALARA (10 CFR 20.1402). Both Section 7 and Appendix D of NUREG-1727 (Reference 6-5) describe a method for determining whether levels of residual radioactivity are ALARA. The method presented is used to estimate when a remedial action provides a net benefit of dose reduction when compared to the cost of performing that action. It also provides example calculations of various remedial actions, some of which can be applied at the SNEC Facility to reach the DCGLs for surface and soil contamination. As stated in Chapter 4 of this plan, comprehensive remediation will be performed at the SNEC Facility Site. NUREG-1727 states that the ALARA requirement is met by performing the remediated action where appropriate. The guide further states that if a remedial action is performed there is no need to evaluate whether it is necessary to meet the ALARA requirement.

The SNEC Facility remediation goal for all structure surfaces is to reduce contamination levels to or below the screening levels (Reference Table 6-2) established by Reference 6-8 and the DandD code. These values were determined using highly conservative parameters to ensure that any residual radioactivity remaining at the site would not result in any significant impact to public health and safety. Therefore, further demonstration need not be performed.

In the case of volumetric contamination, the SNEC Facility remediation plans are to remove residual contamination to below those levels established in Table 6-2. As stated in NUREG-1727, for sites that select the unrestricted release criteria, a mathematical ALARA analysis for removing residual radioactivity from soil at these sites is not necessary, largely because of the high costs of waste disposal. Thus in the case of the SNEC Facility, no further ALARA evaluation is required after the removal of soil contamination to reach the DCGLs.

6.5 REFERENCES

6-1 Code of Federal regulations, Title 10, Part 1402, "Radiological Criteria for Unrestricted Use."

6-2 Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors," January 1999.

6-3 RESRAD, Version 6.1, United States Department of Energy and Argonne National Laboratory, April 1998.

6-4 DandD, Version 3.1 for Windows, United States Nuclear Regulatory Commission and Sandia National Laboratory, July 1996.

6-5 NUREG-1727, "NMSS Decommissioning Standard Review Plan," September 2000.

6-6 NUREG/CR-5512 Volume 1, "Residual Radioactive Contamination from Decommissioning," October 1992.

6-7 NUREG-1549, "Decision Methods for Dose Assessment to Comply with Radiological Criteria for License Termination," July 1998.

6-8 Federal Register: Wednesday, November 18, 1998 (Volume 63, Number 222) Notices, pg 64133, Table 1 – "Acceptable License Termination Screening Values of Common Radionuclides for Building Surface Contamination."

6-9 SNEC Calculation 6900-02-008, Revision 1, "SNEC Facility DCGL Values for Volumetric Contamination," September 2002.

6-10 SNEC Calculation E900-01-005, "Determination of Surface Area Factors", April 5, 2002.

6-11 Rogers & Associates Engineering Branch, URS Corporation, "Calculation of the Sub-Surface DCGLs for the Saxton Nuclear Experimental Corporation Site", RAE-42613-003-4R1, September 20, 2002.

6-12 GPU Nuclear Letter E910-02-039 dated August 20, 2002. "Submittal of Information to NRC in July 31, 2002 Meeting."

6-13 SNEC Calculation E900-01-030 Revision 3, "SNEC Radionuclide List", July 29, 2002.

6-14 GPU Nuclear Report, "1994 Saxton Soil Remediation Project Report", May 1995.

6-15 Argonne National Laboratory Report, "K_d Study of Site Soils and Construction Debris from the SNEC Decommissioning Project", February 2002.

6-16 CoPhysics Corporation Report, "Embedded Pipe Radiation Survey Report, GPU Nuclear Corp. Saxton Experimental Nuclear Co., Saxton, Pa.", January 2002.

6-17 Haley & Aldrich Report, "Summary for RESRAD Parameters Inputs (Revision 1)", October 18, 2001.

6-18 GPU Nuclear Report, "Saxton Experimental Corporation Facility Site Characterization Report", 1996.

Figure 6-1

(Resident Farmer Scenario)

Dose Modeling Logic Chart

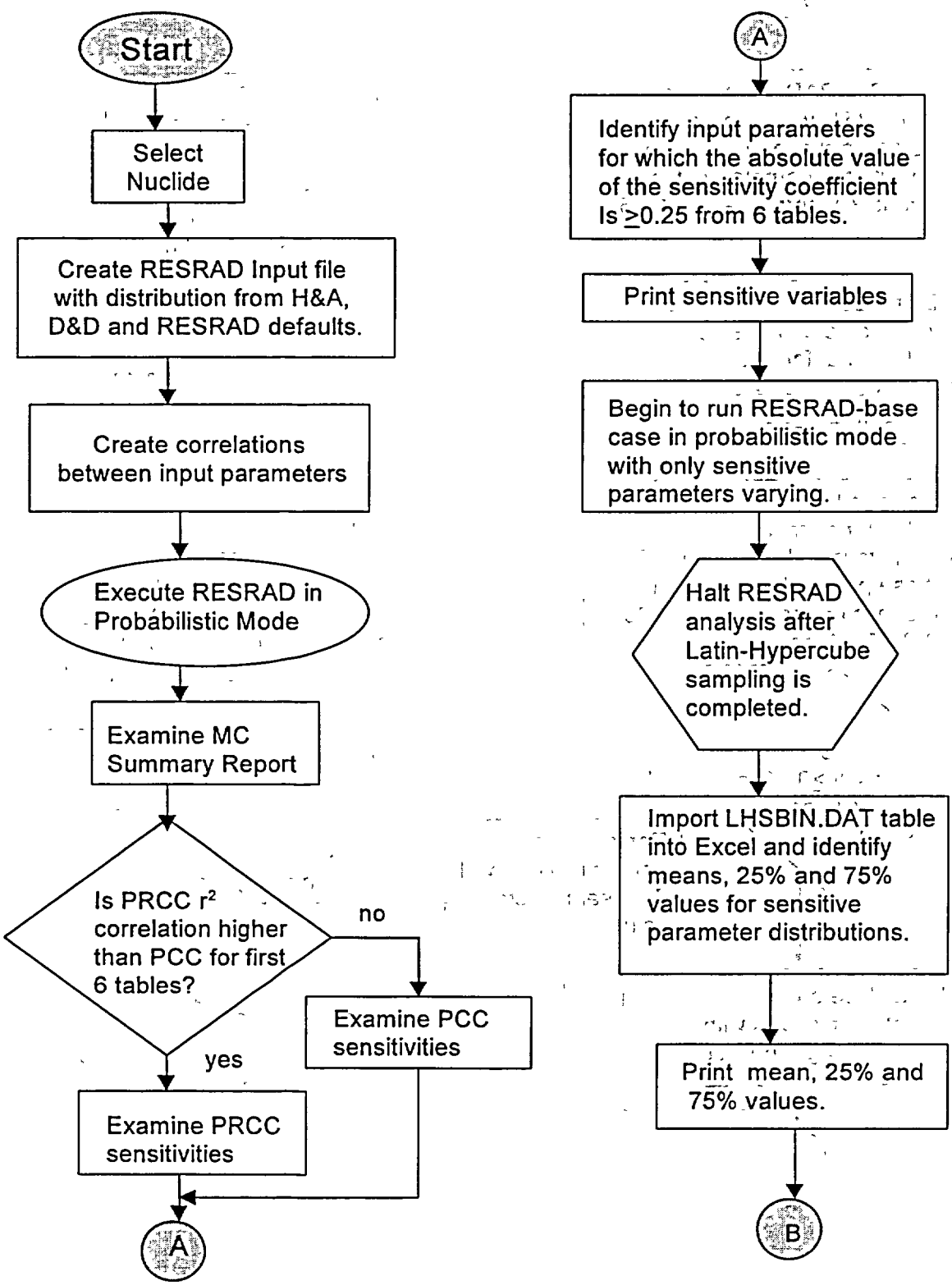
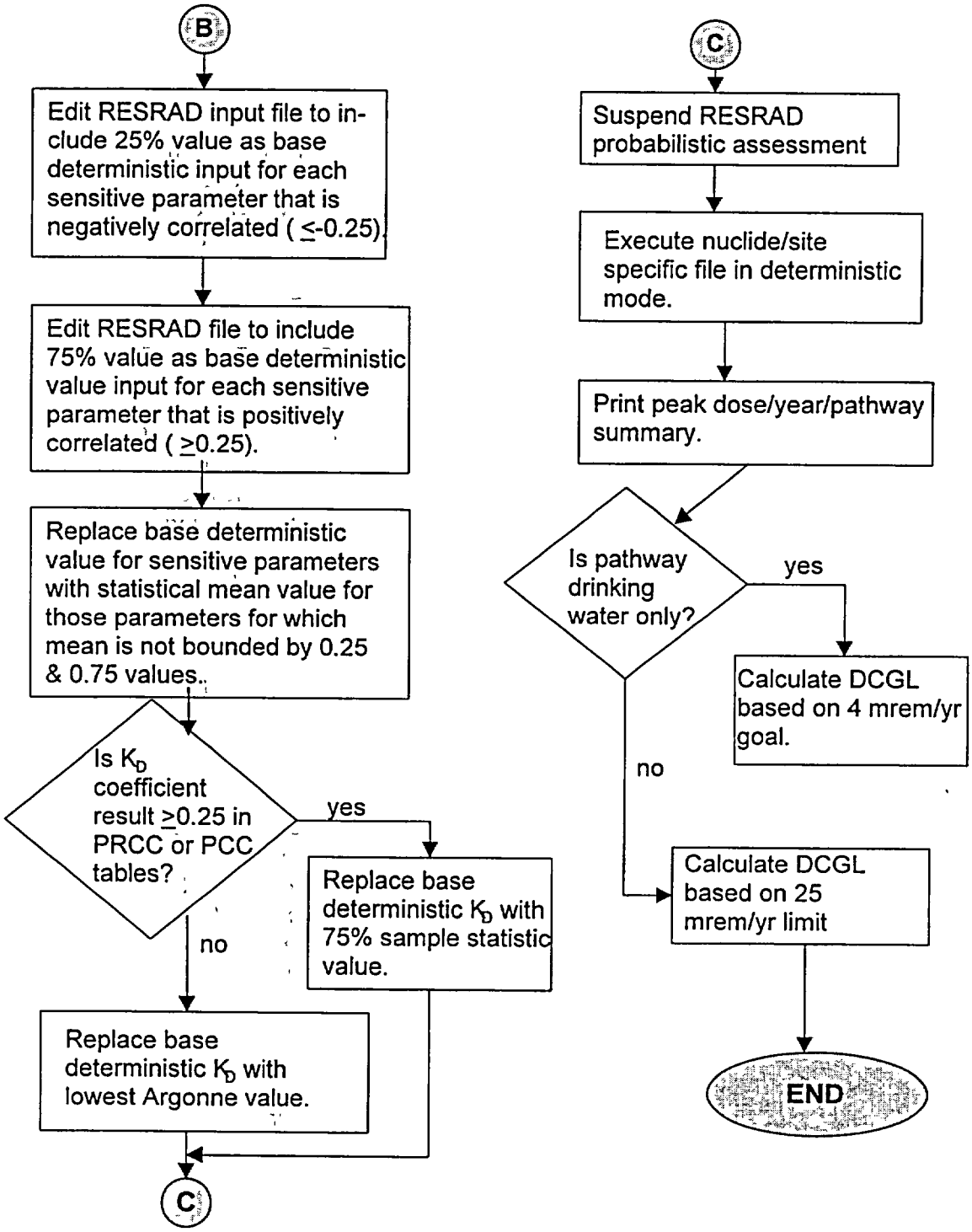


Figure 6.1 (cont.)

Dose Modeling Logic Chart (cont.)



APPENDIX 6.1

DandD DCGL CALCULATIONS

FOR

BUILDING OCCUPANCY SURFACE AREA MODEL

Program : DandD Version 1.0 Build 1.00.02
Session : Single Run Am-241
Description :
Single Run Am-241

Executed : 09/28/99 at 15:57:59

NRC Report

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	dpm/100cm^2
241Am	1.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
241Am	1.0000E+000
237Np	0.0000E+000
233Pa	0.0000E+000
233U	0.0000E+000
229Th	0.0000E+000
225Ra	0.0000E+000
225Ac	0.0000E+000
221Fr	0.0000E+000
217At	0.0000E+000
213Bi	0.0000E+000
213Po	0.0000E+000
209Tl	0.0000E+000
209Pb	0.0000E+000

Basic Parameters

Name	Value	Units	Default

'End Time'	25560.0000	days	365.2500

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 69.98 year(s).

The peak dose of 9.34E-001 TEDE (mrem) occurred 1.00 year(s) after license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	3.85E-005	0.00
Inhalation	9.29E-001	99.54
Ingestion	4.26E-003	0.46
Total	9.34E-001	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
241Am	9.34E-001	100.00
237Np	1.84E-007	0.00
233Pa	3.91E-011	0.00
233U	4.96E-014	0.00
229Th	1.71E-017	0.00
225Ra	5.02E-020	0.00
225Ac	5.64E-020	0.00
221Fr	1.03E-022	0.00
217At	1.05E-024	0.00

213Bi	5.49E-022	0.00
213Po	0.00E+000	0.00
209Tl	1.42E-022	0.00
209Pb	2.15E-024	0.00
Total	9.34E-001	100.00

$$\frac{25 \text{ mrem}}{0.934 \frac{\text{mrem}}{\text{dpm}/100\text{cm}^2}} = 27 \frac{\text{dpm}}{100\text{cm}^2}$$

Program : DandD Version 1.0 Build 1.00.02
Session : Single Run C-14
Description :
Single Run C-14

Executed : 09/28/99 at 15:22:50

NRC Report

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	dpm/100cm^2
14C	1.0000

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
14C	1.0000E+000

Basic Parameters

Name	Value	Units	Default
'End Time'	25560.0000	days	365.2500

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 69.98 year(s).

The peak dose of 6.83E-006 TEDE (mrem) occurred 1.00 year(s) after license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	2.26E-008	0.33
Inhalation	4.37E-006	63.96
Ingestion	2.44E-006	35.71
Total	6.83E-006	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
14C	6.83E-006	100.00
Total	6.83E-006	100.00

$$\frac{25 \text{ mrem}}{6.83E\text{-}6 \frac{\text{mrem}}{\text{dpm}/100 \text{ cm}^2}} = 3.7E6 \text{ dpm}/100 \text{ cm}^2$$

Program : DandD Version 1.0 Build 1.00.02
Session : Single Run Co-60
Description :
Single Run Co-60

Executed : 09/28/99 at 15:27:54

NRC Report

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	dpm/100cm^2
60Co	1.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
60Co	1.0000E+000

Basic Parameters

Name	Value	Units	Default
'End Time'	25560.0000	days	365.2500

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 69.98 year(s).

The peak dose of 3.55E-003 TEDE (mrem) occurred 1.00 year(s) after license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	3.09E-003	87.07
Inhalation	4.29E-004	12.10
Ingestion	2.95E-005	0.83
Total	3.55E-003	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
60Co	3.55E-003	100.00
Total	3.55E-003	100.00

$$\frac{25 \text{ mrem}}{3.55E\text{-}3 \ \frac{\text{mrem}}{\text{dpm}/100\text{cm}^2}} = 7.0E3 \ \frac{\text{dpm}}{100 \text{ cm}^2}$$

Per Table 1 of Fed. Reg Vol 63 No 222 Nov 8 1998

Co^{60} Screening Value = $7.1E3 \ \frac{\text{dpm}}{100 \text{ cm}^2}$

Program : DandD Version 1.0 Build 1.00.02
Session : Single Run Cs-137
Description :
Single Run Cs-137

Executed : 09/28/99 at 15:40:59

NRC Report

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	dpm/100cm^2
137Cs	1.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
137Cs	1.0000E+000
137mBa	0.0000E+000

Basic Parameters

Name	Value	Units	Default
'End Time'	25560.0000	days	365.2500

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now and ran for 69.98 year(s).

The peak dose of 8.93E-004 TEDE (mrem) occurred 1.00 year(s) after license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	7.69E-004	86.13
Inhalation	6.61E-005	7.41
Ingestion	5.78E-005	6.47
Total	8.93E-004	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
137Cs	1.24E-004	13.92
137mBa	7.69E-004	86.08
Total	8.93E-004	100.00

$$\frac{25\ \text{mrem}}{8.93E\text{-}4\ \frac{\text{mrem}}{\text{dpm}/100\ \text{cm}^2}} = 2.8E4\ \frac{\text{dpm}}{100\ \text{cm}^2}$$

Program : DandD Version 1.0 Build 1.00.02
Session : Single Run Eu-152
Description :
Single Run Eu-152

Executed : 09/28/99 at 15:42:47

NRC Report

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	dpm/100cm^2
152Eu	1.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
152Eu	1.0000E+000
152Gd	0.0000E+000

Basic Parameters

Name	Value	Units	Default
'End Time'	25560.0000	days	365.2500

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 69.98 year(s).

The peak dose of 1.97E-003 TEDE (mrem) occurred 1.00 year(s) after license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	1.51E-003	76.70
Inhalation	4.51E-004	22.93
Ingestion	7.38E-006	0.38
Total	1.97E-003	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
152Eu	1.97E-003	100.00
152Gd	7.07E-018	0.00
Total	1.97E-003	100.00

$$\frac{25\ \text{mrem}}{1.97E\text{-}3\ \frac{\text{mrem}}{\text{dpm}/100\text{cm}^2}} = 1.3E4\ \frac{\text{dpm}}{100\ \text{cm}^2}$$

Program : DandD Version 1.0 Build 1.00.02
Session : Single Run H-3
Description :
Single Run H-3

Executed : 09/28/99 at 15:19:52

NRC Report

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	dpm/100cm^2
3H	1.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
3H	1.0000E+000

Basic Parameters

Name	Value	Units	Default
'End Time'	25560.0000	days	365.2500

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 69.98 year(s).

The peak dose of 2.03E-007 TEDE (mrem) occurred 1.00 year(s) after license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	0.00E+000	0.00
Inhalation	1.30E-007	64.17
Ingestion	7.28E-008	35.83
Total	2.03E-007	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
3H	2.03E-007	100.00
Total	2.03E-007	100.00

$$\frac{25 \text{ mrem}}{2.03E\text{-}7 \frac{\text{mrem}}{\text{dpm}/100\text{cm}^2}} = 1.2E8 \text{ dpm}/100\text{cm}^2$$

Program : DandD Version 1.0 Build 1.00.02
Session : Single Run Ni-63
Description :
Single Run Ni-63

Executed : 09/28/99 at 15:31:04

NRC Report

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	dpm/100cm^2
63Ni	1.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
63Ni	1.0000E+000

Basic Parameters

Name	Value	Units	Default
'End Time'	25560.0000	days	365.2500

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 69.98 year(s).

The peak dose of 1.38E-005 TEDE (mrem) occurred 1.00 year(s) after license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	0.00E+000	0.00
Inhalation	1.31E-005	95.13
Ingestion	6.73E-007	4.87
Total	1.38E-005	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
63Ni	1.38E-005	100.00
Total	1.38E-005	100.00

$$\frac{25 \text{ mrem}}{1.38E\text{-}5 \frac{\text{mrem}}{\text{dpm}/100\text{cm}^2}} = 1.8E6 \text{ dpm}/100\text{cm}^2$$

Program : DandD Version 1.0 Build 1.00.02
Session : Single Run Pu-238
Description :
Single Run Pu-238

Executed : 09/28/99 at 16:03:46

NRC Report

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	dpm/100cm^2
238Pu	1.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
238Pu	1.0000E+000
234U	0.0000E+000
230Th	0.0000E+000
226Ra	0.0000E+000
222Rn	0.0000E+000
218Po	0.0000E+000
214Pb	0.0000E+000
218At	0.0000E+000
214Bi	0.0000E+000
214Po	0.0000E+000
210Pb	0.0000E+000
210Bi	0.0000E+000
210Po	0.0000E+000

Basic Parameters

Name	Value	Units	Default

'End Time'	25560.0000	days	365.2500

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 69.98 year(s).

The peak dose of 8.22E-001 TEDE (mrem) occurred 1.00 year(s) after license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	1.17E-006	0.00
Inhalation	8.18E-001	99.55
Ingestion	3.73E-003	0.45
Total	8.22E-001	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
238Pu	8.22E-001	100.00
234U	3.93E-007	0.00
230Th	2.90E-012	0.00
226Ra	9.00E-018	0.00
222Rn	2.40E-022	0.00
218Po	5.41E-024	0.00
214Pb	1.55E-019	0.00
218At	0.00E+000	0.00
214Bi	8.66E-019	0.00

214Po	4.95E-023	0.00
210Pb	8.85E-020	0.00
210Bi	9.56E-022	0.00
210Po	9.54E-021	0.00
Total	8.22E-001	100.00

$$\frac{25\ \text{mrem}}{0.822\ \frac{\text{mrem}}{\text{dpm}/100\ \text{cm}^2}} = 3.0\ E1\ \text{dpm}/100\ \text{cm}^2$$

Program : DandD Version 1.0 Build 1.00.02
Session : Single Run Pu-239
Description :
Single Run Pu-239

Executed : 09/28/99 at 15:52:17

NRC Report

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	dpm/100cm^2
239Pu	1.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
239Pu	1.0000E+000
235U	0.0000E+000
231Th	0.0000E+000
231Pa	0.0000E+000
227Ac	0.0000E+000
223Fr	0.0000E+000
227Th	0.0000E+000
223Ra	0.0000E+000
219Rn	0.0000E+000
215Po	0.0000E+000
211Pb	0.0000E+000
211Bi	0.0000E+000
211Po	0.0000E+000
207Tl	0.0000E+000

Basic Parameters

Name	Value	Units	Default
'End Time'	25560.0000	days	365.2500

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 69.98 year(s).

The peak dose of 9.03E-001 TEDE (mrem) occurred 1.00 year(s) after license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	5.15E-007	0.00
Inhalation	8.99E-001	99.54
Ingestion	4.14E-003	0.46
Total	9.03E-001	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
239Pu	9.03E-001	100.00
235U	1.27E-010	0.00
231Th	1.44E-014	0.00
231Pa	9.24E-015	0.00
227Ac	3.78E-016	0.00
223Fr	3.80E-023	0.00
227Th	6.88E-019	0.00
223Ra	2.98E-019	0.00

219Rn	1.32E-021	0.00
215Po	4.20E-024	0.00
211Pb	1.54E-021	0.00
211Bi	1.10E-021	0.00
211Po	5.12E-025	0.00
207Tl	9.02E-023	0.00
Total	9.03E-001	100.00

$$\frac{25 \text{ mrem}}{\frac{0.903 \text{ mrem}}{\text{dpm}/100 \text{ cm}^2}} = 2.8E1 \text{ dpm}/100 \text{ cm}^2$$

Program : DandD Version 1.0 Build 1.00.02
Session : Single Run Pu-241
Description :
Single Run Pu-241

Executed : 09/28/99 at 15:54:54

NRC Report

Occupancy Input Section

Execution Options
==========
History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	dpm/100cm^2
241Pu	1.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
241Pu	1.0000E+000
241Am	0.0000E+000
237U	0.0000E+000
237Np	0.0000E+000
233Pa	0.0000E+000
233U	0.0000E+000
229Th	0.0000E+000
225Ra	0.0000E+000
225Ac	0.0000E+000
221Fr	0.0000E+000
217At	0.0000E+000
213Bi	0.0000E+000
213Po	0.0000E+000
209Tl	0.0000E+000
209Pb	0.0000E+000

Basic Parameters

Name	Value	Units	Default
'End Time'	25560.0000	days	365.2500

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 69.98 year(s).

The peak dose of 2.84E-002 TEDE (mrem) occurred 57.00 year(s) after license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	1.12E-006	0.00
Inhalation	2.83E-002	99.54
Ingestion	1.30E-004	0.46
Total	2.84E-002	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
241Pu	1.14E-003	4.03
241Am	2.73E-002	95.97
237U	3.27E-010	0.00
237Np	4.39E-007	0.00
233Pa	1.15E-010	0.00
233U	1.07E-011	0.00
229Th	2.52E-013	0.00

225Ra	9.34E-016	0.00
225Ac	1.27E-015	0.00
221Fr	2.32E-018	0.00
217At	2.36E-020	0.00
213Bi	1.23E-017	0.00
213Po	0.00E+000	0.00
209Tl	3.20E-018	0.00
209Pb	4.84E-020	0.00
Total	2.84E-002	100.00

$$\frac{25 \text{ mrem}}{\frac{2.84E\text{-}2 \text{ mrem}}{\text{dpm}/100\text{ cm}^2}} = 8.8E2 \text{ dpm}/100\text{ cm}^2$$

Program : DandD Version 1.0 Build 1.00.02
Session : Single Run Sr-90
Description :
Single Run Sr-90

Executed : 09/28/99 at 15:32:48

NRC Report

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	dpm/100cm^2
90Sr	1.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
90Sr	1.0000E+000
90Y	0.0000E+000

Basic Parameters

Name	Value	Units	Default
'End Time'	25560.0000	days	365.2500

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 69.98 year(s).

The peak dose of 2.89E-003 TEDE (mrem) occurred 1.00 year(s) after license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	7.70E-006	0.27
Inhalation	2.71E-003	93.61
Ingestion	1.77E-004	6.12
Total	2.89E-003	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
90Sr	2.85E-003	98.72
90Y	3.69E-005	1.28
Total	2.89E-003	100.00

$$\frac{25 \text{ mrem}}{\frac{2.89E\text{-}3 \text{ mrem}}{\text{dpm}/100\text{ cm}^2}} = 8.7E3 \text{ dpm}/100\text{ cm}^2$$

7.0 UPDATE OF THE SITE-SPECIFIC DECOMMISSIONING COSTS

NRC's request for additional information dated November 8, 2000 requested additional information with respect to the site specific decommissioning cost information provided in Revision 0 of the SNEC License Termination Plan. GPU Nuclear's response to this request was reviewed and accepted by the NRC in conjunction with their review of the merger between FirstEnergy Corp. and GPU, Inc. The adequacy of decommissioning funding assurance for the SNEC Facility was documented by the Nuclear Regulatory Commission in the "Order Approving Application Regarding Proposed Merger of GPU, Inc. and FirstEnergy Corp. – Saxton Nuclear Experimental Facility (TAC NO. MB0215)" dated March 7, 2001.

Since that time the cost and schedule associated with the current Containment Vessel (CV) concrete removal project has exceeded what was assumed in this response. This has resulted in an overall $7 million increase in the remaining project cost beyond the $19.8 million estimate provided in GPU Nuclear letter E910-01-002 dated February 14, 2001, "Partial Response to Request for Additional Information, RE: License Termination Plan, (TAC NO. MA8076) dated November 8, 2000). Thus the current overall project cost estimate is approximately $63 million. As of July 31, 2002 approximately $50 Million has been spent on the SNEC Decommissioning Project. Thus the remaining cost to complete the project is approximately $13 Million.

GPU Nuclear Letter E910-01-004, dated February 19, 2001, "Parent Guarantee for Decommissioning Funding" committed the SNEC Owners to carry out the required activities or setup a trust fund in favor of the NRC in the event GPU Nuclear failed to perform the required decommissioning activities. The amount of this guarantee is $20 million, which exceeds the remaining cost estimate of $13 million. Thus adequate funding exists to complete the project.

8.0 SUPPLEMENT TO THE ENVIRONMENTAL REPORT

The SNEC Facility Environmental Report was originally submitted in April 1996 to support SNEC Facility decommissioning. Three supplements to the report, in the form of responses to NRC requests for additional information, were provided on July 18, 1996, March 3, 1998 and March 31, 1998. In support of the SNEC Facility License Termination Plan, these documents were reviewed and revisions were made to reflect updates to information. Revision 1 of the SNEC Decommissioning Environmental Report was submitted under separate cover as GPU Letter 1920-00-20025, dated February 2, 2000. Revision 2 of the Updated SNEC Environmental Report is being submitted under a separate cover letter as GPU Nuclear Inc. E910-02-046.

The most significant changes in this document are:

1. Updated information for the SNEC site to reflect its current configuration. Selected plant structures such as the septic system, weir pipe and other underground piping have been removed. A description of the SSGS Intake Tunnel and various Penelec buildings has been added to reflect the impact of decommissioning activities.

2. Updated the estimated occupational exposure from approximately 37 person-rem to 37.84 person-rem based on changes in work scope for CV concrete removal and experience gained on the SNEC Facility Decommissioning Project. This occupational exposure is still well within the GEIS (NUREG 0586) estimate of 344 person-rem,

3. Updated geology, hydrology, endangered species and groundwater flow studies conducted in 2001. In addition population and population trends have been updated to reflect current year 2000 census data.

4. Updated current status of plant dismantlement activities, sources of waste generation (radioactive, hazardous and asbestos), and current decommissioning schedule.

5. Updated accident and exposure analyses for total public dose from radwaste shipping and re-estimated land area occupied by disposal of radwaste.

6. Updated historical and future radwaste volumes. Analyzed these volumes in comparison to NUREG-0586 criteria for a reference test reactor.

7. Included table summarizing SNEC vs NUREG-0586 comparative analyses which includes updated and previously submitted information.

These changes are still within the bounds of the GEIS and so it can be concluded that there are no significant environmental changes at the SNEC Facility associated with License Termination.