Fort St. Vrain Nuclear Station Decommissioning Project

Final Survey Plan for Site Release

Revision 1

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PREFACE

This document describes the methods used by the Public Service Company of Colorado (PSC) to demonstrate that radiation and radioactive contamination levels at the Fort St. Vrain Nuclear Station (FSV) have been reduced to levels below criteria established for unrestricted use. This plan has been developed in accordance with the Fort St. Vrain Decommissioning Plan (DP), FSV Decommissioning Technical Specifications (DTS), Draft NUREG/CR-5849, "Manual for Conducting Radiological Surveys in Support of License Termination," and 10 CFR 50. It supplements and updates the description of the proposed final radiation survey presented in the Fort St. Vrain Decommissioning Plan.

This plan has been developed as administrative guidance. It is intended to provide the basis for the implementing procedures governing the conduct of the final survey. Revisions to this plan may be implemented without prior NRC approval provided the changes do not:

- Involve an unreviewed safety question as defined in 10 CFR 50.59 and do not require a change in the Decommissioning Technical Specifications,
- Reduce the required survey frequency for the classification of the survey unit,
- Increase the action levels for conducting investigation and followup surveys, or
- Affect the statistical treatment of survey data in a manner which could reduce the confidence that the site meets the criteria for unrestricted use.

This plan includes a description of the technical considerations and methods to be used for design and implementation of the final survey. The methods described are derived from regulatory guidance, specifically Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors," Draft NUREG/CR-5849, "Manual for Conducting Radiological Surveys in Support of License Termination," NUREG/CR-5512, "Residual Radioactive Contamination From Decommissioning," and from recent U.S. reactor facility decommissioning experience (U.S. Army Materials Technology Laboratory, Shoreham Nuclear Power Station, Pathfinder, Saxton, Shippingport, and UC Berkeley), taking into account conditions that are specific to the Fort St. Vrain facility.

> Note: For the definitions of terminology used in the Final Survey Plan, please refer to Section 8.0, "Glossary."

1.0 HISTORICAL BACKGROUND

Public Service Company of Colorado (PSC) operated a 330 Mw(e) High Temperature Gas Cooled Reactor (HTGR) from July, 1979 until August, 1989. The plant, designated as the Fort St. Vrain Nuclear Station (FSV), is located approximately 35 miles north of Denver and three and one-half miles northwest of the town of Platteville in Weld County, Colorado.

Construction of FSV was authorized by the U.S. Nuclear Regulatory Commission (NRC) by issuance of a provisional construction permit on September 17, 1968. Construction was completed in December, 1973 and a facility operating license was granted on December 21, 1973. Initial fuel loading commenced on December 26, 1973 and initial criticality was achieved January 31, 1974. After a prolonged period of startup testing, low-power operation and plant modifications, the plant was committed for commercial operation on July 1, 1979. Full power was achieved November 6, 1981.

FSV was a load-following central station power plant which used a HTGR to produce steam for the generation of electric power. The reactor design utilized the same fundamental principles that formed the basis of the 40 Mw(e) prototype HTGR at Peach Bottom, Pennsylvania. That prototype began supplying power to the Philadelphia Electric Company system in March of 1967; began commercial operation on June 1, 1967; and ceased operation on October 31, 1974.

In the nuclear steam supply system for FSV, heat was produced by fission in the HTGR utilizing a uranium-thorium fuel cycle. Graphite was used for the moderator, core structure, and reflector. High temperature helium was used as the primary coolant to produce superheated and reheated steam at a temperature of 1,000 °F to match conventional thermal station conditions. The entire nuclear steam supply system, including the reactor core, graphite moderator and reflector, steam generators and helium circulators, was contained within a Prestressed Concrete Reactor Vessel (PCRV).

The active core was composed of 1,482 hexagonal graphite fuel elements stacked in 247 vertical columns. The fuel was in the form of particles made of a mixture of the carbides of thorium and uranium which was coated with highly retentive coatings of pyrolytic carbon and silicon carbide. The fuel particles and a carbonaceous matrix form bonded fuel rods which were located within the fuel holes. The combination of a relatively large fuel particle (200-500 microns) and a four-layer coating provided the particles excellent fission product retention characteristics.

During the timeframe of operation, FSV operated for approximately 890 effective full-power days. FSV was shutdown on August 18, 1989. On August 29, 1989, the PSC Board of Directors reviewed and confirmed the Executive Management decision that FSV would not be restarted, and that PSC would pursue decommissioning of FSV. The decision to permanently shutdown and decommission FSV was based on related technical and financial considerations.

Problems were identified with the contrc· rod drive assemblies and the steam generator steam ring headers that presented significant technical obstacles which could be overcome, but at significant cost in dollars and time to PSC. In addition, due to the uniqueness of the one-of-a-kind HTGR fuel cycle, the cost to purchase new fuel was prohibitive. This, in conjunction with low plant availability and correspondingly high operating costs, made continued operation of FSV imprudent.

A preliminary decommissioning plan based on the SAFSTOR alternative was filed with the NRC in June, 1989. Subsequent evaluation by PSC resulted in the decision to select the DECON alternative, and the Proposed Decommissioning Plan (PDP) was filed with the NRC in November, 1990. PSC's objective is the dismantlement and decommissioning of FSV to release all site areas for unrestricted use. To accomplish this, a portion of the PCRV structure and the radioactive balance-of-plant equipment which exceed the limits for unrestricted use will be decontaminated or removed as described in the Fort St. Vrain Decommissioning Plan.

In May, 1991 a 10 CFR 50 Possession Only License was granted by the NRC.

The DECON alternative has been chosen for implementation of decommissioning activities at FSV. This involves the decontamination and dismantlement as necessary of plant systems and areas to allow release of the facility for unrestricted use. In the Generic Environmental Impact Statement on Decommissioning Nuclear Facilities (GEIS) (Ref. 1), the NRC concludes that DECON is advantageous if the site is required for other purposes or if the site is extremely valuable. DECON is advantageous to FSV because of its potential conversion to a gas-fired generating station and because the FSV switchyard is valuable for power distribution within the PSC system. The current regulatory process for DECON is also well understood and the choice of DECON avoids the need to plan for contingencies to accommodate new requirements which may be introduced in future regulatory changes. Additionally, the operating staff is available to assist with surveillance and maintenance of the facility during decommissioning activities.

2.0 SITE INFORMATION

2.1 Site Description

The Fort St. Vrain Nuclear Station is located approximately 35 miles north of Denver and 3.5 miles northwest of the town of Platteville in Weld County, Colorado. The site is located in an agricultural area with gently rolling hills. Grade elevation at the plant is 4,790 feet above sea level.

The 2,798 acre site owned by PSC is identified as the Owner-Controlled Area. Farming has been continued on Owner-Controlled area of the site, but there are no farming operations or permanent residences located within the Restricted Area. The Restricted Area is surrounded by a Security fence, and access is controlled for purposes of protection of individuals from exposure to radiation in accordance with the Decommissioning Plan (DP), Section 8, Decommissioning Access Control Plan (Ref. 2).

The station is located approximately two miles south of the confluence of the South Platte River and the St. Vrain Creek. Neither of these two streams are considered navigable. Cooling for the plant is provided by mechanical draft cooling towers. Make-up to the cooling towers is obtained from the two streams, and is supplemented by shallow well water. Nineteen shallow wells are located on the site. The licensee also owns surface water rights in four irrigation ditches which traverse portions of the site.

A considerable amount of data regarding the levels of radiation and radioactive material in the environs has been collected and reported in the annual Radiological Environmental Monitoring Program (REMP) Reports (Ref. 3 & 4). The data indicates that there is little probability that any significant amount of radioactive liquids could have entered the groundwater during power operations. Any effluent which was discharged from the radioactive liquid waste system was diluted by the cooling tower blowdown line prior to release to surrounding surface waters. There were no spills or accidents during operation of FSV which had the potential of contaminating the site. The REMP will be continued throughout the decommissioning to ensure that any contamination of the site environs which could occur as a result of decommissioning activities is detected.

Liquid effluent is released from the FSV site via the Goosequill Ditch. From the concrete-lined Goosequill Ditch, liquid effluent flows into the Jay Thomas Ditch, where additional dilution may occur, and then on to a 25 acre farm pond that contains approximately 32 million gallons of water. The total distance from site discharge to farm pond discharge is approximately 8,700 feet. An alternate flowpath, via a slough, to the St. Vrain Creek and then on to the South Platte River is utilized when discharge via the Goosequill Ditch is not possible.

Within the area surrounding the site, there is no evidence of recent earth movement along any known faults. Shales of the Pierre Formation form the bedrock at the plant location. Subsoils at the site generally consist of a thin layer of sands, capable of supporting light

structures; underlain by medium dense sands becoming gravelly with depth that can support moderate loads; and thence by hard to very hard claystone bedrock, capable of supporting heavy loads, found at a depth of 44 to 54 feet.

The major structures within the Restricted Area include the Reactor Building which contains the PCRV, Turbine Building, Radwaste Compactor Building, New Fuel Storage Building, Technical Support Building which contains the Radiochemistry Laboratory, Mechanical Draft Cooling Towers, Warehouse and Construction Workshops, Evaporation Ponds, and the Electrical Switchyard. The ground surface covering within the Restricted Area is composed primarily of gravel and vegetation, with smaller portions devoted to concrete or asphalt roadways and laydown areas.

The NRC issued PSC an Environmental Assessment of No Significant Impact for the construction and operation of an Independent Spent Fuel Storage Installation (ISFSI). The ISFSI is located on approximately 20 acres of the Owner-Controlled land outside of the Restricted Area for the plant, approximately 1,500 feet northeast of the Reactor Building. The ISFSI facility is operated under a separate license in accordance with 10 CFR 72, and is not included in the scope of this decommissioning.

2.2 Site Conditions for Final Survey

The radiological status of the FSV facility and environs has been evaluated in the FSV Initial Radiological Site Characterization Report (Ref. 5) in order to provide information pertinent to the decommissioning and final survey of FSV. Additional radiological information has been provided by the REMP; the Environmental Radiological Surveillance Program (ERSP); and the study conducted by Colorado State University's Department of Radiology and Radiation Biology from June, 1990 until October, 1990 (Ref. 6) to assess the pre-operational radiological status of the ISFSI site.

The FSV facility will be largely left intact following decommissioning. Dismantlement of structures will be confined to the PCRV (Prestressed Concrete Reactor Vessel), and portions of the Reactor Building, Turbine Building, and Liquid Waste System. Removal will be for purposes of removing contaminated structures and to provide paths for removal of contaminated piping and equipment.

Following defueling, the PCRV contained the majority of the remaining radioactive material inventory. Portions of the PCRV concrete are activated due to direct irradiation from the reactor core, and will be removed prior to Final Survey and disposed of as radioactive waste at a licensed radioactive waste disposal facility.

To date, seventeen balance-of-plant systems have been identified as being contaminated in excess of the limits for unrestricted use. All piping and equipment contaminated in excess of the limits for unrestricted use will be decontaminated and left in-place, decontaminated and free-released, or dismantled and removed from the facility and disposed of as radioactive waste at a licensed radioactive waste disposal facility.

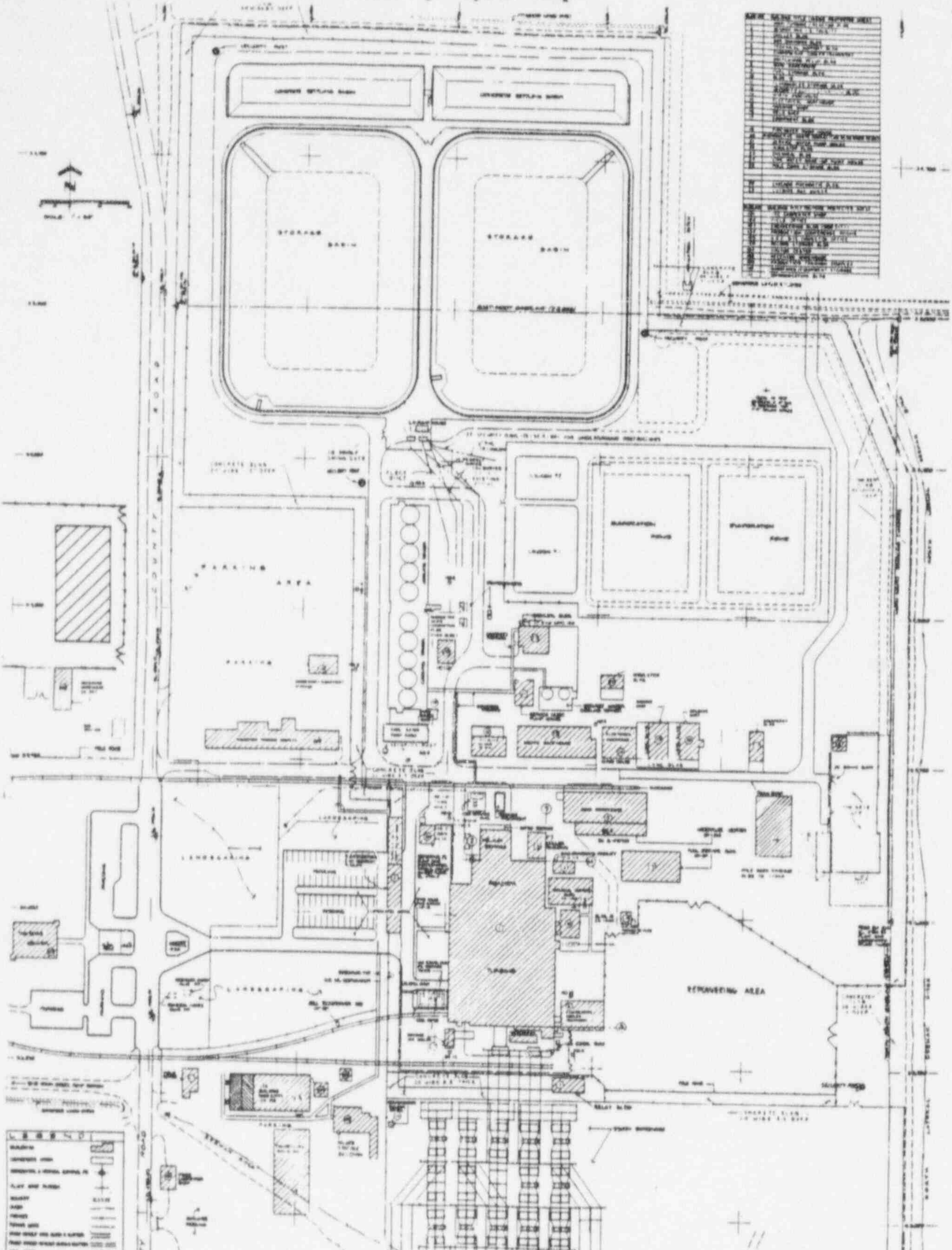
The REMP will be continued to ensure that any contamination of the site environs which could occur as a result of decommissioning activities is detected.

2.3 Scope of Final Survey

The Final Survey will include all pertinent structures, surfaces, systems and components, concentrating on those previously identified as contaminated or potentially contaminated and those identified as contaminated or potentially contaminated during the dismantlement/decommissioning phases. The final survey will include:

- Sampling outside the restricted area of PSC property, soil, pavement, water, and liquid effluent ditch and pond sediment for radioisotopic analysis and measurement of gamma exposure rate,
- Sampling inside the restricted area of PSC property, soil, basin sediment, pavement and water for radioisotopic analysis and measurement of gamma exposure rate,
- Radiological surveys of the PCRV and Reactor Building, and
- Radiological surveys of the Turbine Building, Radwaste Compactor Building, New Fuel Storage Building, Radiochemistry Laboratory, Helium Transfer and Storage System, and Liquid Radwaste System.

FIGURE 2.1
Facility Layout Map



3.0 FINAL SURVEY OVERVIEW

A general overview of the Final Survey Plan and its objectives is provided in this section. The methodology to be used during the final survey are provided in subsequent sections of this plan.

3.1 Survey Objectives

The Final Survey is designed to demonstrate that licensed radioactive materials have been removed from the Fort St. Vrain (FSV) facilities and property to the extent that residual levels of radioactive contamination are below applicable limits approved by the Nuclear Regulatory Commission (NRC). The surface contamination limits at FSV are those provided in Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors," (Ref. 7) for unaffected areas and site-specific limits derived to account for Hard To Detect Nuclides (HTDN) in affected areas. Achieving the final survey objectives demonstrates that Public Service of Colorado (PSC) has met all necessary requirements for termination of the current 10 CFR Part 50 License.

The surface contamination limits for beta-gamma emitting nuclides presented in Table 3.1, "Surface Contamination Limits", will be used to measure the effectiveness of decontamination and dismantlement efforts. Adjustment of the Regulatory Guide 1.86 surface and removable contamination limits for beta-gamma emitting nuclides to account for HTDN and alpha emitting radionuclides in affected areas is discussed in Section 3.3.4. Radiation detection instrumentation selection for performing direct measurements of total surface contamination during the Final Survey is based upon the conclusion that the most readily-measured radionuclides in determining compliance with release limits are activation products dominated by Co-60 and Eu-152 and the fission product Cs-137.

The surface contamination limits for all classifications of alpha emitters are also shown in Table 3.1. In affected areas, site specific guideline values which account for HTDN also include alpha emitting radionuclides to provide additional assurance that release criteria have been obtained. Although there is no history of fuel failure and significant quantities of alpha emitters have not been detected in plant contamination deposits, the past presence of irradiated fuel in the facility provides a potential source of alpha contamination. Methods and instruments are included in the Final Survey Plan to reconfirm the appropriate alpha classification and contamination limits in Table 3.1, and to measure alpha surface activity at levels below the appropriate limits.

Where residual contamination resulting from HTDN is suspect, analyses will be performed to determine the contribution from HTDN in total surface contamination. Based upon the relative fraction of HTDN in residual contamination, site specific guideline values will be determined for nuclides normally detected during field survey. A detailed description of this calculation is provided in Section 3.3.

Prior to, or during early phases of the Final Survey, representative samples from the locations described in Section 2.3 will be acquired and analyzed. The results of these analyses will be used as necessary to reconfirm the:

- Radionuclide inventory
- Significance of Hard to Detect Nuclides in residual contamination
- Appropriate acceptable alpha surface contamination levels
- Methodology and instrumentation required to perform final surveys

The exposure rate, as measured at a distance of one meter from accessible surfaces in facility buildings and open land areas, will be limited to an average of 5 μR/hr and a maximum of 10 μR/hr (individual measurement) above background. The background exposure rate will be determined using the methodology described in Section 4.4.3.f.

TABLE 3.1
SURFACE CONTAMINATION LIMITS

NUCLIDE[a]	AVERAGE[bc]	MAXIMUM[bd]	REMOVABLE[be]
U-nat, U-235, U-238, and associated decay products	5,000 dpm α/100 cm^2	15,000 dpm α/100 cm^2	1,000 dpm α/100 cm^2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1,000 dpm/100 cm^2	3,000 dpm/100 cm^2	200 dpm/100 cm^2
H-3 and Fe-55	200,000 dpm /100 cm^2	600,000 dpm/100 cm^2	40,000 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5,000 dpm /100 cm^2	15,000 dpm /100 cm^2	1,000 dpm /100 cm^2

[a]Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

[b]As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c]Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[d]The maximum contamination level applies to an area of not more than 100 cm^2.

[e]The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

Note: Table 3.1 limits are based upon the acceptable surface contamination levels provided in Regulatory Guide 1.86, as modified for Fort St. Vrain for tritium and iron-55.

3.2 Identity of Contaminants

An estimate of the gross radionuclide inventory was completed in June, 1989 and was included in the Supplement to Applicant's Environmental Report (Ref. 8). The total inventory (source term) due to activation and activated corrosion products three years after shutdown was estimated to be 1.3 x 10^6 curies (Ci), not including the irradiated fuel. Of this amount, approximately 99.6% was estimated to be contained in the activated and contaminated portions of the PCRV, which will be removed prior to conducting the final survey. The remainder of the inventory was estimated to be contained in balance-of-plant

surfaces and system components and constitutes approximately 0.4%, or 5.2×10^3 Ci, which will also be decontaminated or removed prior to conducting the final survey.

The design of the HTGR results in a source term which is unique to FSV. This source term is primarily a result of neutron activation of both metallic and concrete components of the PCRV and neutron activation of impurities contained in graphite components of the PCRV. These activation products include both beta-gamma emitting radionuclides such as Co-60, Eu-152 and Eu-154 which are the major contributors to whole body dose, as well as HTDN such as H-3, C-14 and Fe-55. The latter, although insignificant with respect to the Total Effective Dose Equivalent (TEDE), are significant in terms of the overall radionuclide inventory.

The Decommissioning Plan indicated that the activity in the graphite components of the PCRV is dominated by Fe-55 and H-3, which were generated due to neutron activation of impurities in the graphite. Due to the large volume of graphite and the higher specific activity of Fe-55 and H-3, these two nuclides are the largest contributors to the overall radionuclide inventory which will be removed prior to conducting the final survey. Although not the dominant nuclide in terms of radionuclide inventory, Co-60 is the dominant nuclide contributing to the TEDE in the graphite components.

The dominant nuclides for stainless steel components are Co-60, Fe-55, Ni-63 and Ni-59. The dominant nuclides for carbon steel components are Co-60, Fe-55 and Ni-63. Overall, the most significant contributor to the TEDE in stainless and carbon steel components was determined to be Co-60. The calculated radioactivity composition of the activated metallic portions of the PCRV and internal components three years after shutdown is: Fe-55, 85%; Co-60, 3.0%; and H-3, 12%. These components will be removed prior to conducting the final survey.

The PCRV concrete/rebar mixture contains many activation products due to the presence of trace elements. In the short term, Co-60 is the dominant beta-gamma emitter, while Eu-152 and Eu-154 are the dominant long term beta-gamma emitters. The nuclides contributing most to the total activity are Fe-55 and H-3. Approximately 31 inches of the activated concrete nearest to the reactor core will be removed prior to conducting the final survey.

Due to the unique design of the FSV HTGR primary and secondary systems, there was limited potential for migration of activation products and radioactive contamination to surfaces outside of the PCRV. This conclusion is supported by the contamination survey results reported in the Fort St. Vrain Initial Radiological Site Characterization Report submitted to the NRC in 1991. Ten samples were collected for extensive, independent laboratory analysis of HTDN during the Site Characterization Survey from various levels of the Reactor Building and from the interior surface of the Radioactive Gas Waste System. During normal power operations, the Radioactive Gas Waste System collected gases from contaminated systems throughout the plant. The ratio of radionuclides in the contamination in these samples was found to differ significantly from the ratio found in the activated portions of the PCRV and Internal Components. Fe-55 was found to be the

most significant contributor to the activity found in the samples collected outside of the PCRV. The ratio of Fe-55 to Co-60 in these samples was 2.2 as determined in November, 1991. Accounting for the radioactive decay of Fe-55 and Co-60 through December, 1995 will result in a ratio of 1.3.

Draft NUREG/CR-5849, "Manual for Conducting Radiological Surveys in Support of License Termination," June 1992 (Ref. 9) suggests that it may be appropriate for sites with multiple radionuclides at the time of license termination to account for those radionuclides which would contribute greater than 10% of the total radiation dose from all contaminants or which are present at concentrations which exceed 10% of their respective guideline values. Smear samples were collected to identify the major contaminants in accordance with 10 CFR 61 for the 1993 Dry Active Waste Stream originating from decommissioning activities associated with the PCRV. Decay-corrected through December, 1995, the composition included: Fe-55, 74.2%; H-3, 10.9%; Co-60, 8.6%; and C-14, 1.0%. The balance of the activity, less than 7%, was contributed by nuclides contributing less than 1% each. Of these contaminants, Fe-55, H-3, and C-14 are considered as HTDN since they are not beta-gamma emitters and are not readily detected with most conventional field survey instruments. It is possible that the concentrations of these nuclides could exceed 10% of their respective contamination guideline values. However, due to their significantly smaller dose equivalent factors, their contributions to the total effective dose equivalent would not exceed 10% of the radiation dose from all contaminants. A FSV site-specific method for evaluation of HTDN sample results has been developed and is discussed in Section 3.3.4.

3.3 Determination of Site-Specific Guideline Values

The initial FSV site characterization survey, PCRV concrete characterization and dry active waste samples have indicated the presence of multiple radionuclides, including HTDN. Therefore, surface contamination limits have been established using the guidance provided in Regulatory Guide 1.86, and site-specific guideline values have been developed to account for HTDN. The methods used to determine these site specific guideline values are detailed in Section 3.3.4 and "SV-FRS-TBD-201, *Site Specific Guideline Values for Surface Activity*.

The following site-specific guideline values are proposed for use at FSV.

3.3.1 Alpha and Beta-Gamma Removable Surface Contamination Guideline Values

The levels of readily-detected removable alpha and beta-gamma surface contamination will be limited to the applicable values provided in Table 3.1. For areas of the facility where HTDN may be present, site specific guideline values which account for HTDN and alpha emitting radionuclides are applicable.

No fuel failures were experienced in the past and no reactor-produced transuranic alpha emitters were identified outside the PCRV by surveys performed during

plant operation or in the Initial Radiological Site Characterization. However 5% of the smear samples collected during final survey of the PCRV, Refuel Floor and fuel handling equipment will be analyzed for unidentified (gross) alpha contamination. If removable alpha contamination is detected, additional investigation and survey measurement using approved survey methods will be performed to determine the area classification.

3.3.2 Alpha and Beta-gamma Total Surface Contamination Guideline Values

The levels of readily-detected alpha and beta-gamma total surface contamination (fixed plus removable) will be limited to the applicable values provided in Table 3.1. For areas of the facility where HTDN may be present, site specific guideline values which account for HTDN and alpha emitting radionuclides are applicable.

3.3.3 Exposure Rate Guideline Values

While no formal criteria exist that establish an acceptable level for exposure rate, the NRC has provided interim guidance which directs FSV to use an average limit of 5 μR/hr above background for reactor-generated gamma emitting isotopes as a limiting level for direct exposure from "residual" radioactivity (Ref. 11). This recommended limit is also consistent with statements within NUREG-0586. Based upon this guidance, 5 μR/hr above background, averaged over 10 m^2 for facility buildings and 100 m^2 for open land areas, is established as the exposure rate limit for FSV. Individual exposure rate measurements shall be limited to 10 μR/hr above background. The background for open land and indoor structural areas will be determined as discussed in Section 4.4.3.f.

The exposure rate for all radionuclides will be measured at a distance of one meter from accessible surfaces in facility buildings and open land areas. These measurements will be compared to the background for that survey unit using the methodology discussed in Section 5.2.2.

3.3.4 Hard To Detect Nuclide Guideline Values

To account for the presence of HTDN, surface contamination limits for readily detectable radionuclides may be modified based on the fraction of HTDN present and the methodology discussed in this section. Increased limits for tritium (H-3) and iron-55 (Fe-55) total and removable surface contamination were approved by the NRC under separate letters, dated 6/15/94 and 1/18/95 (Ref. 11 and 12). The contamination limits were increased from the values which had been previously inferred from Regulatory Guide 1.86 to allow average total surface contamination up to 200,000 dpm/100cm^2 and maximum total surface contamination up to 600,000 dpm/100cm^2. The removable contamination limit was increased to 40,000 dpm/100cm^2.

For affected survey units, the guideline values for average total activity and removable activity will be calculated using Equation 3.1 (derived from Equation A-2 in Draft NUREG/CR-5849, Appendix A), the acceptable levels of surface contamination from Regulatory Guide 1.86 and the increased limits for H-3 and Fe-55.

$$\textit{Site-Specific Guideline Value (SGLV)} = \frac{F}{\left(\frac{f_1}{G_1} + \frac{f_2}{G_2} + \ldots \frac{f_n}{G_n}\right)} \tag{3.1}$$

Where:

SGLV = The acceptable level of surface contamination for nuclides normally detected during field measurement, adjusted for H-3 and Fe-55 and other radionuclides not readily detectable

f_i = Fraction of the total activity contributed by each nuclide,

G_i = Guideline value for each nuclide. (From Regulatory Guide 1.86 or as specified by the NRC), and

F = Detectable nuclide fraction.

For affected survey units, analysis results of samples which are representative of the contamination which accumulated during power operation and decommissioning, and of the conditions expected at the time of final survey will be used for these calculations. In general, the calculated guideline values from a variety of samples collected within similar areas of the facility and having similar nuclide composition will be averaged to determine the appropriate guideline value for survey. Initial guideline values to account for HTDN have been established using this methodology. The basis for these values is provided in FSV-FRS-TBD-201. For the determination of the FSV removable contamination guideline value, a limit of 20,000 dpm/100cm^2 for H-3 and Fe-55 has been selected for ALARA purposes. The guideline values may be re-evaluated and altered based on continued sample analysis in accordance with established guidance. The removable contamination limit for H-3 and Fe-55 may be increased up to but not to exceed the NRC approved limit of 40,000 dpm/100cm^2 with adequate ALARA justification.

For unaffected survey structures and plant systems outside of the reactor building and radioactive waste processing areas, H-3 and Fe-55 are not considered to be nuclides of concern. In consideration of the absence of beta-gamma emitting nuclides in these areas, which is well supported by the results of operational and decommissioning surveys, it is not reasonable to propose that H-3 and Fe-55 exist

at measurable levels. For these surfaces, an adjustment to the release criteria for average total activity and removable contamination will generally not be performed. However, in the event that beta-gamma contamination is verified in excess of 25% of the guideline value, the surface will be reclassified as affected (suspect affected for building surfaces and structures) and resurveyed after any necessary adjustment to the release criteria has been made.

3.3.5 Soil and Water Activity Guideline Values

The TEDE for the average concentrations of radioactive materials above background in soil and water will be determined in accordance with the methodology contained in NUREG/CR-5512, Volume 1 (Ref. 13), or through use of a similar methodology, such as that contained in the RESRAD modelling program to assure that the total effective dose equivalent could not exceed 10 mrem during a period of one year. Using this methodology, site-specific guideline values (activity limits) will be established for radionuclides contributing to the TEDE. These values may be used for comparison with appropriate action levels.

3.3.6 Administrative Action Levels

Administrative action levels have been established identifying additional investigative actions based on the Table 3.1 residual contamination limits and the exposure rate limits identified in Section 3.3.3. Administrative action levels include:

Level 1: Removable alpha or beta-gamma activity in excess of 25% of the guideline value for unaffected areas or 50% of the guideline value for non-suspect affected areas or MDA for the equipment used to perform the survey (whichever is greater) or a single measurement greater than the GLV.

Action: Unaffected areas - investigate if single measurement >MDA; reclassify if measurement exceeds 25% of the GLV
Non-suspect affected areas - investigate if single measurement exceeds the AL; reclassify if > 10% of the measurements exceed AL or a single measurement >GLV

Level 2: Alpha or beta-gamma total surface activity measurement (fixed plus removable) in excess of 25% of the average guideline value for unaffected areas or 50% of the average guideline value for non-suspect affected areas or MDA for the equipment used to perform the survey (whichever is greater) or a single measurement greater than the GLV.

Action: Unaffected areas - investigate if single measurement >MDA; reclassify if measurement exceeds 25% of the GLV
Non-suspect affected areas - investigate if single measurement exceeds the AL; reclassify if > 10% of the measurements exceed AL or a single measurement >GLV

Level 3: Alpha or beta-gamma total surface activity (fixed plus removable) in excess of 75% of the average guideline value .

Action: Investigation

Level 4: Alpha or beta-gamma removable surface activity in excess of 75% of the guideline value .

Action: Investigation

Level 5: Exposure rates greater than 5 μR/hr above background.

Action: Determine if average over 10 m^2 for facility structures or 100 m^2 for open land areas exceeds 5 μR/hr above background or individual measurement exceeds 10 μR/hr above background. If yes, investigation and/or remediation required.

Level 6: Activity in soil or sediment greater than 25% of guideline values in unaffected areas or greater than 75% of the guideline values in affected areas.

Action: Unaffected areas - reclassify if activity is verified greater than 25% of the GLV
Affected areas - Determine if elevated activity in area (not to exceed 100 m^2) exceeds $(100/A)^{1/2}$ times the GLV, where A is the area of elevated activity in m^2; if not, determine if the weighted mean for the area of elevated activity exceeds the GLV. If yes, remediation required.

Where numerically smaller than the FSV release limits, Administrative action levels will initiate activities to ensure that sufficient additional survey data is collected to adequately characterize survey units for either release or additional remedial measures For results greater than the guideline values actions will require investigation and/or remediation or reclassification. In no instance are the Administrative action levels intended to replace the established FSV limits for determining the suitability for unrestricted use. The actions to be taken upon exceeding these levels are further discussed in Sections 4.0 and 5.0.

3.4 Organization and Responsibilities

Final Survey Plan development and implementation will be performed by qualified members of the Westinghouse Team, primarily the SEG Radiation Protection Department. Position descriptions delineating responsibilities and interfaces are contained in the Decommissioning Plan (DP) and in the FSV Decommissioning Project Radiation Protection Manual (Ref. 14).

3.5 Quality Assurance

3.5.1 General Provisions

The Decommissioning Project Quality Plan (PQP) is based on the requirements of 10 CFR 50, Appendix B as they apply to decommissioning activities. The PQP identifies the Final Survey Quality Assurance Plan which will be implemented during performance of the Final Site Survey. The Final Survey Quality Assurance Plan identifies the applicable criteria of the NRC approved SEG Quality Assurance Program and the extent to which they apply to implementation of the Final Survey Plan. Examples of the QA Program application are described in the subsections that follow. Additionally, the FSV Decommissioning Project Radiation Protection Program incorporates an internal quality control verification process which is described in this section.

a. Selection and Training of Personnel

Qualification requirements and responsibilities have been established for key personnel conducting the final survey and are controlled by the FSV Decommissioning Project Radiation Protection Manual. A training and qualification program will be established for technician and supervisory personnel selected to conduct the final survey and will include formal classroom training and performance-based training. Training and qualification records will be maintained for all personnel.

b. Instrumentation Selection, Calibration and Operation

Selection and use of instrumentation will ensure sensitivities will be sufficient to detect radionuclides at the minimum detection requirements specified in Section 4.2.2. Furthermore, the selection and use of instrumentation will assure the validity of the survey data. The selection of this instrumentation has been made based on lessons learned from similar decommissioning and Final Survey Projects. Instrument calibration will be performed either under approved procedures within the FSV Decommissioning Project Radiation Protection Manual using calibration sources traceable to the National Institute of Standards and Technology (NIST), or by qualified vendors with the results traceable to NIST. Measurements will be performed using approved written procedures for each instrument. Issue, control, and operation of all survey instrumentation will be established by instrumentation control procedures.

c. Survey Documentation

Records of final surveys will be maintained in survey packages according to procedures contained in the FSV Decommissioning Project Radiation Protection Manual. A separate package will be prepared for each survey area. The survey package will be the primary method of controlling and tracking the hard copy records of final survey results. The typical records compiled in a survey package are shown below:

- Final Survey Package Cover Sheet
- Survey Unit Diagram
- Photographs of the survey unit
- Printout of smear survey analyses
- Printout of gamma spectroscopy results (if performed)
- Printout of direct survey raw data results
- Documentation of changes or corrections to survey data
- Result of HTDN Evaluation (if performed)
- Survey Package Validation Checklist

d. Quality Control - Verification Sampling

Verification field and laboratory measurements will be performed independently on selected survey units on an ongoing basis. Instruction regarding the type and number of Quality Control measurements will be contained within the related survey package. QC measurements are collected to verify the conclusions regarding radiological status and suitability for release for unrestricted use. The results of these analyses will be compared to the action levels and guideline values which were applicable to the initial set of measurements. If the QC results do not exceed applicable action levels and guideline values, the results are considered acceptable. In the event that QC measurements do exceed the action levels or guideline values, an

investigation of the inconsistency will be performed, which may include additional measurements.

e. Written Procedures

All final survey tasks which are essential to survey data quality will be implemented and controlled by procedures contained in the FSV Decommissioning Project Radiation Protection Manual.

f. Verification of Procedures and Processes

Verification of the procedures and processes will be performed in accordance with requirements for technical verification of procedures contained within the FSV Decommissioning Project Radiation Protection Manual.

g. Chain of Custody

Procedures contained within the FSV Decommissioning Project Radiochemistry and Radiation Protection Manuals will establish responsibility for custody of samples and survey data between the point of measurement or collection until final results are obtained. Samples and survey data will be maintained in the possession of the individual, under direct surveillance or secured. When custody is transferred (e.g., when samples are sent off-site to another lab for analysis), a Chain of Custody Record will accompany the sample for tracking purposes. Chain of Custody Records will be maintained in the records management system.

h. Records Management

Generation, handling and storage of the original final survey design and data packages will be controlled by an approved procedure contained within the FSV Decommissioning Project Radiation Protection Manual.

i. Independent Review of Survey Results

The survey package from each survey unit will be given independent review to verify all documentation is complete and correct, and that release limits have been met. Also, an Independent Third Party has been selected who will perform a confirmation survey upon completion of the final survey.

j. Control of Surveyed Areas and Systems

Administrative (i.e., procedural) and physical controls for survey units will be established to preclude the possibility of contamination subsequent to completion of the final survey.

k. Control of Vendor Supplied Services

Quality-related services, such as instrument calibration and laboratory sample analysis, will be procured only from qualified vendors whose internal Quality Assurance Program is subject to audit in accordance with the FSV Project Quality Plan.

3.5.2 Final Survey Quality Control Procedure

A Final Survey procedure contained in the FSV Decommissioning Project Radiation Protection Manual will establish the quality activities not addressed in other procedures. These activities may include:

- Conduct of QC verification sampling and survey measurements
- Verification of survey measurement data
- Testing of computer calculations
- Documentation of surveys
- Custody of instruments, samples and measurement data

3.6 Training

To the maximum extent possible, personnel assigned to implement the Final Survey Plan will be selected from SEG Radiation Protection Department personnel currently engaged in decommissioning and dismantlement phases of the project.

Minimum qualifications and prerequisite training have been established within the framework of the existing FSV Decommissioning Project Radiation Protection Manual and in accordance with the DP.

Before assignment to Final Survey, personnel will receive additional formal training including:

- Overview of the Final Survey Plan
- Objectives of final survey
- Procedures governing conduct of the final survey
- Operation of the appropriate field and laboratory instrumentation used in the final survey
- Performance-based training in the collection of final survey measurements and samples

Records of training, including testing to demonstrate qualification, will be maintained in accordance with established procedures.

3.7 Laboratory Services

The FSV Radiochemistry Program is currently being conducted in accordance with an approved QA Program. On-site laboratory radioanalytical capabilities will be utilized to the maximum extent possible to support the final survey. On-site capabilities include gamma spectroscopy (HPGe) of smear and bulk samples, low energy beta/x-ray spectroscopy, liquid scintillation, and gas proportional counting. Off-site laboratory sample analysis will be procured only from qualified vendors whose internal QA Program is subject to audit in accordance with the Project Quality Plan. Contracts are currently in place with two qualified vendors for specialized radiological analysis of samples on an as-needed basis.

3.8 General Survey Plan

Three categories or types of survey units have been established including:

- Structures, to include building interiors and exteriors, and the exterior surfaces of plant system components,
- Plant systems, to include the interior surfaces of process piping and components and ventilation ductwork, and
- Open land areas, to include facility grounds and liquid effluent pathway

These categories have been selected in order to group similar physical characteristics into survey units. The survey effort for each survey unit will be based upon its classification as affected or unaffected as discussed in Section 3.8.2.

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

The final survey planning and implementation process for each survey unit will involve the following activities:

- History file preparation
- Initial classification
- Walkdown
- Turnover for final survey (affected survey units only)
- Survey design
- Modifications
- Work planning and scheduling
- Survey instructions
- Field support
- Survey measurements
- Investigation, if required
- Reclassification, if required

- Restoration and isolation, if required
- Access control measures

These steps are described in the following sections.

3.8.1 History File Preparation

The history file will be a compilation of relevant operational and decommissioning data in a standardized format. The purpose of this process is to provide a substantive basis for the survey unit classification, and hence the level of intensity of the final survey. Similarly, for structures and open land areas, the extent of radioactive materials involvement in the area (if any) is summarized. The history file may contain:

- Review of the system description
- Operating history which could affect radiological status
- The FSV Initial Radiological Site Characterization Report
- Radiological surveys performed during decommissioning phases
- Other relevant information

3.8.2 Initial Classification

The initial classification of the survey units into affected and unaffected areas provides an overall planning basis for the final survey. All areas of the FSV site (including structures, plant systems and open land areas) will not have the same potential for residual contamination and therefore do not require the same level of survey coverage to achieve an acceptable level of confidence that an area satisfies the established release limits. Affected and unaffected areas are defined as follows:

Affected areas: Areas that have potential radioactive contamination (based on plant operating history) or known radioactive contamination (based on past or preliminary radiological surveys). Affected areas within a room or building are further subdivided into suspect affected and non-suspect affected areas, to allow a more concentrated survey effort in those areas most likely to be contaminated.

Non-suspect affected areas: The designation assigned to ceiling and wall surfaces above two meters within an affected area where it is not expected that radioactive materials exist at average levels in excess of 25% of the removable or total activity release criteria, or the MDA for the equipment used to perform the survey, whichever is larger; and it is not expected that individual measurements exist in excess of 50% of the release criteria, or the MDA for the equipment used to perform the survey, whichever is larger. This designation is based on engineering judgement, in consideration of operational history, characterization data, operational

surveys performed in support of decommissioning, routine surveillance and turnover surveys.

Suspect affected areas: The designation assigned to floor and wall surfaces below two meters within an affected area. Also, the designation assigned to all additional surfaces if there is reason to believe that there was licensed material present, unless it meets the criteria for a non-suspect affected survey unit.

Unaffected areas: All areas not classified as affected. These areas are not expected to contain residual radioactivity, based on a knowledge of site history and previous survey information. The presence of licensed material in excess of MDA or previous remediation would preclude a survey unit from being classified as unaffected.

Survey data contained in the FSV Initial Radiological Site Characterization Report, operational history, operational surveys performed to support decommissioning, routine surveillances, turnover surveys (for affected survey units only), and assessment of Westinghouse Team and PSC personnel knowledgeable of the facility conditions will be utilized when applying the classification criteria approved for Fort St. Vrain.

3.8.3 Walkdown

The walkdown will be a key activity in the preparation of the survey design. For systems, it will include review of system flow diagrams, piping drawings and a physical walkdown of the system. Structures and open land areas will also be walked down. The principal objective is to assess the physical scope of the survey area and to identify the breakdown into units and subunits. Specific requirements will be identified for accessing the survey area and support functions necessary to conduct the final surveys, such as scaffolding, component disassembly, interference removal, engineering modifications, electrical tagout and system alignment to provide access for surveys. Safety concerns such as access to confined spaces, high walls, and ceilings will be identified. It is noted that for survey units involved with decommissioning activities, the walkdown is best completed when the final configuration is known, usually near or after completion of decommissioning activities.

3.8.4 Turnover for Final Survey

Prior to acceptance of a survey unit for the final survey, a number of conditions will be satisfied. These include:

- Decommissioning activities having the potential to contaminate the survey unit must be completed.
- All tools and equipment not required to perform the survey must be removed.

- Housekeeping and area cleanup must be completed.
- Decontamination of affected structural areas and system components must be completed.
- Operational radiological surveys have been performed in areas where remediation has occurred to ensure no additional remediation is necessary (affected survey units only).

A physical walkdown of the survey unit will be performed prior to turnover to ensure the above conditions have been met. Scaffolding needed to be left in place for conducting the final survey should be identified during turnover preparation. Results of the operational radiological survey verifying the status of the area will be included in the final survey package.

3.8.5 Survey Design

Establishing a uniform grid system is one of the methods that can assist in the systematic selection of measuring and sampling locations while providing a mechanism for referencing a measurement to specific locations. Grids are also convenient means to determine average activity levels.

A salient feature of the FSV Final Survey Plan is to minimize (limit) establishing a uniform grid system without compromising the quality of survey data and ensuring the following:

- Reference of radiological survey data to a specific survey location in order to evaluate significance of elevated activity levels (hot spots)
- Correlation of radiological survey data to a specific surface of known area in order to estimate average activity levels
- Collection of an adequate number of measurements or samples from within a designated surface area to enable statistical evaluation

The methods used will provide sound statistical conclusions for determination of the final survey status. Additional detail is provided in Section 4.0.

3.8.6 Modifications

After the survey design is prepared, component disassembly or system modification requirements will be identified. In instances where disassembly or modifications are required for surveys, a work control process will be used to coordinate activities.

This process will include prerequisites for the protection of systems and structures from potential contamination after completion of the survey.

3.8.7 Work Planning and Scheduling

Upon completion of any required engineering review, the physical modifications will be specified. Field work will be implemented via a work control process described in documented procedures. The process will identify components which require opening, identify modifications, indicate restoration requirements and indicate whether a system is to be isolated or returned to service. The process will also be used to initiate support work and tagouts necessary for surveys of structural and open land survey units. The survey unit support work will then be placed on the work schedule for implementation.

3.8.8 Survey Instructions

The final survey instructions will be contained within the survey package. These instructions will specify the number and type of radiological measurements to be taken at each location or component identified in the survey design. The instructions will also identify other samples to be collected. The survey instructions will be prepared in accordance with procedures contained in the FSV Decommissioning Project Radiation Protection Manual.

3.8.9 Field Support

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

3.8.10 Survey Measurements

Final survey measurements will be conducted in accordance with procedures contained in the FSV Decommissioning Project Radiation Protection Manual and specific survey instructions for the survey unit provided in the survey package. A sufficient number of measurements will be taken to conclusively demonstrate that the release limits have been met. The measurements will be obtained by conducting surveys using approved methods and techniques such as surface scans, direct measurements of surface contamination, smear samples for removable surface contamination, and exposure rate measurements. These measurements are discussed in Section 4.0.

3.8.11 Investigation

The purpose of performing investigation surveys is to confirm the initial final survey measurement, and if verified to exceed action levels, to define the bounds of the elevated activity. Depending on the results of the investigation survey, the survey unit may require remediation and/or reclassification and resurvey. The

investigation survey will generally consist of a scan survey over 100% of the accessible surfaces and measurements for removable and total activity. Typically, the investigation survey will include an area of at least nine square meters, centered around the point of interest. A minimum of five measurements will be collected from each of the grids. The average for each of the grids will be calculated and compared to the average guideline value.

Investigation surveys will be performed within an unaffected survey unit if an individual final survey measurement for removable or total surface activity indicates the presence of licensed material (in excess of MDA). The action levels to initiate investigation surveys within non-suspect affected survey units are individual final survey measurements for removable or total surface activity in excess of 50% of the guideline value. The action levels to initiate investigation surveys within suspect affected survey units are individual final survey measurements for removable or total surface activity in excess of 75% of the guideline value.

For plant system survey units, the investigation survey will consist of additional measurements for total and removable activity at the location(s) identified, measurements at adjacent locations, and measurements at other biased locations within the plant system where residual activity would be most probable.

If during the investigation survey the initial measurement results cannot be confirmed, the initial measurement results will be flagged as suspect (anomaly) data in the final survey data base. The computer down loads, filed as part of the survey package, will be annotated as such. The investigation measurement results will be considered as the final survey of record and will be included in the calculation of the mean and standard deviation for the survey unit to determine compliance with the unrestricted use criteria.

If during the investigation survey the initial measurement results are confirmed, and remediation and/or reclassification is required, the initial measurement results will be archived as characterization data in the final survey data base. The computer down loads, filed as part of the survey package, will be annotated as such. The results of the post-remediation final survey, or the final survey subsequent to reclassification, will be considered as the final survey of record and will be included in the calculation of the mean and standard deviation for the survey unit to determine compliance with the guideline value. For instances where resurvey is performed, the final report will be annotated indicating that the initial data for a survey unit has either been replaced or supplemented with subsequent survey data.

If during the investigation survey the initial final survey measurement results are confirmed, but remediation and/or reclassification is not required, the initial measurement results and the investigation survey results will be considered as the final survey of record and will be included in the calculation of the mean and

standard deviation for the survey unit to determine compliance with the guideline values.

3.8.12 Reclassification

The result of investigating the presence of elevated activity may require that a survey unit be reclassified to ensure adequate survey coverage. In unaffected survey units, if the investigation verifies that removable or total surface activity exceeds 25% of the guideline values, the survey unit will be reclassified as an affected area and resurveyed. In non-suspect affected survey units, if greater than 10% of the initial final survey measurements are verified to exceed 50% of the guideline values, the survey unit will be reclassified as suspect affected and resurveyed. Additionally, if an individual measurement exceeds the guideline values, the survey unit will be reclassified and resurveyed.

3.8.13 Restoration and Isolation

a. Systems

After final survey measurements have been obtained, reviewed and approved, and any necessary verification survey measurements have been completed, the system will be restored and components replaced as specified in the work control process. If required, the system will be isolated to protect against intrusion of radioactive contamination. Isolation and control of plant systems after completion of the final survey will be performed under an approved procedure.

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

b. Structures and Open Land Areas

After measurements are completed in structures and open land areas, controls will be used to prevent or minimize possible radioactive contamination. These will be controlled by a procedure, as appropriate. Numerous structural survey units have been defined within the Radiological Controlled Area (RCA) of the FSV facility. These include all of the structural survey units which comprise the Reactor Building, portions of the Turbine Building, the Radwaste Compactor Building, the New Fuel Storage Building, Technical Support

Building which contains the Radiochemistry Laboratory, and the Liquid Effluent System. As surveys are completed in contiguous RCA survey units (with completion of the surveys being indicated by completed, approved release records), reduction of the RCA boundaries will be made. Removal of such areas from the RCA provides additional assurance that material containing radioactivity is not used in, or transported through areas where final surveys have been initiated.

3.8.14 Access Control Measures

Since all decommissioning activities will not be complete prior to the start of final survey, measures will be implemented to protect survey units from contamination subsequent to final survey. In all cases, decommissioning activities creating a potential for the spread of contamination must be completed within each survey unit prior to final survey. Additionally, decommissioning activities which create a potential for the spread of contamination to adjacent survey units will be evaluated and controlled by the Final Survey Group and the Radiation Protection Department.

Prior to collection of final survey data from specific survey unit(s) where there is potential for contamination subsequent to survey, implementation of personnel training, along with one or more of the following control measures is required:

- Installation of barriers to control ingress and egress to repower construction areas,
- Installation of postings requiring personnel to perform contamination monitoring prior to clean area access,
- Locking entrances to non-vital clean areas of the facility,
- Installation of chain locks on valves and blind flanges in plant systems,
- Installation of tamper-evident labels.

After the completion of final survey within a contiguous area of the facility, routine smear surveys will be performed to identify conditions which could affect the final status. Based upon the results of routine monitoring, or in the instance of a potentially-contaminating event within a survey unit or group of survey units, an investigation survey may be performed.

The investigation survey will consist of a 10% re-survey of the survey unit(s), concentrating on areas of highest contamination potential. If the investigation survey indicates a significant change in the radiological status of the survey unit which exceeds the action levels appropriate for the initial classification of the survey unit, then additional measurement and/or reclassification and remediation may be necessary.

A description of potentially-contaminating events and the results of investigation surveys will be included in the final report.

3.9 Schedule

The final survey is planned to be completed in four major phases which encompass distinct portions of the facility. The first phase is expected to include unaffected outside area survey units, unaffected structural survey units outside the RCA, and unaffected systems and structural survey units within the Turbine Building. The second phase is expected to include unaffected systems and structural survey units within the Reactor Building, and the liquid effluent pathways. The third phase is expected to include affected systems and the remaining survey units affected by the removal and temporary storage of irradiated and/or contaminated components from the facility. The final phase is expected to include laydown areas and temporary radioactive waste storage areas. Upon completion of each phase, the release records will be compiled and the survey units will be made available for NRC Confirmatory Surveys.

3.10 Final Survey Reporting

Interim Reports as well as a Final Summary Report will be prepared for submittal to the NRC to meet the intent of Regulatory Guide 1.86 for final survey reporting. The reports will follow the guidance of Draft NUREG/CR-5849 regarding content and format. Interim Reports will be submitted upon completion of each of the major survey phases discussed in Section 3.9 above. A Final Summary Report will be prepared summarizing results of the entire decommissioning project. The Interim and Final Summary Reports are described in Section 6.0, "Final Survey Reporting".

4.0 SURVEY PLAN AND PROCEDURES

4.1 General

The design approach of the FSV Final Survey reflects the final configuration of the facility, which will largely be left intact with the majority of equipment abandoned in place. The major survey effort will be confined to the survey units contained within the Reactor Building, portions of the Turbine Building where radioactive materials were used or handled, the Radwaste Compactor Building, the New Fuel Storage Building, the Technical Support Building which contains the Radiochemistry Laboratory, the Liquid Effluent System, Liquid Effluent Discharge Pathways, and remaining portions of the plant systems identified as contaminated or potentially contaminated in the FSV Initial Radiological Site Characterization Report. These survey units will be classified as affected. If radioactive material process or storage areas are established outside these survey units to support decommissioning operations, these areas will be identified and classified on a case-by-case basis.

The remainder of the areas within the scope of the survey will be classified as unaffected. All radioactive material handling, movement and storage on the site has been controlled under approved procedures. No significant FSV actions have resulted in a radiological impact on the site environs as evidenced by extensive site surface measurements and soil sampling reported in the FSV Initial Radiological Site Characterization Report, the Radiological Environmental Monitoring Program (REMP), the Environmental Radiological Surveillance Program (ERSP) and the study conducted by Colorado State University's Department of Radiology and Radiation Biology from June 1990 until October 1990. The later study was used to assess the pre-operational radiological environmental status of the Independent Spent Fuel Storage Facility (ISFSI) site. In addition, procedures have been established to document events that could result in a radiological impact to the site during decommissioning.

The Final Survey Plan and implementing procedures are designed to focus primarily on remaining plant structures and systems in the affected areas. Instrumentation has been selected and survey procedures developed to detect and measure surface activity levels (primarily Co-60, Cs-137 and Eu-152) and exposure levels in these affected areas.

4.2 Instrumentation

Radiation detection and measurement instrumentation for final surveys has been selected to provide reliable operation and adequate sensitivity to demonstrate that the measurements taken are sufficient to conclusively demonstrate that the release limits have been met. Commercially available portable and laboratory instruments and detectors produced by several manufacturers have been selected based upon detection sensitivity, operating characteristics and expected performance in the field. A listing of typical detectors and their detection characteristics are summarized in Tables 4.1 and 4.2. However, surveys will be performed using the most suitable equipment available and

survey measurements shall not be limited to this listing. Recording instruments (survey meters) for use with these detectors have also been evaluated. Instrumentation to be used for exposure rate measurements and special purpose measurements are also described.

4.2.1 Instrument Detector Description

The principal detectors selected for final survey measurements are shown in Table 4.1, "Final Survey Instrumentation Summary". The detectors used for total surface activity monitoring will, for the most part, be operated with computer-based data logging survey meters.

4.2.2 Detection Sensitivity

The detection sensitivity of the detectors selected for surface activity measurements has been evaluated. These results are illustrated in Table 4.2, "Typical Detection Sensitivities for Tc-99 and Th-230". The results are shown for the principal instruments that are expected to be used for field and laboratory beta-gamma and alpha total surface activity measurements.

Field and laboratory instrument minimum detectable activities are calculated using Equation 4.1. The MDA is dependent upon several factors: sample count time, background count time, background count rate and detector efficiency. Smear counters for measurement of removable surface activity are of modern design (anti-coincidence low-background) and will be used to determine the activities for both alpha and beta radiation. The MDA for scan survey measurements is calculated using Equation 4.1 by defining the counting time (t_s) as the time the point of interest is below the sensitive area of the detector.

Count times are selected to ensure that the measurements are sufficiently sensitive with respect to applicable guideline values. For example, for unaffected areas, the count times associated with measurements for total activity (fixed point measurements), removable activity (smears) and gamma spectral analysis (soil and sediment samples) will be adjusted as required to ensure an MDA of <25% of the applicable guideline value. For affected areas, the MDA for measurements of removable surface activity will also be set at <25% of the site specific guideline value, and the MDA for measurements of total surface activity will be set at <50% of the site-specific guideline value.

It should be noted that it may not be possible to scan or take exposure rate measurements at 25% of the applicable guideline value due to significant background interference. The exposure rate measurement instrument sensitivities for field equipment in the normal modes of operation are:

Pressurized ion chamber (Reuter-Stokes): 1-3 μR/hr

Ludlum Model 44-2/sodium iodide detector: 2 - 3 μR/hr.

TABLE 4.1
FINAL SURVEY INSTRUMENTATION SUMMARY

Radiation Detected	Detector Type	Detector Area-Density	Manufacturer & Model #	Units
Surface Beta-Gamma, Alpha	Gas flow proportional	125 cm^2 0.8 mg/cm^2	Ludlum 43-68	cpm
Surface Beta-Gamma, Alpha	Gas flow proportional	300 cm^2 0.8 mg/cm^2	Ludlum 43-47	cpm
Surface Beta-Gamma, Alpha	Gas flow proportional	550 cm^2 0.8 mg/cm^2	Ludlum 43-37	cpm
Surface Beta, Alpha	ZnS(Ag) Plastic Scintillation	125 cm^2 1.2 mg/cm^2	Ludlum 43-89	cpm
Surface Beta-Gamma	Geiger-Mueller	15.5 cm^2 1.7 mg/cm^2	Ludlum 44-40	cpm
Surface & Liquid Activity Beta/X-ray	Liquid Scintillation	N/A	Beckman LS 3801	cpm
Removable Surface Beta-Gamma, Alpha	Gas flow proportional	24.2 cm^2 80 $\mu g/cm^2$	Tennelec LB-5100	cpm
Gamma Exposure Rate	Pressurized Ion Chamber (PIC)	8 L Sphere 2.85 g/cm^2	Reuter-Stokes RSS-112	μR/hr
Gamma Exposure Rate	Sodium Iodide scintillation	1" X 1"	Ludlum 44-2	μR/hr (Cs-137)
Gamma Spectroscopy	High-purity Germanium	2" X 2"	Canberra Genie-PC®	Geometry Dependent
Background and Special Measurements	High-purity germanium	2" X 2"	EG&G Ortec Nomad	Geometry Dependent

TABLE 4.2
TYPICAL DETECTION SENSITIVITIES FOR Tc-99 AND Th-230

Instrument	Radiation	Count time (min)	Background (cpm)	Efficiency[a] (cpm/dpm)	Background Count Time (min)	MDA (dpm/100 cm^2)	Scan Survey MDA (dpm/100 cm^2)
Ludlum Model 43-68	beta	0.17	750	0.21	1	960	2300
Ludlum Model 43-68	alpha	0.17	3	0.20	5	122	570
Ludlum Model 44-40 (scaler mode)[b]	beta-gamma	0.2	50	0.12	5	3580	5000
Tennelec LB-5100	beta	1	2.3	0.45	1	21.7	N/A
Tennelec LB-5100	alpha	10	0.1	0.31	10	2.5	N/A

Table 4.2 Notes:

[a]Sources used for efficiencies are traceable to the National Institute of Standards & Technology (NIST). Sources consist of Tc-99 or Th-230 uniformly deposited over areas of 17.3 cm^2 or 100 cm^2.

The efficiency is determined by counting the source with the detector in a fixed position one-half cm from the source.

Scan survey sensitivity (MDA) was calculated assuming a scan rate of 5 cm/second.

[b]Due to its limited sensitivity, the Ludlum 44-40 detector should only be used for scan surveys when size constraints preclude use of larger detectors.

4.2.3 Minimum Detectable Activity Calculation

MDA values for field and laboratory counting instrumentation are determined using the following equation:

$$MDA = \frac{\frac{2.71}{t_s} + 3.29\sqrt{\frac{R_b}{t_s} + \frac{R_b}{t_b}}}{E*(a/100)} \tag{4.1}$$

Where:

MDA = the minimum amount of activity that can be statistically detected above background with a 95% probability and with a maximum of 5% probability of falsely interpreting background activity as activity due to contamination (dpm/100 cm^2),

t_s = sample counting time (minutes),

R_b = background count rate in counts per minute (cpm),

t_b = background counting time (minutes),

E = counting efficiency (cpm/dpm), and

a = area of the detector, or the area sampled for smear samples (cm^2)

4.2.4 Calibration and Maintenance

Instruments and detectors used to conduct final surveys will be calibrated and maintained in accordance with instrumentation procedures contained in the FSV Decommissioning Project Radiation Protection Manual. If vendor services are used, these services will be conducted in accordance with approved procedures and an internal QA Program is subject to audit in accordance with the PQP. Radioactive sources used for the purpose of calibration will be traceable to the National Institute of Standards and Technology (NIST) for both FSV and vendor operations.

4.3 Survey Plan

4.3.1 Classification

All areas of the FSV site (including structures, plant systems and outdoor areas) will not have the same potential for residual activity and will not require the same level of survey coverage to achieve an acceptable level of confidence that the site satisfies the established release limits. Therefore, each survey unit will be

classified as "affected" or "unaffected" in accordance with final survey procedures. Affected and unaffected areas are defined in Section 3.8.2. Affected areas within a room or building may be further subdivided into suspect affected and non-suspect affected areas, to allow for a concentrated survey effort in those areas most likely to be contaminated, as follows:

- Floors and lower walls (up to two meters above the floor) of affected areas will be designated as suspect affected and receive 100% survey coverage.

- Upper walls, ceilings and exterior surfaces of equipment (more than two meters above the floor) may be designated as suspect affected or non-suspect affected. These areas may be designated as non-suspect affected if:

  (1) It is not expected that average removable or total activity will exceed the larger of 25% of the unrestricted use criteria, or the MDA for the equipment used to perform the survey, and

  (2) It is not expected that individual measurements for removable or average total activity will exceed the larger of 50% of the unrestricted use criteria, or the MDA for the equipment used to perform the survey.

- An affected survey unit will be classified as suspect affected if there is reason to believe that there was licensed material within the survey unit, unless it meets the above criteria for a non-suspect affected area.

- Upper walls, ceilings and exterior surfaces of equipment classified as non-suspect affected will not require 100% survey coverage but will require a sufficient number of measurements (usually a minimum of 30 measurements each) from horizontal and vertical surfaces to ensure that the mean value is less than the average surface activity limit at the 95% confidence level.

Initial classification of individual survey units is based on data provided in the FSV Initial Radiological Site Characterization Report, history of radioactive materials involvement or potential for contamination of the survey unit and recommendations of Westinghouse Team and PSC personnel knowledgeable of the facility conditions. Data from operational surveys performed to support decommissioning, routine surveillances, and turnover surveys (for affected survey units only) will also be used to support initial classification. Table 4.3, "Initial Facility Survey Area Classification," illustrates the initial classification of facility structures, systems and outside locations. These classifications will be either verified or updated during conduct of the final survey.

Classification of individual survey units will be performed following the logic depicted in Figure 4.1, "Survey Unit Classification Process."

Reclassification

Based on information obtained during final survey activities, it may be necessary to reclassify certain survey units to ensure that the measurement frequency is appropriate for the contamination potential. Survey units will be reclassified as follows:

- For unaffected survey units associated with building surfaces and structures or plant systems, if activity in excess of 25% of the guideline values for average total activity or removable activity is verified, the area will be reclassified as a suspect affected area and resurveyed.

- For unaffected open land survey units, if activity in excess of 25% of the guideline value is verified in soil or sediment samples, the area will be reclassified as an affected area and resurveyed.

- For non-suspect affected survey units associated with building surfaces and structures, if greater than 10% of the initial final survey measurements are verified to exceed 50% of the guideline values for average total activity or removable activity or if an individual measurement exceeds the guideline value, the area will be reclassified as an suspect affected area and resurveyed.

TABLE 4.3
INITIAL FACILITY SURVEY AREA CLASSIFICATION

LOCATION	AFFECTED	UNAFFECTED
INSIDE RESTRICTED AREA		
Reactor Building	100%	
Turbine Building	10%	90%
Search and I.D. Building		100%
Gas Charging Building		100%
Technical Support Building	20%	80%
Evaporative Cooling Building		100%
Switchyard Relay Building		100%
Main Warehouse Building	10%	90%
New Fuel Storage Building	100%	
Building 10	10%	90%
Flammables Storage Building		100%
Security Administration Building		100%
North Warehouse Building		100%
Electrical Warehouse Building		100%
Machine Shop Building	10%	90%
Weld Shop Building		100%
Equipment Building (PSC Carpenter Shop)		100%
Construction Workshop Building		100%
Firewater Pumphouse		100%
Radioactive Waste Compacting Building	100%	
Service Water Pumphouse		100%
Insulator Building		100%
Chemical Building		100%
Circ. Water Make-up Pumphouse		100%

TABLE 4.3 (CONT.)
INITIAL FACILITY SURVEY AREA CLASSIFICATION

LOCATION	AFFECTED	UNAFFECTED
Pole Barn Storage Building		100%
QA/QC Lab. and Snubber Building		100%
Kennedy Building		100%
Chicago Pneumatic Building		100%
Outside Aux. Boiler Building		100%
ACM Diesel Generator Building		100%
Outside Temporary Storage/Laydown Areas	100%	
OUTSIDE RESTRICTED AREA		
Employee and Construction Parking Lots		100%
PSC Carpenter Shop (Storage Building)		100%
Field Office		100%
Engineering Building		100%
Production Conference Rooms		100%
N.R.C. Site Inspector's Office		100%
Record Storage		100%
Visitor Center		100%
Receiving Warehouse		100%
Production Training Complex		100%
Decommissioning Project Offices		100%
Gardening/Equipment Storage		100%
Communication Building		100%
East and West Storage Basins		100%
Sewage Lagoon #1 and #2		100%
East and West Evaporation Ponds		100%
East and West Settling Basins		100%

TABLE 4.3 (CONT.)
INITIAL FACILITY SURVEY AREA CLASSIFICATION

LOCATION	AFFECTED	UNAFFECTED
Mechanical Draft Cooling Towers		100%
Liquid Effluent Pathways	80%	20%
Farm Pond	100%	
PLANT SYSTEMS		
System 22, Secondary Cooling System		100%
System 25, Liquid Nitrogen System		100%
System 29, Gas Charging System		100%
System 31, Feedwater and Condensate System		100%
System 32, Feedwater Heater Vents and Drain		100%
System 33, Water Treatment System		100%
System 41, Circulating Water System		100%
System 42, Service Water System		100%
System 43, System Injection System		100%
System 44, Domestic Water System		100%
System 45, Fire Protection System		100%
System 48, Alternate Cooling System		100%
System 51, Turbine Generator and Auxiliaries		100%
System 52, Main Turbine Steam Supply System		100%
System 53, Extraction Steam System		100%
System 54, Turbine Lube Oil Purification		100%
System 75, Turbine Building Vents and Drains, Control Room Service Building HVAC		100%
System 84, Auxiliary Boiler and Heating System		100%
System 91, Hydraulic Power System		100%

TABLE 4.3 (CONT.)
INITIAL FACILITY SURVEY AREA CLASSIFICATION

LOCATION	AFFECTED	UNAFFECTED
System 11, PCRV Penetration Seal Leak Collection System	100%	
System 13, Fuel Handling Equipment System	100%	
System 14, Fuel Storage Equipment System	100%	
System 16, Auxiliary Equipment	100%	
System 21, Primary Coolant System and Helium Circulator Auxiliary System	100%	
System 23, Helium Purification System	100%	
System 24, Helium Storage System	100%	
System 46, Reactor Plant Cooling Water System	100%	
System 47, Purification Cooling Water System	100%	
System 55, Turbine Vents and Drains	10%	90%
System 61, Decontamination System	100%	
System 62, Radioactive Liquid Waste System	100%	
System 63, Radioactive Gas Waste System	100%	
System 72, Reactor Building Drain System	100%	
System 73, Reactor Building HVAC System	100%	
System 79, Radiochemistry Laboratory Ventilation	15%	85%
System 93, Controls and Instrumentation	80%	20%

4.3.2 Designation of Survey Units and Measurement Locations

Measurement locations will be clearly identified to provide a method of referencing survey results to survey unit locations. Whenever it is appropriate and cost effective, gridding such as that illustrated in Figure 4.2, "Reference Grid Layout," will be used. However, the physical grid layout may be substituted with the use of surface markings. Due to the large number of obstructions and non-uniform surfaces that will remain in the facility, gridding will be used mainly for portions of suspect affected areas (1 x 1 meter grids) and in affected open land areas (10 x 10 or 5 x 5 meter grids). Non-suspect affected or unaffected areas will not generally be gridded. Where gridding is used, measurements will be obtained in accordance with Section 4.3.3.

Based upon the HTGR design and the large number of systems and equipment to be abandoned in place, engineering judgement will be used to logically divide a survey area into survey units (or subunits) to support reproducible QC and NRC Confirmatory Survey measurements. This will be accomplished by selecting existing facility characteristics such as; horizontal and vertical structural support beams, systems, components, piping runs, physical sheet steel weld seams and concrete pour seams to define the boundaries of survey units within survey areas. Actual lengths, widths and distances will be included to support reproduciblilty.

A survey unit is usually a contiguous area with similar characteristics and contamination potential. The size of the survey unit will be chosen to assure that the total number of data points (usually 30) and/or the spacing (frequency) of measurements/sampling enables a statistical evaluation of the data collected. Survey units will not include both affected and unaffected areas. For plant systems, a survey unit may be defined to include an entire system; however, individual survey units will typically not include more than one system.

In addition, the use of photographs, structural and actual system drawings or maps will be included to support survey unit (or subunit) identification. This method of area designation will also serve to assist survey personnel to define areas of elevated activity requiring additional investigational surveys and/or remediation. Areas where gridding may not be feasible include, but are not limited to, interior structural surfaces, system piping, portions of interior walls within the Reactor Building, main steam and feedwater piping areas and where interferences exist and they are not expected to be removed.

4.3.3 Measurement Frequency

Measurement frequencies are selected to allow for a concentrated survey effort in those areas most likely to be contaminated, taking into account the type and size of the survey unit, as follows:

Building Surfaces and Structures

- **Suspect Affected Areas**

The final survey of suspect affected areas includes a scan of 100% of accessible surface area and a minimum of 30 measurements per survey unit for removable activity (smears), total activity (fixed point measurements) and exposure rate. The location of these measurements will ensure uniform coverage of the area and the investigation of potentially elevated areas of activity identified during the scan survey.

If the area is gridded, measurements will be taken at each grid intersection. Where survey units are not gridded, the number of measurement locations will be based in part on the size of the area being surveyed. To ensure adequate coverage in survey units that are not gridded, the following measurement frequencies have been established:

For areas ≤20 square meters	A minimum of 30 measurement locations
For areas >20 square meters	Equivalent to 1 meter intervals

- **Non-suspect Affected Areas**

The final survey of non-suspect affected areas above 2 meters includes a minimum of 30 measurements per survey unit for removable activity, total activity and exposure rate, and a scan survey of the accessible surfaces at each measurement location. In general, these areas will not be gridded. Measurement locations will be selected from locations most likely to accumulate activity and the investigation of potentially elevated areas of activity identified during the scan. The number of measurement locations will be based in part on the size of the area being surveyed.

For areas ≤600 square meters	A minimum of 30 measurement locations
For areas >600 square meters	A minimum of 1 measurement location for each 20 square meters surveyed

- **Unaffected Areas**

The final survey of unaffected areas includes a scan of approximately 10% of the accessible surface area comprising floors and walls below 2 meters, and a minimum of 30 measurements per survey unit for removable activity, total activity, and exposure rate. In general, these areas will not be gridded. Measurement locations will ensure uniform coverage of the area and the investigation of potentially elevated areas of activity identified during the scan. The number of survey locations will be based in part on the size of the area being surveyed.

For areas ≤1500 square meters	A minimum of 30 measurement locations
For areas >1500 square meters	A minimum of 1 measurement location for each 50 square meters surveyed

It is expected that the final survey of building exteriors will be in accordance with the protocol established for unaffected building surfaces and structures.

Plant Systems

Plant systems will be classified as either affected or unaffected. The exterior surfaces of plant system components will be surveyed in accordance with the protocols for building structures.

- **Affected Plant Systems**

The final survey of the interior surfaces of affected plant systems will include a scan survey of approximately 25% of the accessible surface within each system, and a minimum of 30 measurements of removable activity and total activity within each survey unit. Measurement locations will be selected from locations where activity is most likely to accumulate. If appropriate, sampling and analysis of scale and sediment will be performed. Exposure rate measurements will be limited to interior surfaces of tanks and large vessels.

- **Unaffected Plant Systems**

The final survey of the interior surfaces of unaffected plant systems will include a minimum of 30 measurements within each system of removable activity, total activity and scan survey to be performed in the immediate vicinity of the measurement. Measurement locations will be selected from locations where activity is most likely to accumulate. If appropriate, sampling and analysis of scale and sediment will be performed. Exposure rate measurements will be limited to interior surfaces of tanks and large vessels.

Open Land Areas

Open land areas will be classified as affected or unaffected. Open land areas included in the final survey are the restricted area and the area adjacent to the liquid effluent pathways. The final survey for open land areas with paved surfaces will be in accordance with the protocols established above for building surfaces and structures.

- **Affected Open Land Areas**

The final survey of affected open land areas without paved surfaces includes an exposure rate scan survey of 100% of the accessible surface area, and a minimum of 30 exposure rate measurements and surface soil samples per survey unit. As described in the Section 4.3.2, these areas will normally be gridded into 10 meter by 10 meter, or 5 meter by 5 meter grids. Soil samples collected within each 10 meter by 10 meter grid will be from locations equidistant between the center and each of the 4 corners. Within each 10 meter by 10 meter grid, 4 soil samples will be collected and an exposure rate measurement will be made at 1 meter from the surface at the location of the soil samples. Soil samples collected within each 5 meter by 5 meter grid will be from the center of each grid. Within each 5 meter by 5 meter grid, 1 soil sample will be collected and an exposure rate measurement will be made at 1 meter from the surface at the location of the soil sample. The location of these measurements will ensure uniform coverage of the area and the investigation of potentially elevated areas of activity identified during the scan. The number of measurement locations within each survey unit will be based in part on the number of grids in the area.

If activity is verified in excess of 75% of a guideline value, additional samples will be collected to determine the extent of the activity.

For areas ≤750 square meters	A minimum of 30 measurement locations is required
For areas >750 square meters	A minimum of 4 measurement locations are required for each 100 square meters being surveyed

The final survey of the liquid effluent pathway will consist of representative soil and sediment samples. The open land areas adjacent to the liquid effluent pathway will be surveyed in accordance with protocols established for unpaved surfaces within affected open land areas.

- **Unaffected Open Land Areas**

The final survey of unpaved surfaces in unaffected areas will include an exposure rate scan survey of approximately 10% of the accessible surface area and a minimum of 30 exposure rate measurements and surface soil samples. It should be noted that it may not be possible to scan or take exposure rate measurements at 25% of the applicable guideline value due to significant background interference.

In general these areas will not be gridded, however measurement locations will be marked. The measurement locations will ensure uniform coverage of the area and the investigation of potentially elevated areas of activity identified during the scan. The final survey of settling ponds will consist of gamma spectral analysis of representative sediment samples.

4.3.4 Survey Maps

Survey maps will be used to document the measurement locations. Figure 4.3, "Final Survey Unit Map," illustrates a typical survey map of a structural survey unit for the PCRV to be used in the absence of physical gridding. Maps will be prepared for specific survey units to identify structures, systems or equipment which define boundaries of survey units. Where grid layout cannot be performed and gridding is not practical or cost effective, the locator/survey maps will provide the means of reproducing survey measurement locations.

4.3.5 Survey Point Identification

Survey points will be identified by a unique reference location ID code or number. The numbering convention will allow the survey data to be easily referenced to survey points identified on maps, drawings, etc. An individual reference location will have a unique identification code as determined by its survey unit (or subunit) ID and the number of the survey points within that unit.

4.3.6 Surface Scans

Scanning surveys will be performed to screen large areas, to search for areas above the average release limits and to detect localized areas above the maximum release limit. The scanning methods used (instrument and survey technique) will be designed to detect 75% of the average total surface activity release limits. If an area of elevated activity is identified during the scan of a survey unit, the location will be marked and fixed point measurements for removable and total activity will be collected at that location in addition to the prescribed set of systematic measurements for the survey unit.

4.3.7 Surface Activity Measurements

Surface activity measurements will be taken at measurement locations and frequencies based upon the classification of affected or unaffected areas. Specific guidance regarding the location and number of measurements will be provided in the survey package. The general set of measurements will consist of direct (total) beta-gamma and removable beta-gamma at each measurement location. In areas and systems identified as alpha affected, direct (total) surface and removable surface alpha measurements will also be taken.

Direct measurement of surface activity will not be performed on interior surfaces of plant systems where the results of decommissioning surveys indicate only the presence of HTDN. The evaluation of these surfaces will be based upon the results of indirect measurements.

When surveys indicate that activity levels above the average contamination release limits may be present, appropriate follow-up investigation and/or measurements will be performed. Areas of elevated activity within buildings or structures will be tested to assure that the average surface activity level within a contiguous square meter containing the elevated area is less than the release limit.

4.3.8 Exposure Rate Measurements

Exposure rate measurements will be obtained at one meter from accessible surfaces and compared to the background exposure rate. Individual exposure rate measurements which exceed the background exposure rate by more than 5 μR/hr will require further investigation to establish the average and maximum exposure rates in the area. If these exposure rates exceed the guideline values, and the cause is determined to be the result of structural configurations, a notation will be made in the survey record. If these exposure rates exceed the guideline values, and the cause is determined to be the result of licensed material, remediation will be required.

4.3.9 Soil and Water Sampling

Soil and water samples will be collected in accordance with a final survey procedure. Affected areas, such as the liquid effluent pathways, will normally be gridded into 10 meter by 10 meter, or 5 meter by 5 meter grids. Soil samples collected within each 10 meter by 10 meter grid will be from locations equidistant between the center and each of the 4 corners. Within each 10 meter by 10 meter grid, 4 soil samples will be collected. Soil samples collected within each 5 meter by 5 meter grid will be from the center of each grid. Within each 5 meter by 5 meter grid, 1 soil sample will be collected. For unaffected areas, a total of 30 random soil and water samples will be collected. Specific guidance regarding the location and number of soil and water samples will be provided in the survey packages. Additional soil and water samples will be collected if a contamination

event or spill occurs, or if survey measurements indicate areas of elevated activity from licensed material.

4.3.10 Special Sampling and Measurements

a. Sampling of Sediment and Loose Material

Samples of loose paint, dust or other sediment will be collected for analysis as part of biased sampling and measurements. Such samples may be collected in drain receptacles, sumps, and other catchments in affected areas. All storm drain catchments will be sampled and surveyed. These samples will be analyzed for principal gamma emitting radionuclides.

b. Embedded Piping Surveys

Specialized measurements will be made to demonstrate that normally inaccessible piping (e.g., embedded piping) is below the release limits for surface activity. This will include but not be limited to the use of calibrated detectors extended into piping runs in a controlled manner and the use of data acquisition equipment to document the measurements.

For embedded piping, measurement of total and removable surface activity will be performed at biased locations. The selection criteria for measurement locations are: (1) locations where residual activity would be most probable (e.g., drains, low points, heat exchange surfaces), and (2) portions of the plant system which are reasonably accessible.

The detectors used for direct measurement of total surface activity within embedded piping include cylindrical gas-flow proportional detectors ranging in diameters of 0.5" to 1.5"; 125 cm^2 gas-flow proportional detectors; 15.5 cm^2 Geiger-Mueller detectors; and other detector types and sizes as situations indicate. The detectors are fitted with a delivery device to provide a consistent and reproducible calibration/measurement geometry, and to protect the detector from damage or potential contamination.

The method used to assess removable surface activity within embedded piping consists primarily of smear media positioned using a remote handling tool. For inaccessible portions, foam rubber balls or masslin towels may be passed through the system piping to provide a qualitative assessment of removable activity.

To support the conclusion that sections of embedded piping meet the criteria of release for unrestricted use, surface activity measurements for total and removable surface activity will typically be performed prior to, during, and after decontamination processes to evaluate the effectiveness of the decontamination technique(s). The decontamination processes are controlled

to ensure that all portions receive similar treatment by the decontamination process. Based upon the effectiveness of the decontamination process, as defined by the measurement results obtained, the radiological condition of any inaccessible portion of the embedded piping can be inferred.

When surveys performed during dismantlement indicate the presence of only HTDN (e.g., H-3), indirect measurements will be used to assess radiological status and suitability for unrestricted use.

4.3.11 Sampling for Hard To Detect Nuclides

Samples will be taken and analyzed from random survey locations in affected areas of the facility, such as portions of the liquid effluent pathway, reactor building, radwaste areas, plant system interior surfaces and the remaining PCRV structure. The ratios of HTDN to readily detected radionuclides will be compared to previously established ratios used to establish site specific limits which account for HTDN. If these new ratios are determined to be non-conservative, new site specific guideline values will be calculated in accordance with Section 3.3.

4.4 Background Level Determination

4.4.1 General Requirements

Background response will be established for each type of instrument to be used for surface activity and exposure rate measurements. Exposure rate measurements require determination of the background response of detectors at one meter from surfaces. The relative background responses of the pressurized ion chamber and micro-R instruments will be determined. In addition, backgrounds will be determined for specialized detectors and detector systems. These include: large area detectors and detectors for in-situ monitoring (GM, Gas-flow Proportional, and NaI (Tl) scintillator).

Background values for the various materials of building construction will be established for surface activity and exposure rate measurements. Additional consideration will be given to the various factors which influence background in a specific survey unit, such as structural configuration and the location of the measurement, i.e. concrete walls with concrete floor, concrete walls with metal grate floor, etc.

4.4.2 Objectives of Background Determination

The objectives of background determinations for FSV final survey measurements are to:

- Assure reliable instrument operation;

- Establish the reference background values for each type of instrument-detector to be used in the survey;
- Define the contribution from natural background and assess the variability in background responses for principal detectors under different applications and conditions of use;
- Assess the variability in background due to materials of construction, measurement location and area configuration; and
- Determine the need for correction factors or special measurements to establish the background value for final survey measurements.

4.4.3 Background Measurements

Locations will be identified as needed, to acquire background measurement data for each type of measurement. Collection of background data will be performed in accordance with approved procedures appropriate for the instrumentation used. Background determinations for each type of final survey measurement will be performed as described below.

a. Direct Surface Beta-Gamma Measurements

To determine the natural material background component for direct surface beta-gamma measurement, data will be collected from areas exhibiting similar characteristics but unaffected by licensed activities. The number of background locations is dependent upon the survey unit size and complexity. The counts will be accumulated by a scaler in the preset time accumulation mode. To determine the background component due to local area exposure rate, measurements will be made in the vicinity of the survey unit using a beta absorber sufficient to stop all survey unit beta radiation from being measured.

b. Direct Surface Alpha Measurements

If it is necessary to correct for direct surface alpha background, similar protocols to those used to determine the direct surface beta-gamma background are used. Special counting techniques may be required to assess the influence of naturally occurring radionuclides. Background counting times may be accumulated and mean backgrounds used because of the low numbers of counts observed. If radon is a problem, the detector will be equilibrated to the radon atmosphere before doing the alpha background and survey.

c. Removable Surface Beta-Gamma Measurements

Background determinations of beta-gamma smear counters will be made by taking measurements of blank smear media or a series of smears collected from an area verified free of licensed material.

d. Removable Surface Alpha Measurements

Background determination for removable surface alpha measurement uses the same protocols as that for removable surface beta-gamma background.

e. Soil and Water Activity Measurements

Soil and water samples will be collected from the environs within areas unaffected by licensed operations in order to establish the environmental background for activity in soil and water. The samples collected will include soil samples from open land areas, and water and stream sediment samples from upstream locations in the South Platte and St. Vrain rivers.

f. Exposure Rate Measurements

Outdoor exposure rate reference values have been previously established and reported within the 1989 Radiological Environmental Monitoring Program Report (15.8 μR/hr) (Ref. 15), the 1990 FSV Site Characterization Study conducted by Colorado State University (13.3 μR/hr) (Ref. 16), the 1990 Colorado State University ISFSI Site Background Radiation Study (14.2 μR/hr), the 1990 Radiological Environmental Monitoring Report (16.25 μR/hr), and the 1991 Fort St. Vrain Initial Radiological Site Characterization Report (15-20 μR/hr). These values will be either updated or confirmed by the outdoor background exposure rate determination for final surveys.

Experience with previous final survey projects has shown that the above values are generally inappropriate for use as indoor background exposure rates. For indoor surveys, there is a large and variable influence on the local background due to the geometry of the natural radioactive materials (e.g., K-40) in the walls and floor of the area being surveyed. In addition, the materials of construction can greatly influence the local background by shielding indoor areas from the outdoor background.

Therefore, to demonstrate that the exposure rate due to the presence of licensed materials in indoor structural areas does not exceed the limit of 5 μR/hr when averaged over $10m^2$, it is necessary to establish the local background in the area of interest. The background exposure rate for a given survey unit will be based on the exposure rates from areas exhibiting similar characteristics but unaffected by licensed activities. The same or similar areas that were used for background evaluation during the initial site

characterization will be used, such as a concrete grain elevator, an old Atlas Missile silo, and on-site locations that were unaffected by plant operations.

Locations will be chosen for establishing background to ensure representativeness in the areas being surveyed. As needed, methodology will be developed to address high variability or exposure rates that are statistically below the background levels established. It is anticipated that measured exposure rates below grade in the Reactor Building may be negative when compared to the assigned background value. In these cases, the background value will be adjusted downward based on a statistical analysis (mean and variation) of the measured data. In-situ gamma spectroscopy may also be used as needed to ensure the proper assignment of background values.

For surfaces and volumes having a potential for neutron activation, in-situ gamma spectroscopy may be performed to determine the contribution to the total exposure rate from licensed materials.

g. Specialized Measurements

Detector background is affected when detectors are inserted inside massive components or within piping embedded in concrete. This is particularly noticeable when NaI(Tl) detectors are used. Thus it will be necessary either to provide a mockup of an embedded pipe, for example, or to develop empirical correction factors for backgrounds when surveying such equipment. To this end, a series of background measurements will be performed in embedded piping and in large components which can be ascertained to be free of radioactive contamination.

h. Verification of Background Measurement Population

For background soil and water samples where the activity for nuclides of interest exceeds 10% of the release limit at the 95% confidence level, the population of background measurement will be tested to ensure that the number of measurements in the data set is adequate to support the population statistics. The total number of background measurements required will be calculated as shown below:

$$n_i = \left[\frac{(t_{95\%,df})\ s}{a\ (\bar{x})} \right]^2 \tag{4.2}$$

where:

n_i = number of data points required

$t_{95\%,df}$ = t statistic for 95% confidence at df = n-1 degrees of freedom

df = n-1 degrees of freedom where n is the number of initial data points

s = standard deviation of initial measurements

a = variable depending upon background variation, this value is typically 0.2

$\bar{x}$ = mean of mutual determination when greater than, or equal to 10% of the release limit

4.4.4 Documentation And Control of Background Measurements

Background measurements will be collected and recorded in accordance with a final survey procedure.

4.5 Sample Analysis

As indicated in Section 3.7, an in-depth sample analysis capability is available on-site to support final surveys. Routine samples of sediment, paint chips and debris will be evaluated using gamma spectroscopy. The need for additional sampling and analysis will be determined on the basis of this initial evaluation.

FIGURE 4.1

Survey Unit Classification Process

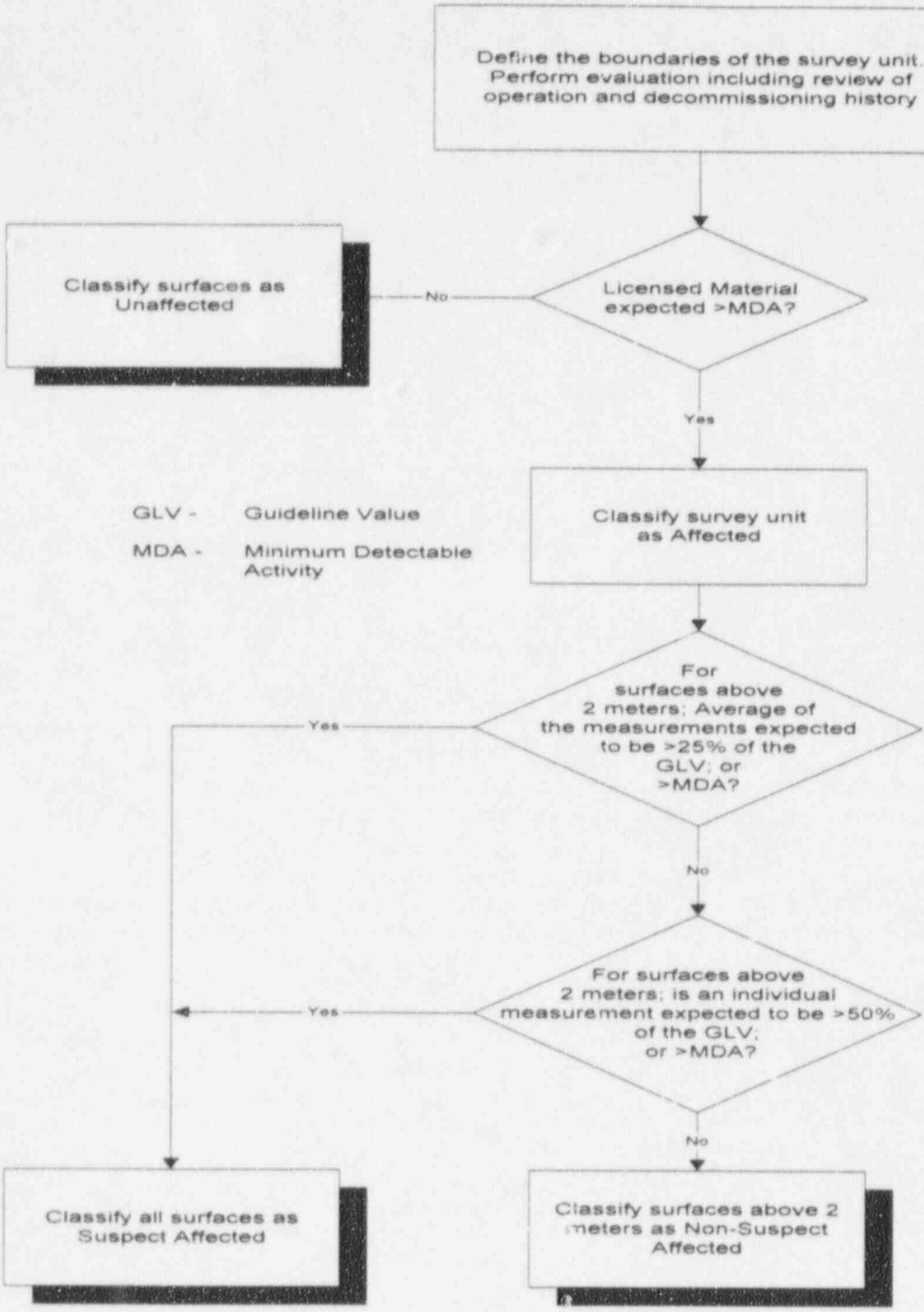


FIGURE 4.2

Reference Grid Layout

SURVEY UNIT LOCATION CODE ______________________

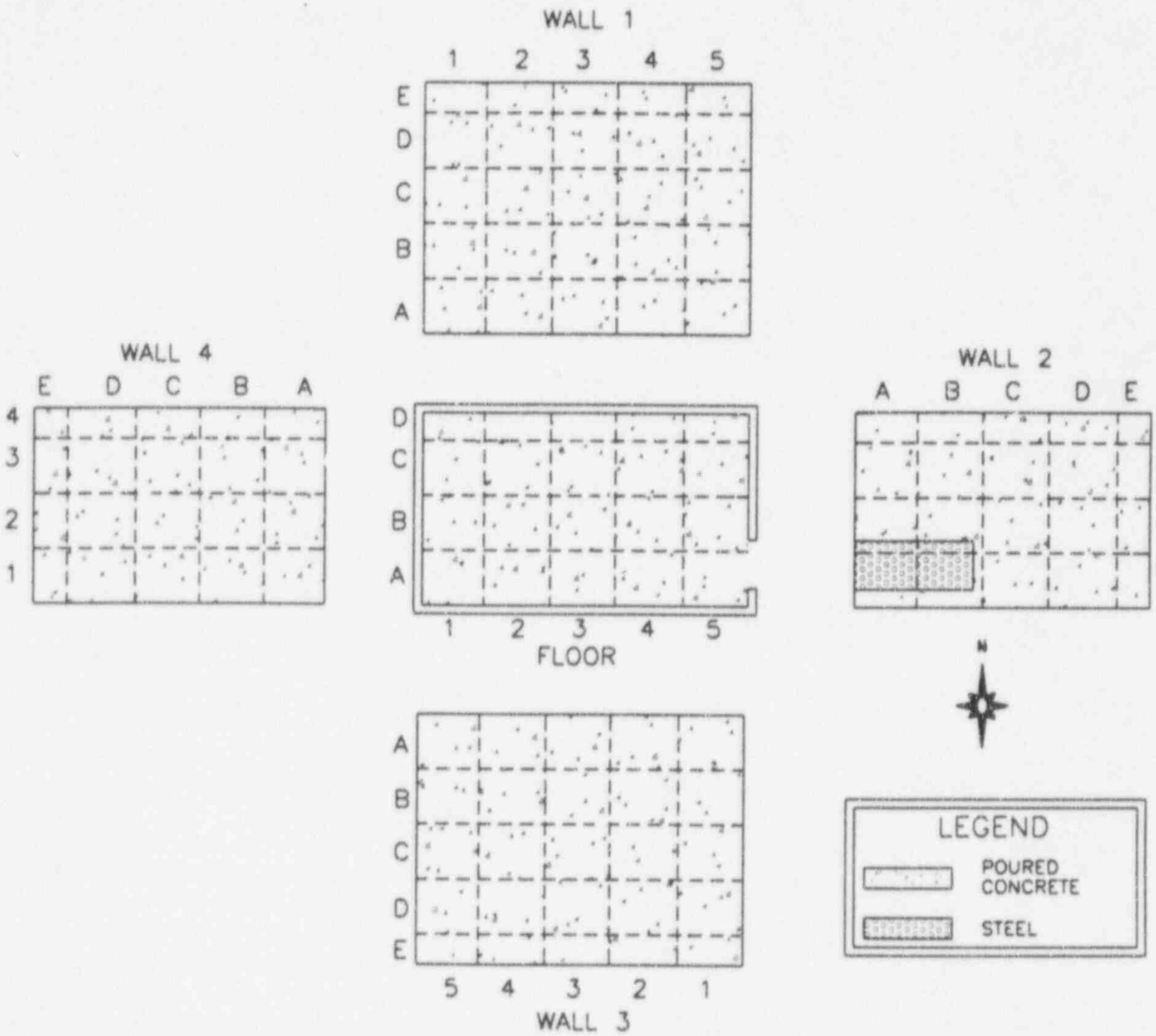
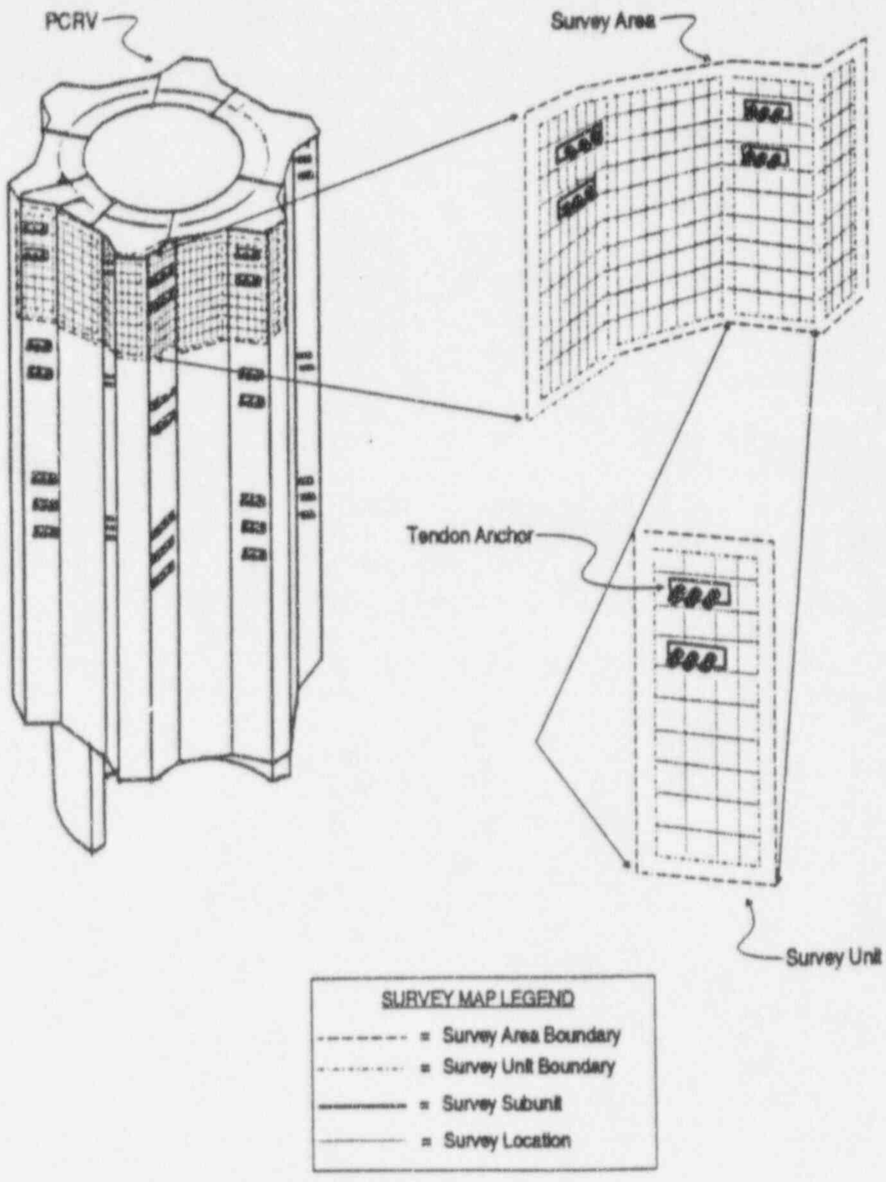


FIGURE 4.3

Final Survey Unit Map (Example)



5.0 DATA INTERPRETATION

All measurements will be reported in units appropriate for comparison with the Table 3.1 surface contamination limits and the exposure rate criteria. Total and removable surface activity measurements will be reported in units of dpm per 100 cm^2. Exposure rate measurements will be reported in units of μR/hr. Concentrations in soil and water will be reported in units of pCi/g and pCi/ml, respectively.

The background values associated with the final survey program will be continuously evaluated. For example, the result of assigning a background value which was inappropriately high could result in the majority of the measurements within a given survey unit being negative. Net measurement results skewed low will be evident when reviewing the graphical presentation and summary statistics for each survey unit. Investigations will be documented and the results included in the final survey report.

The upper bounds of the mean at the 95% confidence level will be calculated for each type of measurement performed within a given survey unit and compared to the Table 3.1 surface contamination limits, the exposure rate criteria, and the concentration guideline values for soil and water. This comparison will determine if a survey unit is acceptable for release for unrestricted use. These calculations will be specific to a given survey unit and will be presented along with other relevant statistics for the survey unit.

Figures 5.1 through 5.6 indicate the typical decision process and logic to calculate, analyze and interpret the data for the final surveys.

5.1 Conversion of Measurements to Reporting Units

5.1.1 Direct Measurements - Total Surface Activity

Measurements of total surface activity will be converted from observed gross counts per minute to net dpm/100 cm^2. By subtracting the background counting rate for the instrument and correcting the net count rate for geometry, efficiency, and detector surface area the results in dpm/100 cm^2 units are obtained.

If large area detectors are used, the acceptable observed activity will be limited to that activity which would be acceptable when confined to an area of 100cm^2. Since it cannot be discerned that the observed activity is uniformly distributed, it will be assumed that the observed activity could be attributed to an area of 100cm^2 or less. In the event that activity is observed when using a large area detector which is in excess of what would be acceptable for an area of 100cm^2, an investigation survey will be performed to demonstrate compliance with the average and maximum site-specific guideline values.

Total surface activity measurement results will be reviewed to ensure that the applied background values are appropriate. If any background values are

determined to be inappropriate they will be adjusted as necessary. All adjustments, and the justification for such adjustment, will be documented to ensure traceability.

5.1.2 Removable Contamination Measurements

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

5.1.3 Exposure Rate Measurements

Exposure rate data will normally be collected using NaI(Tl) or similar micro-R meters. Results from these meters will be compared to the response of a Reuter-Stokes pressurized ion chamber (PIC). The PIC is an industry recognized, standard quality device which measures exposure rate at environmental levels with negligible energy dependence. The PIC will be used to characterize the response of the micro-R meters relative to the true exposure rate. A correction coefficient will be calculated and applied to each micro-R meter to correct the reading to the PIC value.

5.1.4 Soil and Water Activity Measurements

Soil and water samples will be collected in accordance with approved procedures as discussed in Section 4.3.9. Laboratory analysis will be performed to identify the presence of licensed materials. The analysis results for individual samples will be compared to the background levels as determined in Section 4.4.3.e. The concentrations of licensed materials in soil and water above background levels will be reported in units appropriate for comparison to the site guideline values.

5.1.5 Hard to Detect Nuclide Measurements

Sampling of surfaces, structures and systems for HTDN will be performed as discussed in Section 4.3.11. The concentration of HTDN will be determined through laboratory analysis of these samples.

5.2 Comparison With Release Limits

Individual measurement results for total activity and exposure rate will be compared against the average and the maximum guideline values. Measurement results less than the average guideline value will be deemed acceptable. Measurement results greater than the maximum guideline value will indicate the need for remediation. Measurement results greater than the average guideline value but less than the maximum guideline value will require investigation to determine if the average of a series of measurements

collected from one square meter for total activity, to 10 square meters for exposure rate, exceeds the average guideline value.

The method outlined below will be used to demonstrate attainment of the release limits (Table 3.1 surface contamination limits and exposure rate criteria).

5.2.1 Attainment of Release Limits for Surface Activity

a. Total Surface Activity (fixed plus removable contamination):

Individual measurements:	Not to exceed 15,000 dpm/100 cm^2 or the site-specific guideline value (surface area not to exceed 100 cm^2).
Average measurements:	Not to exceed 5000 dpm/100 cm^2 or the site-specific guideline value when averaged over an area not to exceed 1 m^2.

b. Removable Surface Activity:

Individual measurements:	Not to exceed 1000 dpm/100 cm^2 or the site-specific guideline value.

5.2.2 Attainment of Exposure Rate Limit

Individual measurements:	Exposure rate does not exceed 10 μR/hr above background .
Average measurement - structures and surfaces	Exposure rate does not exceed 5 μR/hr above background when averaged over an area not to exceed 10 m^2.
Average measurement - open land	Exposure rate does not exceed 5 μR/hr above background when averaged over an area not to exceed 100 m^2.

5.2.3 Attainment of Soil and Water Limits

For residual radioactivity in soil and water, the guideline values are the nuclide-specific concentrations which could result in an annual total effective dose equivalent (TEDE) of 10 mrem to an average individual in a population group exposed to radioactive material after decommissioning. The TEDE for concentrations of licensed material in soil and water will be determined in accordance with the methodology contained in NUREG/CR-5512, Volume 1, or an equivalent methodology such as RESRAD to assure that the annual TEDE will not exceed 10 mrem. Where multiple nuclides are present, the individual nuclide concentrations will be compared to their respective guideline values using the unity equation. For these instances, the sum of the nuclide concentration-to-guideline value ratios will not exceed 1.

Elevated area guidelines will be defined as the value which individual measurements may not exceed under any conditions. Areas having elevated activity will be tested to ensure that the sum of the concentration-to-guideline value ratios for individual samples does not exceed three; and that the average nuclide concentration-to-guideline value ratio within the area does not exceed $(100/A)^{1/2}$ where A is the area of elevated activity in square meters.

If the above conditions are satisfied, the weighted mean for the $100m^2$ containing the area of elevated activity will be tested in accordance with Draft NUREG/CR-5849, Equation 8-10 to ensure the weighted mean does not exceed the guideline value of 10 mrem/yr (TEDE) as follows:

$$\overline{x_w} = \frac{1}{n_s} \sum_{i=1}^{n_s} x_i \left[1 - \sum_{k=1}^{n_k} A_k \right] + \sum_{k=1}^{n_k} y_k A_k \qquad (5.1)$$

Where:

$\bar{x}_w$ = weighted mean including elevated area(s),

n_s = number of systematic and random measurements,

x_i = systematic and random measurements at point *i*,

n_k = number of elevated areas,

A_k = fraction of $100m^2$ occupied by elevated area *k*,

y_k = area of elevated activity in area *k*.

5.2.4 Attainment of HTDN Limits

A limited number of samples will be collected from facility surfaces, structures and systems to determine the relative fractions of HTDN. These fractions will be compared to those used to establish the reduced limits which account for the presence of HTDN in affected areas. In the event that the relative fraction of HTDN is found to differ in a non-conservative manner from that which has been previously established, the guideline value will be adjusted for the related areas of the facility.

5.2.5 Calculation of the 95% Confidence Level of the Mean

The 95% confidence level will be used to further demonstrate attainment of the release limits once the individual measurements have demonstrated compliance with the guideline value. The confidence interval is calculated using Normal statistics (one-tailed test) at the 95% confidence level.

$$U_a = \bar{x} + t_{1-\alpha,df} \frac{s_x}{\sqrt{n}} \tag{5.2}$$

Where:

- U_a = upper confidence limit of sample mean,
- $\bar{x}$ = sample mean,
- $t_{1-\alpha,\, df}$ = student *t* statistic for the degree of confidence and degrees of freedom; *df* (degrees of freedom) is equal to n - 1; and α is 0.05 for this test,
- S_x = sample standard deviation,
- n = number of measurements in the survey unit.

5.2.6 Calculation of the Critical Level

The critical level is the net activity (total minus background) for reaching a decision of nondetection if the net activity is less than the critical level, and the decision of detection if the net activity is greater than the critical level. The critical level is determined for reporting purposes, however the critical level is not compared to the guideline value for demonstrating suitability for release for unrestricted use. For surface activity measurements, the critical level will be determined as follows:

$$L_c = \frac{1.645 \sqrt{\frac{R_b}{t_s} + \frac{R_b}{t_b}}}{E*(a/100)} \tag{5.3}$$

Where:

L_c = minimum amount of activity that can be statistically detected above background with a 95% probability (dpm/100 cm^2),

R_b = background count rate in counts per minute (cpm),

t_s = sample counting time (minutes),

t_b = background counting time (minutes),

E = counting efficiency, (cpm/dpm), and

a = detector area, or the area sampled for smear samples (cm^2).

FIGURE 5.1

Total Surface Contamination Data Processing
Unaffected Survey Units

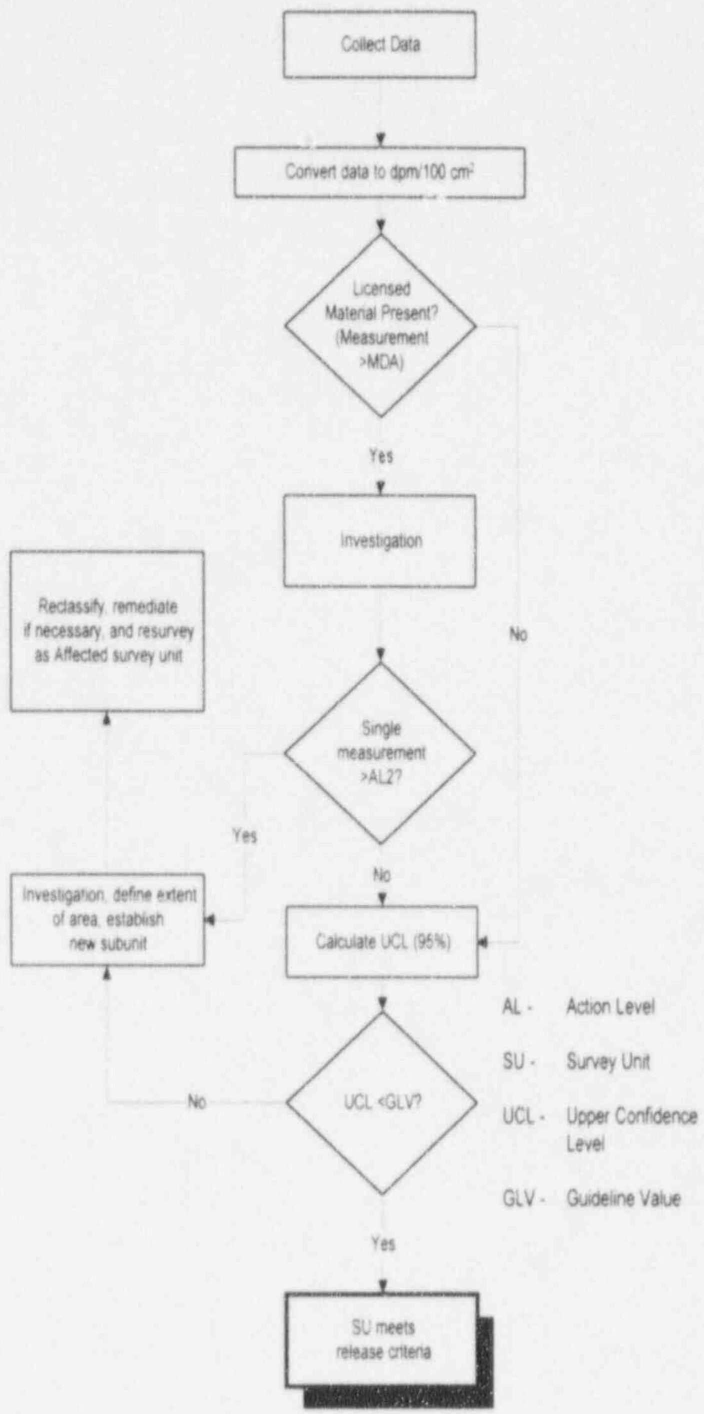


FIGURE 5.2

Removable Surface Contamination Data Processing
Unaffected Survey Units

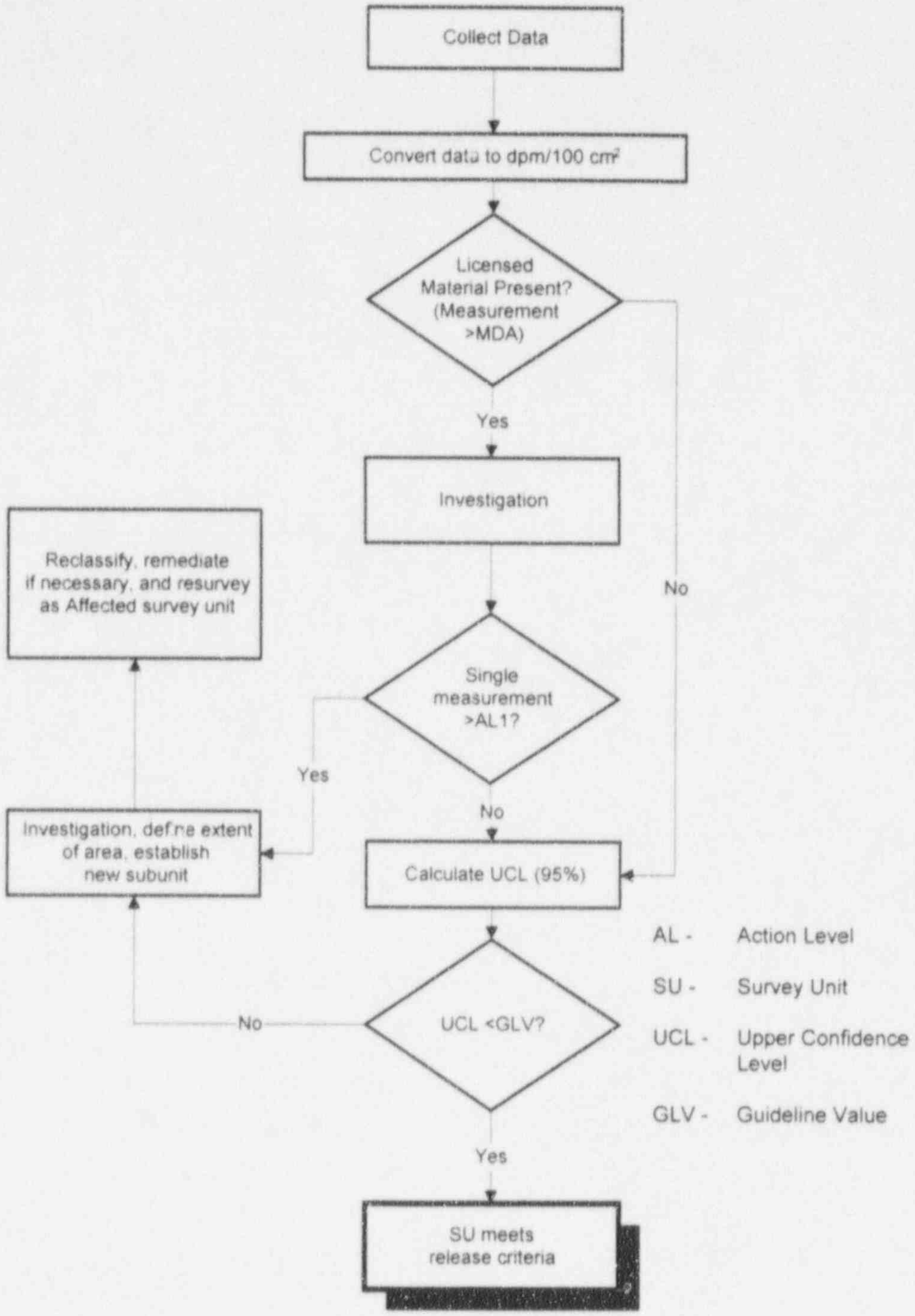


FIGURE 5.3

**Total Surface Contamination Data Processing
Non-Suspect Affected Survey Units**

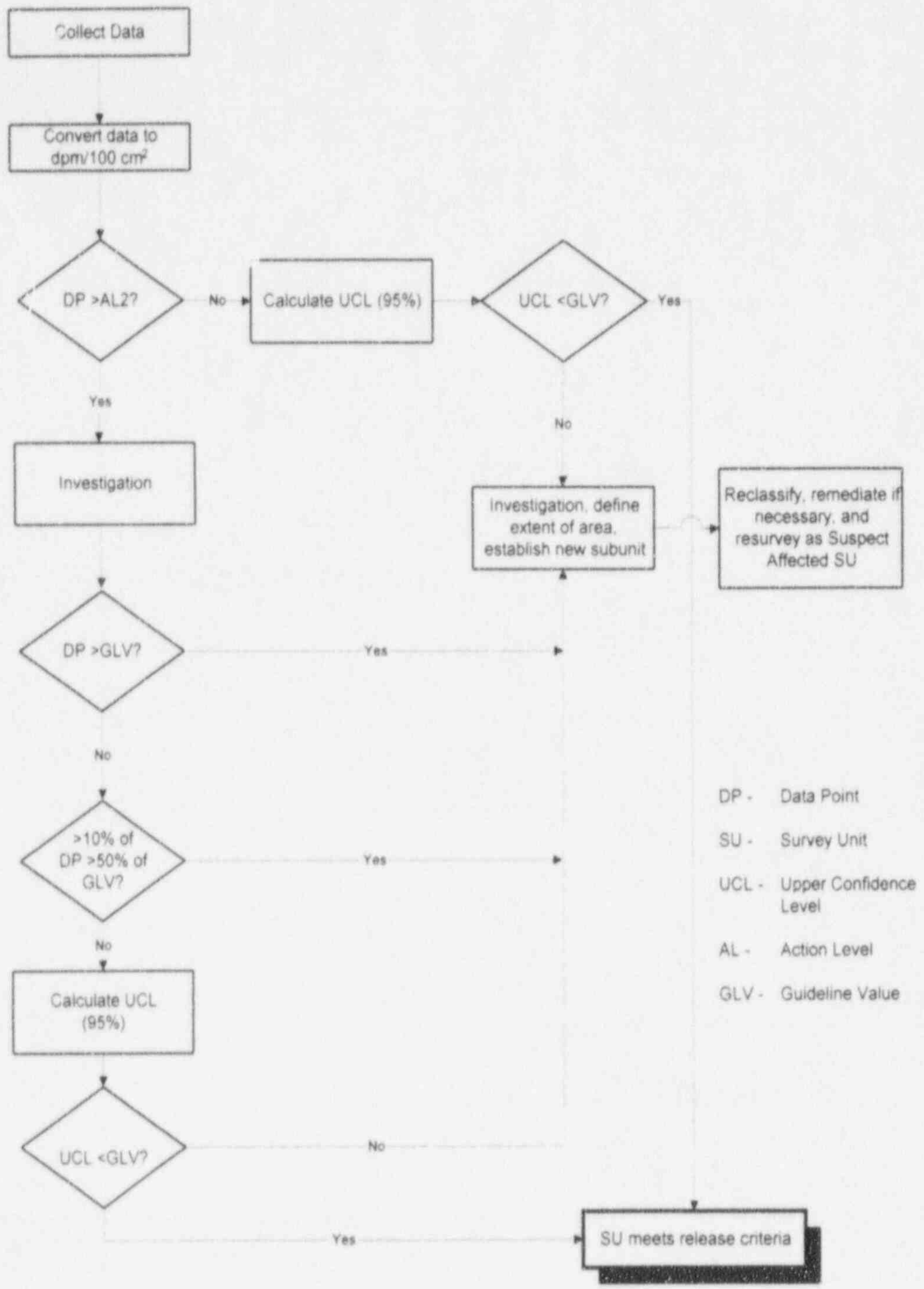


FIGURE 5.4

Removable Surface Contamination Data Processing
Non-Suspect Affected Survey Units

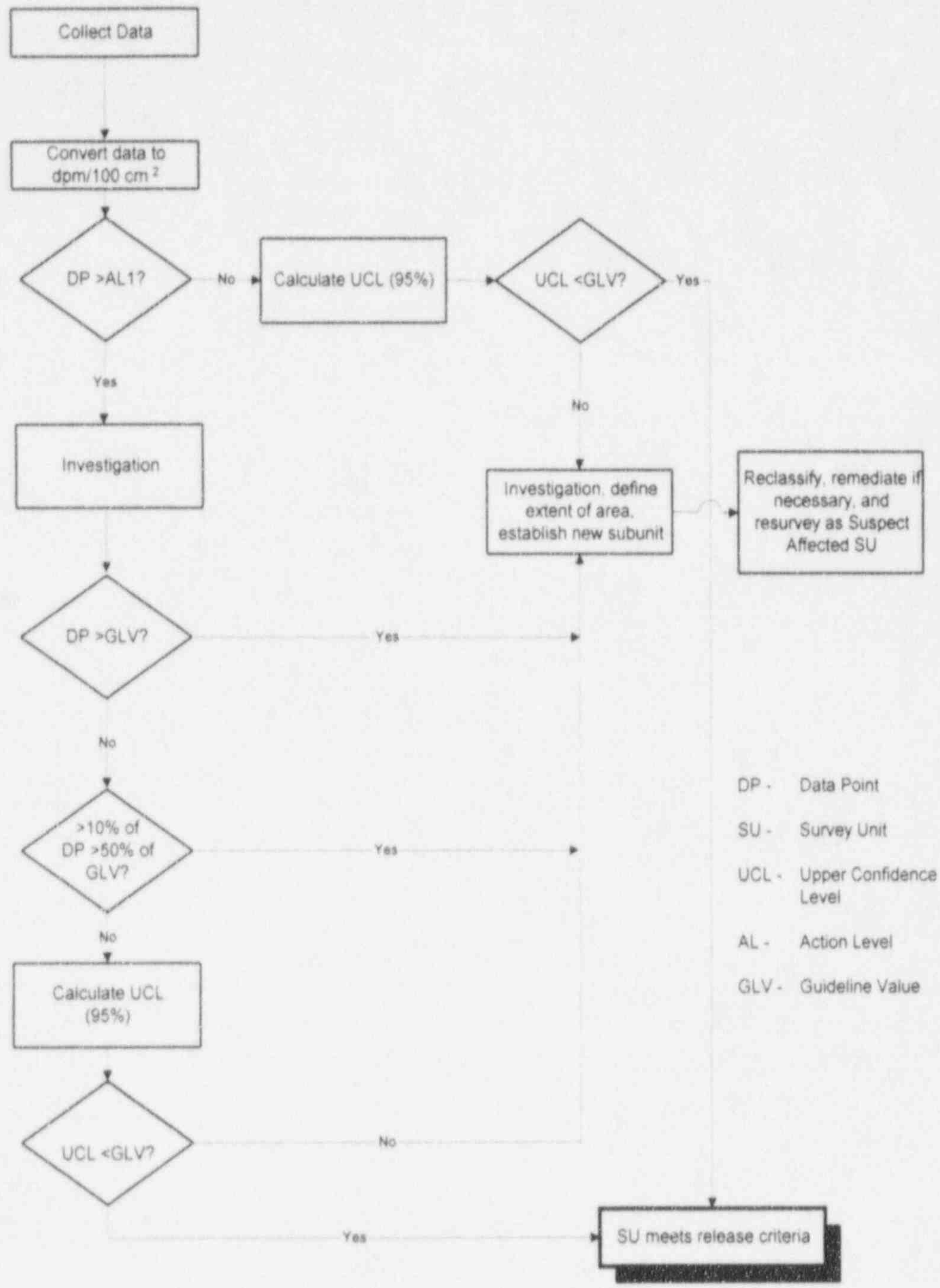


FIGURE 5.5

Total Surface Contamination Data Processing
Suspect Affected Survey Units

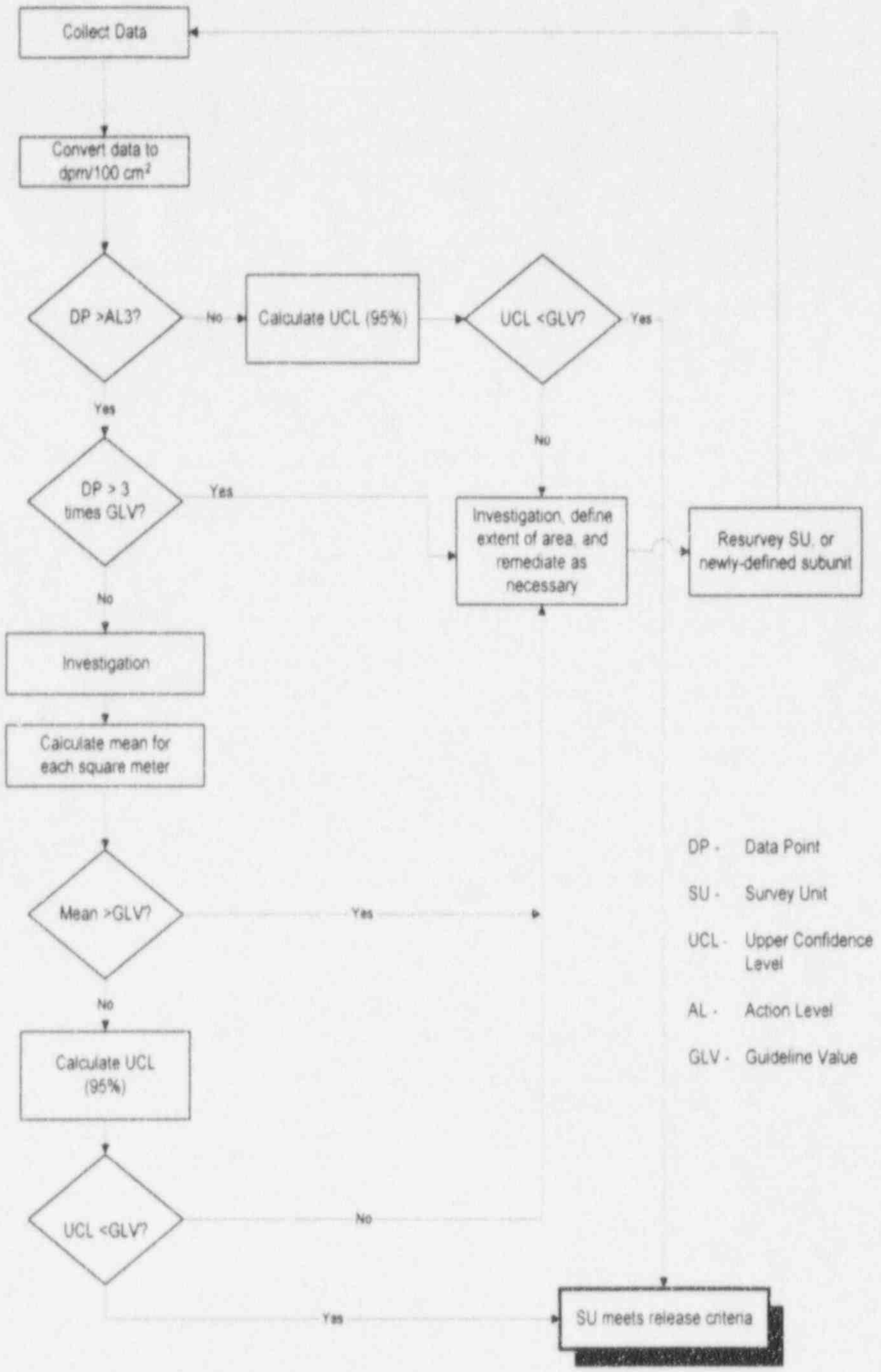


FIGURE 5.6

Removable Surface Contamination Data Processing Suspect Affected Survey Units

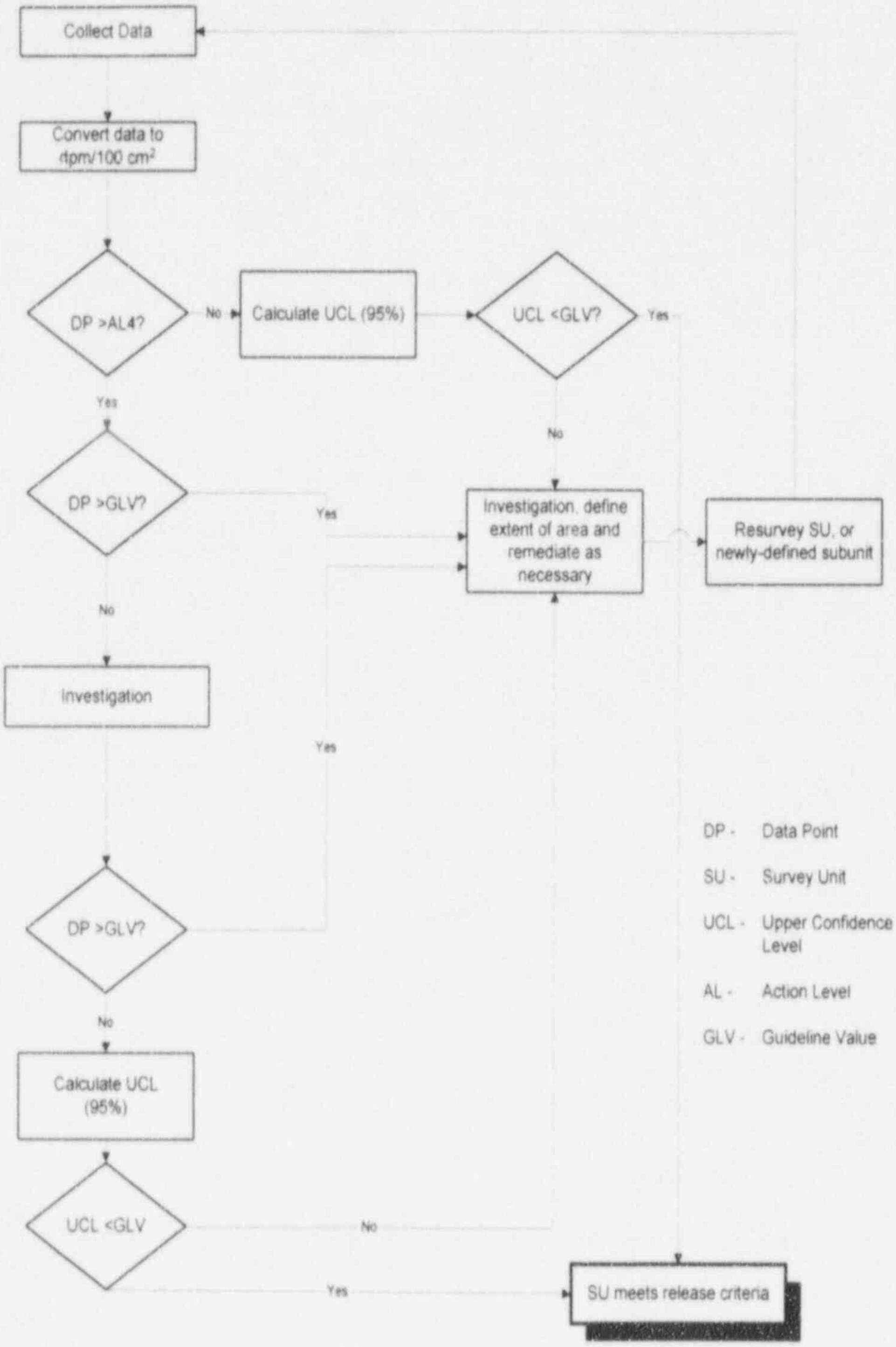


FIGURE 5.7

Exposure Rate Data Processing

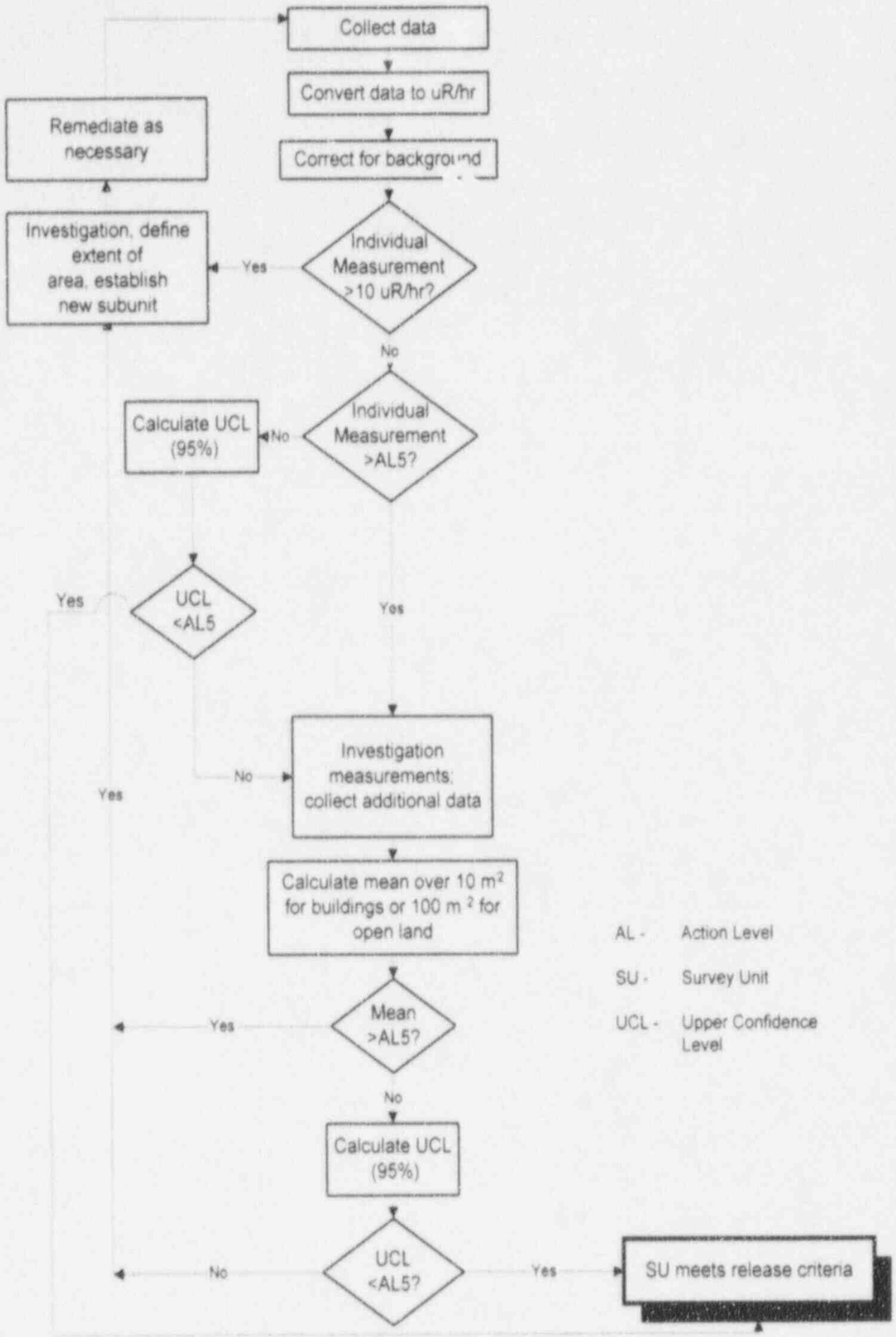
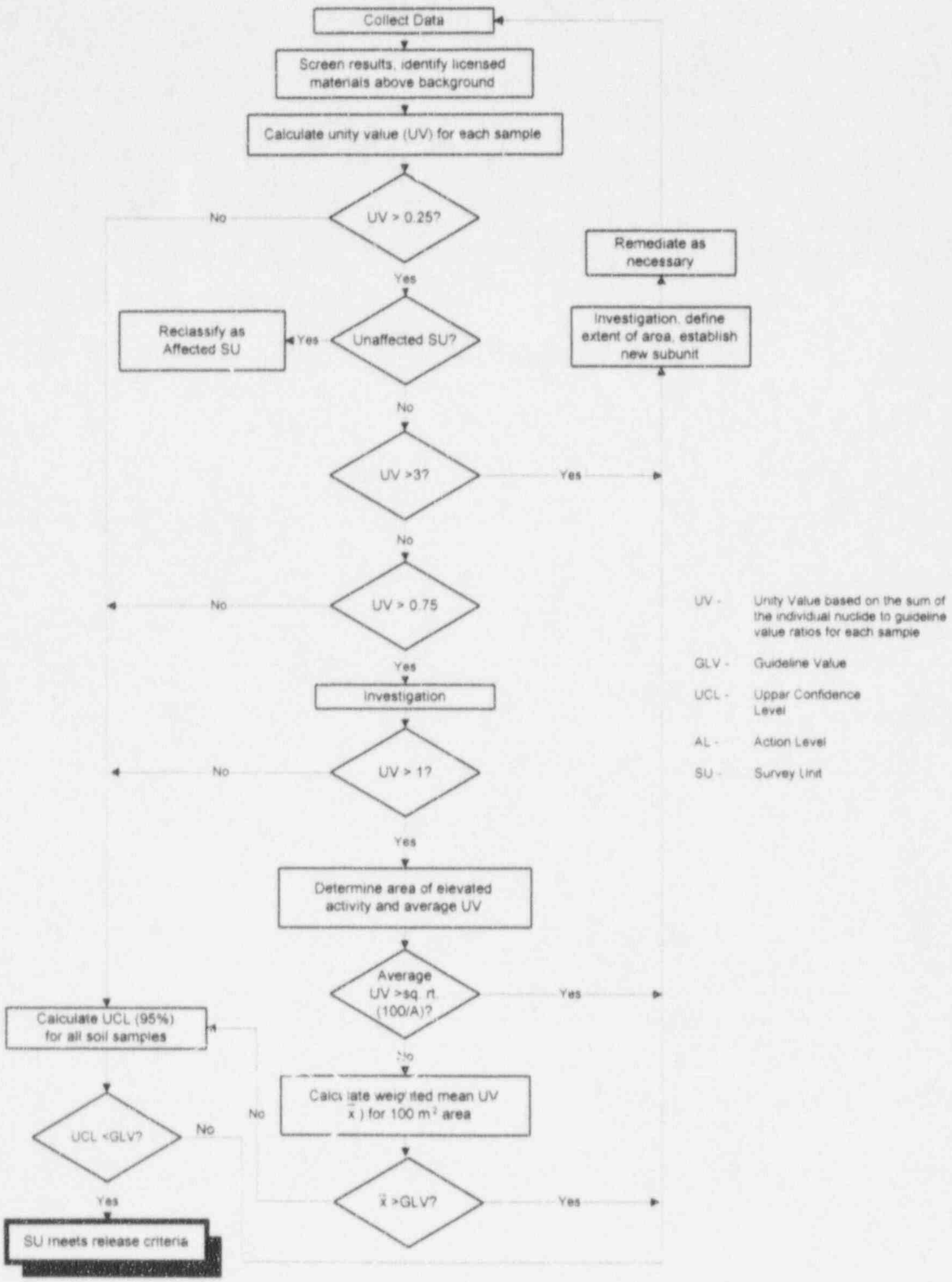


FIGURE 5.8

Soil Sample Data Processing



6.0 FINAL SURVEY REPORTING

Upon completion of each major Final Survey phase an interim report will be prepared for review by the Nuclear Regulatory Commission. This report will meet the intent of Regulatory Guide 1.86 for Final Survey reporting and will:

- identify the premises
- Demonstrate that reasonable efforts have been made to reduce residual contamination to ALARA levels
- Describe the scope of the Final Survey and general procedures followed
- Present the results of surveys in terms of the applicable guideline values

Upon completion of all Final Survey phases, a Final Report will be prepared which addresses the topics outlined below. The report will provide adequate data and discussion of each topic to meet the intent of Draft NUREG/CR-5849.

6.1 Topical Outline

The following outline illustrates a general format for the Final Report regarding document volumes, topical outline and content. The outline below may be adjusted to provide a clear presentation of the information. The level of detail will be sufficient to clearly describe the Final Survey Program and certify the results. The applicable nuclides will be identified. The report will demonstrate that the instrumentation selected was appropriate to reliably measure radioactivity at the guideline values. The report will be able to conclude that residual activity is below the applicable guideline values.

1.0 Background Information

- 1.1 Reason for Decommissioning
- 1.2 Management Approach

2.0 Site Description

- 2.1 Type and Location of Facility
- 2.2 Ownership
- 2.3 Facility Description
  - Facility Grounds
  - Facility Structures
  - Facility Systems

3.0 Operating History

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

4.0 Decommissioning Activities

- 4.1 Objectives
- 4.2 Results of Previous Surveys
  - Site Characterization
  - Radiological Environmental Monitoring
  - Radiological Effluent Reports
- 4.3 Decontamination and Dismantlement Activities
- 4.4 Decommissioning Processes
  - Decommissioning Contractor
  - Demolition and Dismantling
  - Shipping, Storage and Disposal of Materials
  - Security Precautions and Safeguards

5.0 Final Survey Methodology

- 5.1 Sampling Parameters
- 5.2 Background Levels
- 5.3 Major Contaminants Identified
- 5.4 Guidelines Established
- 5.5 Equipment and Techniques Applied
  - Survey Instruments and Equipment
  - Detection Sensitivity
- 5.6 Survey Process
  - Survey Unit Classification
  - Reference Locators
  - Surface Scans
  - Surface Activity Measurements
  - Exposure Rate Measurements
  - Soil, Water and Sediment Sampling
  - Special Sampling
- 5.7 Survey Control
  - Quality Control and Quality Assurance
  - Personnel Training and Qualification
  - Instrument Calibrations
  - Custody of Data
  - Data Verification
  - Survey Records
- 5.8 Data Analysis and Statistical Evaluation

6.0 Final Survey Results

- 6.1 Data Results and Interpretation
  - Data Reduction and Review
  - Statistical Evaluation
- 6.2 Comparisons to Release Criteria
- 6.3 Comparisons to Independent Third-Party Survey

7.0 Summary

APPENDIX I - Survey Unit Release Records

1. Verification Surveys
2. Survey Maps

APPENDIX II - Westinghouse Team Quality Assurance Documentation

1. Qualification Records
2. Quality Assurance Surveillances

6.2 Reporting of Survey Findings

6.2.1 Summary

A summary of the measurement results and overall conclusions showing that the facility meets the release criteria will be provided. As applicable, a tabular data summary will present the results for each major category of survey unit such as: structures, open land areas and plant systems. This tabulation will identify the number of survey units, the number and type of measurements such as: total surface beta-gamma, total surface alpha, removable surface beta-gamma and removable surface alpha activity concentration, and gamma exposure rate. For surface contamination, exposure rate and concentrations in soil and water, the average and maximum values, and upper the limit of the confidence interval about the mean will be reported for comparison to the release criteria. Typically, these results will also be illustrated in a graphical presentation to illustrate the individual data points and the statistical distribution of the results.

6.2.2 Summary Data Reporting for Each Survey Unit

Within the release record for each survey unit (and/or subunit), the number of measurements and the applicable statistical distribution will be presented in graph form. These will be reported in units of dpm/100 cm^2 for each type of surface activity measurement: total surface beta-gamma, total surface alpha, removable surface beta-gamma and removable surface alpha activity concentration. Exposure rate measurements will be reported in units of μR/hr, and soil and water activity in units of pCi/g or pCi/ml, respectively.

The applicable results of special sampling measurements, e.g., sediment, paint, concrete and other debris will be reported in the release record for each survey unit.

6.2.3 Detailed Data Reporting

All measurement results used to demonstrate that the site meets the guideline values for unrestricted use will be presented in the final report. Typically, these measurement results will be presented graphically for each survey unit with summary statistics provided. All measurement results, both positive and negative will be included in the statistical evaluation. The critical level will be included as a part of the summary statistics, or indicated as a horizontal line imposed on the graphs with the measurement set being displayed. This graphical presentation will facilitate rapid review to identify measurement values above or below the critical level.

Removable surface activity measurements greater than the critical level will also be indicated in the final survey report.

The data presentation for soil and water activity concentrations may not reference the critical level. Alternatively for each sample, activity concentrations due to licensed material in excess of the MDA will be presented along with the corresponding fraction of the site-specific guideline value in a tabular format.

If a survey unit requires remediation or reclassification, the results of the initial fixed point measurements will be archived as characterization survey data and the survey unit, or newly-defined subunit, will be resurveyed. The results of resurvey will be considered as the final survey of record and will be included in the calculation of the mean and standard deviation for the survey unit to determine compliance with the unrestricted use criteria. For instances where resurvey is performed, the final report will be annotated indicating that the initial data for a survey unit has either been replaced or supplemented with subsequent survey data.

The results of final survey measurements within a survey unit that do not pass the statistical analysis (95% confidence level of the mean) will be included in the final survey report. An investigation survey will be performed. The results of the investigation survey that indicate, after statistical testing, that the survey unit meets the release criteria will be contained in the final survey report along with the results of the initial final survey measurements. The results of the investigation survey that indicate the survey unit does not meet the release criteria along with the initial survey measurements will be transferred to a characterization data base file. If remediation is required, a second final survey will be performed and the results of the second final survey after remediation will be included in the final survey report.

7.0 REFERENCES

1. NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," August 1988.

2. "Fort St. Vrain Decommissioning Plan", Public Service of Colorado, November 05, 1990; Revised July, 1991; Revised April, 1992; Approved November, 1992.

3. Annual Radiological Environmental Report, Summary Report for the Period January 01, 1991 - December 31, 1991", Public Service of Colorado, April 24, 1992.

4. "Annual Radiological Environmental Report, Summary Report for the Period January 01, 1990 - December 31, 1990", Public Service of Colorado, April 12, 1991.

5. "Fort St. Vrain Initial Radiological Site Characterization Report", April 30, 1992.

6. "Results of ISFSI Site Background Radiation Study", Department of Radiology and Radiation Biology, Colorado State University, November 02, 1990.

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

8. "Supplement to Applicant's Environmental Report Post Operating License for Proposed Decommissioning of the Fort St. Vrain Nuclear Station", April, 1992.

9. J. Berger, NUREG/CR-5849, "Manual for Conducting Radiological Surveys in Support of License Termination," June 1992 (Draft).

10. NRC Memorandum P.B. Erickson (NRC) to Seymour H. Weiss (NRC); Subject: "Summary of Meeting with Public Service Company of Colorado (PSC) To Discuss Preliminary Decommissioning Plan of June 30, 1989," dated August 24, 1989.

11. NRC Letter, Pittiglio to Crawford, dated June 15, 1994, "Approval of a Modification of Facility Release Criteria for Tritium and Iron-55 Surface Contamination at Fort St. Vrain Nuclear Generating Station."

12. NRC Letter, Pittiglio to Crawford, dated January 18, 1995, "Response to Proposed Modification of Removable Surface Contamination Release Criteria of Removable Surface Contamination for Tritium and Iron-55 at Fort St. Vrain Nuclear Generating Station."

13. NUREG/CR-5512, U.S. Nuclear Regulatory Commission, "Residual Radioactive Contamination From Decommissioning", Volume 1, October, 1992.

14. Fort St. Vrain Decommissioning Project Radiation Protection Manual, Scientific Ecology Group.

15. Fort St. Vrain Nuclear Station Radiological Environmental Monitoring Program Annual Summary Report for 1989.

16. "PSC Site Characterization Data," Department of Radiology and Radiation Biology, Colorado State University, 1990.

17. "Shoreham Decommissioning Project Termination Survey Plan", Shoreham Nuclear Power Station, April, 1993.

18. PSC Letter, Warembourg to Austin (NRC), dated September 21, 1994, "Response to Comments Regarding the Final Survey Plan for Site Release."

19. PSC Letter, Fisher to Austin (NRC), dated January 11, 1995, "Response to Comments Regarding the Final Survey Plan for Site Release."

20. PSC Letter, Fisher to Austin (NRC), dated January 20, 1995, "Response to Comments Regarding the Final Survey Plan for Site Release."

21. NRC Letter, Pittiglio to Crawford, dated January 26, 1995, "Response to NRC Comments on the Final Survey Plan for Site Release for Fort St. Vrain Nuclear Generating Station," Approval of Final Survey Plan.

8.0 GLOSSARY

Administrative Action Level - A contamination level used as the investigation threshold in survey units to evaluate the need for additional investigation, reclassification or remediation.

Affected Survey Unit - The designation assigned to a survey unit which has a history of radioactive materials, or where the results of characterization, turnover and/or final surveys have verified radioactive materials in excess of the MDA for the analysis/survey equipment.

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

Background Value - A numerical value, statistically evaluated and expressed in the appropriate unit of measurement, defining the background radiation.

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

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

Characterization Survey - A radiological survey and supporting evaluations performed to establish the FSV baseline radiological condition for planning decommissioning activities. The Characterization Survey activities are described in and controlled by the FSV Decommissioning Project Radiation Protection Manual. The Characterization Survey results are contained in the FSV Radiological Site Characterization Report.

Component - An individual equipment item, e.g., a valve, pump, tank, or motor.

Confidence Interval - A range of values derived from a sample such that there is a probability α, that a population parameter being estimated, e.g., a mean value, lies within the range (Ref. 15).

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

Critical Level - The net activity (total minus background) for reaching a decision of non-detection if the net activity is less than the critical level, and the decision of detection if the net activity is greater than the critical level. The critical level is determined for reporting purposes, however, the critical level is not compared to the guideline value for demonstrating suitability for release for unrestricted use.

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

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

Elevated Area Guideline - The value which individual measurements may not exceed under any conditions.

Final Survey - Radiological measurements, evaluations and supporting activities undertaken to demonstrate that the FSV facility satisfies the criteria for unrestricted use. Also referred to as Termination Survey.

Final Survey Group - Westinghouse personnel selected to design and implement the final survey.

Final Survey Report - A report describing the methods and results of the Final Survey. It initiates the NRC review and final inspection of the facility for termination of the facility license. It is also called the Final Report.

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

Hard-to-Detect Nuclide - (HTDN). A nuclide emitting radiation(s) of low energy or intensity such that detection utilizing typical field instrumentation is difficult. Also: Difficult-to-Detect, Difficult-to-Measure.

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

Interim Report - A report prepared at the conclusion of a major phase of the Final Survey. The interim report will contain a description of the areas surveyed and survey results. Survey results in this report will be compiled and submitted in the Final Report.

NRC - U.S. Nuclear Regulatory Commission

Non-Suspect Affected Survey Unit (Within an Affected Area) - The designation assigned to ceiling and wall surfaces above two meters within an affected area where it is not expected that radioactive materials exist at average levels in excess of 25% of the removable or total activity release criteria, or the MDA for the equipment used to perform the survey, whichever is larger; and it is not expected that individual measurements exist in excess of 50% of the release criteria, or the MDA for the equipment used to perform the survey, whichever is larger.

Operational Radiological Survey - A radiological survey performed in accordance with procedures contained in the FSV Decommissioning Project Radiation Protection Manual. Operational surveys are distinct from, and usually performed prior to, final surveys.

Open Land Area - The category of survey units which includes site grounds within the Restricted Area, the liquid effluent pathway, the Farm Pond and selected owner-controlled areas. Also: Outdoor Area, Environs.

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

Release Criteria - A term used to identify the radiological requirements for release of the FSV facility for unrestricted use. These requirements, which consist of specified limits for residual contamination and radiation levels, are specified in the FSV Decommissioning Plan. Also: Release Limit, Guideline Value, Site-Specific Guideline Value.

Release Limit - The principal numerical limits in the facility release criteria presented in Table 3.1.

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

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

Restricted Area - The group of major buildings on the FSV site located inside the security fence. This group includes of the Reactor, Turbine, Radwaste Compactor, and Maintenance buildings.

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

Site Characterization Report - A report (including addenda) which documents the surveys, calculations and evaluations and presents the results of the Fort St. Vrain Initial Radiological Site Characterization.

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

Surfaces and Structures - All FSV Nuclear Generating Station site buildings and their surfaces (generally identified as civil structures). For purposes of the Final Survey, all structures such as platforms, restraints, supports and other physical items are considered to be components. External surfaces of piping systems, heating and ventilation systems, tanks, stacks, etc., are also treated as components in the Final Survey. Also: Structures.

Survey Area - The most general category, comprised of surfaces to be further defined as one or more survey units, the bounds of which are defined by existing facility physical features such as the column-and-row layout of the facility, intersection of walls, or structural I-beams.

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

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

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

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

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

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

Survey Unit - A contiguous area (usually) with similar characteristics and contamination potential. Survey units are established to facilitate the survey process and aid in the statistical evaluation of the survey data. Since survey units are designed to be contiguous areas with similar characteristics and potential for contamination, the actual size of a survey unit is not deemed to be critical, provided each survey unit contains a sufficient number of measurement locations. Generally, 30 measurement locations are required for each survey unit.

The three categories of survey units are: 1) Structures and Surfaces, 2) Plant Systems, and 3) Open Land (environs) Areas. Survey units are classified as affected, those with the potential of being contaminated, and unaffected. In addition, affected survey units within a room or building are subdivided into suspect affected and non-suspect affected survey units to allow for a concentrated survey effort in those areas most likely to be contaminated. By definition, floors and walls below two meters in an affected area are classified as suspect affected. The walls above two meters and the ceiling in an affected area may be classified as either suspect affected or non-suspect affected depending on the potential for contamination.

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

Suspect Affected Survey Unit (Within an Affected Area) - The designation assigned to floor and wall surfaces below two meters within an affected area. Also, the designation assigned to all additional surfaces if there is reason to believe that there was licensed material present, unless it meets the criteria for a non-suspect affected survey unit.

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

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

Unaffected Survey Unit - A survey unit having little or no potential for the presence of licensed material based upon operational history, characterization data, and routine surveillance. The presence of licensed material (in excess of MDA) or previous remediation would preclude a survey unit from being classified as unaffected.

Verification Survey - A radiological survey which consists of repeat measurements at a specified fraction of the survey measurement locations in a survey unit, usually selected at random, to provide an independent check of final survey measurements. Also called a replicate survey.

Work Instruction - A document used to guide performance of a task.