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July 26, 1991
Fort St. Vrain
Unit No. 1
P-91240

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

ATTN: Dr. Seymour H. Weiss, Director
Non-Power Reactor, Decommissioning and
Environmental Project Directorate

Docket No. 50-267

SUBJECT: INITIAL RADIOLOGICAL SITE CHARACTERIZATION PROGRAM

REFERENCES:
1) NRC letter Weiss to Crawford dated July 1, 1991 (G-91132)
2) PSC letter Crawford to Weiss dated May 15, 1991 (P-91138)

Dear Dr. Weiss:

Attachment 1 to this letter is the Westinghouse Electric Corporation (Westinghouse) copyrighted Fort St. Vrain (FSV) Initial Radiological Site Characterization Program. In response to your position outlined in Reference 1, the attached copyrighted version of this document replaces the proprietary version of the FSV Initial Radiological Site Characterization Program transmitted by Reference 2. The copyrighted version is identical to the previously submitted proprietary version except that the brackets that enclosed the information which was considered proprietary by Westinghouse have been removed. Attachment 2 is the Westinghouse Copyright Notice associated with the FSV Initial Radiological Site Characterization Program.

Attachment 3 is a letter from Westinghouse to Dr. Thomas Murley, Director, Nuclear Regulatory Commission. This letter provides the Westinghouse position on this matter. Please return the proprietary version of the FSV Initial Radiological Site Characterization Program to Westinghouse as requested in Attachment 3.

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If you have any questions regarding the attached information, please contact Mr. M. H. Holmes at (303) 480-6960.

Very truly yours,

A. C. Crawford

A. C. Crawford
Vice President
Nuclear Operations

ACC/GDS
Attachments

cc: Regional Administrator, Region IV

Mr. J. B. Baird
Senior Resident Inspector
Fort St. Vrain

Mr. Robert M. Quillen, Director
Radiation Control Division
Colorado Department of Health
4210 East 11th Avenue
Denver, CO 80220

P-91240
Attachment 1

FORT ST. VRAIN NUCLEAR GENERATING STATION

INITIAL RADIOLOGICAL SITE CHARACTERIZATION PROGRAM FSV-P-SCP-100

PROGRAM DESCRIPTION

May 10, 1991

Prepared By:

SCIENTIFIC ECOLOGY GROUP, INC.

Wholly owned subsidiary of Westinghouse Electric Company

1560 Bear Creek Road

Oak Ridge, Tennessee 37830-2530

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1.0 INTRODUCTION

1.1 Purpose

The purpose of the Initial Site Characterization Program (SCP) is to collect and analyze radiological survey data needed to determine the extent of decontamination/dismantlement activities at the Fort St. Vrain (FSV) facility. This will be accomplished by providing radiological data that indicates whether FSV equipment, systems and structures have become internally or externally contaminated and/or activated. This data will be used to verify and supplement information provided by Public Service Company of Colorado (PSC), (Request For Proposal dated December 8, 1989, Reference 7.1). This data will also serve as the baseline radiological information for future reference during the decommissioning project.

The SCP defines the methodologies that will be used to radiologically characterize structures and systems at FSV. The program defines the information needed to accurately characterize the FSV facility. This data will be obtained for the purposes of project planning, cost estimation, evaluation and verification of historical data, and to determine the decontamination, dismantling and sequencing techniques necessary for effective plant decommissioning. The information generated from the SCP will be compared against the established regulatory criteria for site release. Using this criteria, Westinghouse will be able to determine the extent of decontamination/dismantlement activities.

1.2 Scope

The SCP addresses the use of FSV historical radiological information, methods for selection of sample locations, methods for analyzing data, requirements for documentation and quality assurance.

This program will be used by PSC and Westinghouse for completing the FSV facility decommissioning. The scope of this program is limited to the radiological characterization of the site structures and buildings, balance of plant (BOP) systems and equipment, the prestressed concrete reactor vessel (PCRV) and internal parts. The determination of whether site materials are suitable for unrestricted release is beyond the scope of the site characterization program and will be addressed in the final radiation survey program. The SCP divides the FSV structures and systems into four (4) major elements which are:

- Structural Characterization (Section 4.1)

- BOP System and Equipment Characterization (Section 4.2)
- PCRV and Internals Characterization (Section 4.3)
- Environmental Characterization (Section 4.4)

1.3 Organization and Responsibilities

The SCP will combine the efforts and expertise of various personnel to ensure that all facets of the program are properly implemented. The SCP Organization will provide the sound technical guidance to ensure that the characterization program is effectively implemented. The SCP organizational structure is shown in Appendix A.

Responsibilities of the SCP organization members are as follows:

- The Westinghouse Project Director (WPD) is responsible for overall project management.
- The Westinghouse Technical Services Manager is responsible for administrative project support and management and reports to the WPD.
- The Scientific Ecology Group (SEG) Project Director is responsible for overall technical direction and management of the radiological site characterization operations.
- The Scientific Ecology Group SCP Manager is responsible for implementation of the radiological site characterization operations and reports to the SEG Project Director. This position will assume the authorities and responsibilities of Project Radiation Protection Manager during the decommissioning project.

The SCP Manager will interface with FSV health physics, maintenance and operations staff. This interface will ensure all characterization activities are planned, coordinated and scheduled in conjunction with normal FSV operations. The SCP Manager responsibilities include:

- Managing and planning characterization activities.
- Supervising the site characterization staff.
- Performing evaluations for report preparation.

- Coordinating the preparation of the initial site characterization report.

The site characterization staff will have the primary responsibility for implementing the SCP procedures, performing surveillance and documenting the characterization results.

1.4 SCP Management Oversight and Controls

The SCP management organization will provide direction and oversight to personnel performing site characterization activities. Some specific management oversight and controls that will be used include:

- Personnel will meet the applicable training and qualifications as described in the SEG QAP-107, "Quality Plan, Initial Radiological Site Characterization FSV/PSC" (Reference 7.6).

- Onsite gamma isotopic analysis of characterization samples will be performed in accordance with FSV radiochemistry procedures.

- Specific SEG site characterization implementing procedures will be provided for each major site element to be characterized (structural, BOP system and equipment, PCRV and environmental).

- Instruments used to perform radiological characterization surveys will be calibrated, operated and maintained in accordance with approved SEG site characterization procedures.

- Characterization sample locations will be identified in the specific work packages for each area being characterized.

- FSV maintenance procedures will be used when characterization activities require system isolation or entry. Station Service Requests (SSRs) will be generated when applicable.

The quality assurance responsibilities for the SCP are as follows:

- The SEG Quality Assurance Manager is responsible for performing surveillance and auditing of the site characterization activities.

- The Westinghouse Nuclear and Advanced Technology Division (NATD) Quality Assurance Manager is responsible for surveillance of SEG quality assurance.

- PSC quality assurance personnel will conduct oversight of the Westinghouse quality assurance surveillance and auditing functions through the PSC Quality Assurance (PSC QA) monitoring program. In addition to monitoring of Westinghouse QA activities, PSC QA may perform monitoring of physical site characterization work.

The interface between SEG QA, NATD QA and PSC QA is described in FSV-QA-001, "FSV Quality Assurance Interfaces" (Reference 7.7).

2.0 GENERAL SITE INFORMATION

2.1 FSV Site Description

FSV is owned by Public Service Company of Colorado. PSC was licensed by the Nuclear Regulatory Commission to operate the 330 MWE High Temperature Gas-Cooled Reactor (HTGR). Commercial operation began in July 1979, and ended in August 1989.

FSV is located approximately thirty-five miles north of Denver and three and one-half (3 1/2) miles northwest of the town of Platteville in Weld County, Colorado. PSC owns the 2798 acre site and has designated a distance of 100 meters from the reactor building as the exclusion area boundary.

2.2 Historical Radiological Information

Historical information regarding the pre-operational and operational phases of FSV will be made available by PSC to assist in establishing the radiological status of the facility. These documents are controlled and stored at the FSV records center.

The historical files will provide the operational information necessary to assist in site characterization efforts and aid in determining the radiological status of the facility. This information will be reviewed and used to determine the most appropriate methods for performing site characterization activities.

A review of FSV operational history will be performed, as appropriate, to determine the radiological status of FSV structures and systems. The results of this review will be applied in determining the location for radiological measurements.

The types of documentation that will be reviewed include:

- Radiation work permits (RWPs) of work performed, that would be applicable to site characterization.
- Surveillance reports for routine and special surveys containing the radiation and contamination levels for specific areas within the plant.
- Radioactive material releases, spills and incident reports.
- PSC site characterization records used to represent the radiological status of FSV.

A preliminary review of site radiological data has identified the following site structures as having a potential to be radioactively contaminated. These structures are:

- Reactor Building
- Turbine Building
- Radiochemistry Laboratory
- Helium Storage Building
- Waste Compacting Building
- New Fuel Storage Building

3.0 SAMPLING METHODOLOGY

3.1 General Considerations

This section of the SCP describes the methods that will be used to determine the current radiological status of FSV structures, equipment and systems, reactor components, and the immediate site environment. This data will be used in conjunction with PSC site characterization records and FSV historical information to verify the facility radiological status, as presented in the PSC proposed decommissioning plan.

The SCP will utilize both unbiased and biased characterization schemes to obtain radiological data. Both methods (unbiased and biased) will employ the same analytical techniques for determining radioactive concentrations and radiation levels.

Work packages will be developed to describe the characterization requirements for specific FSV structures and systems. Work packages will be implemented in accordance with approved SEG implementing procedures. Work packages will typically include details such as; type of survey (biased or unbiased), number of sample locations, radiological history, type of instrumentation to be used, and if applicable, maps and diagrams needed to identify sample locations. Work packages will be approved by the SCP Manager or designee.

3.2 Unbiased Survey Method

Unbiased survey methods will be used in areas where no historical information is available or the areas are expected to have little or no surface contamination present. The unbiased survey method selects the sample locations for radiological measurement and is not influenced (biased) by previous data or historical information. Areas that are known to be contaminated will be excluded from the unbiased survey program. The unbiased results will be evaluated statistically to determine if an adequate number of sample locations have been monitored to accurately establish the radiological status of FSV structures and systems.

The unbiased survey method provides reasonable confidence that surface radiation levels (both fixed and removable) can be evaluated to meet established statistical requirements. The confidence levels must fall to within plus or minus 20% for all surfaces that exhibit contamination greater than background. This degree of confidence will be achieved by taking approximately 30 measurements on a surface and using these measurements to determine if the number of measurements meets the desired confidence and error limits.

Structures, buildings and large areas will be divided into smaller manageable divisions called Survey Areas. Examples of Survey Areas include; a building elevation, the roof of a building, a room or even several small buildings. These Survey Areas will be further broken down into Survey Units. Survey Units are areas that are expected to have similar modes of contamination deposition patterns. Survey Areas will typically be divided into the following Survey Units:

- Floors - areas of heavy traffic
- Walls - settling of dust, sprays or leaks
- Horizontal surfaces - dust settling

Each Survey Unit will have approximately 30 survey locations that are randomly distributed, to the extent practical, over the entire surface area. A list of unbiased survey locations will be prepared prior to the survey and be incorporated into the work package.

After the initial measurements are taken for a Survey Unit, the number of sample location measurements will be evaluated to determine the mean surface radiation levels. Upon completing the measurements for a Survey Unit, a statistical evaluation will be conducted to determine if a sufficient number of sample locations has been evaluated. This evaluation will be performed as part of the final data analysis using the equation listed below (Reference 7.12). This equation is only applicable to radiation measurements greater than established background.

$$N=\left(\frac{ts}{r\bar{x}}\right)^2$$

where

N = required number of measurements

n = number of sample locations actually measured (typically 30)

t = the "student 't' statistic" (for example; the value for 't' at 95% confidence and 29 degrees of freedom is 1.699)

s = estimated population standard deviation of the average surface radiation level

r = acceptable relative error (0.20)

$\bar{x}$ = the average surface radiation level

- If N is computed to be less than the number of initial survey location measurements (n), no further measurements will be required.
- If N is greater than the number of initial survey location measurements (n), then N minus n equals the number of additional evenly distributed measurements that will be taken.

Upon completion of the measurements, the sample results will be evaluated and compared to the calculated background distribution. If the evaluation identifies inconsistent or unexpected surface radiation levels, the area or unit will receive further evaluation. Surfaces requiring reevaluation will be divided into different sample locations and new measurements will be performed. The new data will be evaluated and compared with the initial sample location data. If a contaminated area cannot be segregated from an uncontaminated area, the entire area will be considered contaminated and treated accordingly. This data will be compiled and stored with the work package and the results will be incorporated into the final characterization report.

3.3 Biased Survey Method

The biased survey method will be used tc .sess structures and systems which are known or potentially contaminated. PSC site characterization records, FSV historical information, and SEG unbiased survey data will be used to determine if, and how, biased survey techniques will be employed. Areas will be selected based on the potential for deposition of radioactive material (e.g., crud traps). Heavily traveled areas, horizontal surfaces and surfaces under known contaminated systems will be included as part of the biased survey method (e.g., pipe or pump leakage, floor drains, sumps, etc.).

This method of surveillance will define the radioactivity levels in areas where there is a high probability of contamination. Biased sample locations will be selected to ensure that the radiological characterization is adequate to define known or suspected contaminated areas throughout the facility. In addition, survey locations found to be contaminated using the unbiased method will be resurveyed using this biased method.

Biased survey locations will also be divided into Survey Areas and Survey Units in the same manner as the unbiased survey method. However, when necessary these areas will be divided into smaller divisions than the unbiased survey areas.

3.4 Radiological Survey Methods

Measurements will be performed in accordance with the SEG site characterization procedures and will meet applicable standards provided by industry guidance documents. Radiological measurements will be performed by qualified personnel using calibrated portable survey instruments. Instrumentation used for characterization surveys will be calibrated in accordance with approved SEG implementing procedures. Sources used for instrument calibration will be traceable to the National Institute for Standards and Technology (NIST).

Radiological measurements taken at a sample location will normally be as follows:

- A beta/gamma radiation measurement approximately one centimeter (cm) from the surface.
- A smear of approximately $100cm^2$ area, collected and counted for gross beta/gamma activity and where appropriate gross alpha activity.
- A direct gamma radiation measurement approximately one centimeter (cm) from the surface.
- A direct gamma radiation measurement approximately one meter from the surface.
- A direct alpha radiation measurement as close t[illegible] surface as possible on approximately 10% of the survey locations when applicable.

Bulk sampling of various materials such as insulation, concrete, paint chips and sediment deposits will be analyzed for gamma isotopic content on a case-by-case basis. The analysis of bulk samples will be performed by qualified personnel or vendors using calibrated equipment.

The requirement for performing specific radiation measurements will be evaluated by the SCP Manager or designee. If it is determined that a radiation measurement is not appropriate, the SCP Manager or designee may modify the requirements. Sufficient radiological data will be provided in the work package, to support modifications of radiological measurements.

3.4.1 Removable Surface Contamination Determinations

Smear samples will be taken at all selected sampling locations. All smear samples will be analyzed for gross beta/gamma activity and at least one smear with the highest detectable activity from each Survey Unit will be counted for gross alpha activity.

If activity is detected, smears with the highest gross activity from each Survey Unit will be selected for quantitative and qualitative gamma isotopic analysis. These smears will be analyzed by high resolution gamma spectroscopy. Additional radiological analysis for non-gamma isotopes may be performed if gross beta/gamma results do not correlate well with the gamma spectral analysis. If elevated gross alpha activity is detected, selected smears will be analyzed for uranium, thorium and transuranic isotopes. All smears collected in the New Fuel Storage Building will be counted for gross alpha activity.

3.4.2 Total Surface Contamination Determinations

Total surface contamination (fixed and removable) will be measured at each sample location. The type of equipment and detectors will be defined in approved SEG implementing procedures and the work packages for the specific types of characterization. Direct beta/gamma measurements will be made at approximately one cm from the surface of interest. Direct alpha measurements will be made as close to the monitoring surface as possible, when applicable.

If a Geiger-Mueller (GM) type detector is used, it will be a thin-window (less than 7 mg/cm^2) pancake type detector. Alternate detectors may be used when access constraints preclude the use of the pancake detectors and will be defined in the work packages. The count rate obtained will be corrected for background, geometry, detector area and efficiency to yield measured results in dpm/100 cm^2.

In situations where surface contamination is found and there is a likelihood of deposition below the surface, (i.e., a porous surface), it will be necessary to determine the depth of the contamination. This task will be performed by core sampling or other appropriate methods. Sampling activities of this nature will be performed in accordance with applicable FSV station procedures.

3.5 Data Collection

Radiological data will be collected by a team of qualified characterization specialists working to approved work packages and procedures. This team will receive day-to-day directions from the SCP Manager or designee.

Radiological survey records will provide traceability and are the primary method for documenting the current radiological status of FSV structures and systems. Characterization data will be collected and compiled to provide a basis for decisions regarding the scope and methods for decommissioning and dismantling of the FSV facility. In addition, the SCP data will provide initial information to plan and schedule decommissioning activities.

To accurately characterize the FSV facility, SCP personnel will use industry standard survey techniques and instrumentation to measure radiation levels and radioactivity concentrations. The various survey methods will result in measurement and documentation of some, or all of the following:

- The type of equipment and instrumentation used to survey the facility and analyze samples collected.

- The amount of fixed and removable gross beta/gamma and alpha surface contamination.
- The isotopic distribution of gamma emitting nuclides.
- Surface radiation exposure rates.

The radiological data collected during SCP will be recorded to provide specific information and identification of the FSV structures and systems. This information will typically include:

- Area locations and surface activity levels of the FSV structures and systems.
- Depth and estimated activity levels of contaminant penetration into surfaces, when applicable.
- Location and activity levels, if any, of the radioactive materials in soil.
- Location and estimated activity levels in accessible systems and equipment.
- Activity levels induced by activation in the PCRV and associated components, as they become available.

This data will also be used to verify the radiological status of the FSV facility as described in the proposed decommissioning plan. Based on the characterization results, structures and systems will be segregated by level of radioactive contamination. These are as follows:

- Category 1 - Equivalent to background.
- Category 2 - Greater than background, but meeting the release criteria for unrestricted use.
- Category 3 - Greater than the release criteria for unrestricted use.

Structures and systems with results in Category 1 are equivalent to background and will be considered free of contamination.

Structures and systems with results in Category 2, will be considered to be slightly contaminated as a result of FSV operations and will be treated accordingly.

Structures and systems with results in Category 3, will be classified as contaminated. Items in this category will either be decontaminated or disposed of as radioactive waste during the decommissioning project.

4.0 CHARACTERIZATION SCHEME

This section is divided into four subsections: Structural Characterization, BOP System and Equipment Characterization, PCRV and Internals Characterization and Site Environmental Characterization. Each subsection describes the means of obtaining the required data and a rationale for the approach to be taken.

4.1 Structural Characterization

Structural characterization will determine the radiological status of the buildings and other structural facilities on the FSV site. The characterization will assess the surface radiation levels of the building including floors, walls, beams and other horizontal surfaces.

The scope of the characterization effort will depend on the activities that occurred in the structure. For example, the reactor building will receive considerable attention, whereas the effort in offices outside of the protected area will be less extensive. The structures selected for the biased survey method were based on a preliminary review of historical use and survey data.

The following is a brief description of the structures that have been selected for the biased survey method:

- Reactor Building

  The reactor building houses the prestressed concrete reactor vessel (PCRV), fuel handling area, fuel storage wells, fuel handling facilities, decontamination and radioactive liquid/gas waste processing systems/equipment, the hot service facility, various support equipment, storage and laydown areas.

- Turbine Building

  The turbine building houses the turbine generator with condensing, feedwater and other auxiliary systems. Also in the turbine building is an auxiliary bay area which includes reactor plant ventilation equipment, access control area with decontamination facilities, and HP counting equipment.

- Radiochemistry Laboratory

  The radiochemistry laboratory is used to perform various analyses on plant materials to determine radioactive concentrations. The radiochemistry laboratory stores radioactive calibration sources for counting instrumentation.

- Helium Storage Building

  The helium storage building houses the storage tanks for the purified helium use in the reactor as primary coolant.

- Radioactive Waste Compacting Building

  The radioactive waste compacting building is used as a lay down, radwaste storage area and dry active waste compactor.

- New Fuel Storage Building

  Non irradiated reactor fuel was stored in the building prior to being loaded into the PCRV.

The unbiased survey method will be used for all other site buildings and structures. Historical data and information provided in the proposed decommissioning plan indicates that no evidence of radioactive contamination exists in these areas.

The structural survey will include the determination of total fixed and removable gross beta/gamma surface contamination, gross alpha surface contamination, if appropriate, and the gamma exposure rate at one meter from the surface. If, or when, surface contamination is detected, quantitative and qualitative gamma isotopic analysis will be performed of smear samples or of sample material removed from the surface in question. In addition, if surface contamination is found on porous materials such as concrete, measurements may be made to determine contaminant depth.

The structural characterization effort will be divided into unbiased and biased sampling schemes. Areas will be divided into Survey Areas and Survey Units and the current radiological status will be determined and documented. This sample population will then be compared to a background population to determine if it has been affected by FSV operations.

4.2 BOP System and Equipment Characterization

Characterization activities will determine the radiological status of potentially contaminated systems which are within the reactor building. The scope of the characterization will include the assessment of interior surfaces of reactor building components and systems that are potentially contaminated. In addition, the radiological status of all balance of plant systems (BOP) external to the PCRV will be determined. Radiological surveys will be performed on the non-contaminated systems at "worst case" sample locations and other areas as needed.

Biased surveys will be performed on systems and components identified as being potentially contaminated. The goal is to document the radiological status of the BOP systems and equipment for future use during the decommissioning project. This data will also be incorporated into the final site characterization report. Potentially contaminated systems will be characterized by selecting representative sample locations for direct measurement and sample collection. Sample collection will frequently require the opening of systems to expose the interior surfaces so they can be surveyed. The survey will include determination of total surface contamination (fixed and removable) and gross alpha contamination, if required.

The BOP systems and equipment within the reactor building will be radiologically characterized. Determination regarding systems that could potentially contain radioactive contamination will be based upon the following criteria:

- Systems or equipment which carry fluids or gases that have circulated through the PCRV, radioactive waste systems or spent fuel storage systems.
- Systems and equipment used to collect, circulate, or discharge air from the reactor building.
- Systems and equipment used to collect or drain fluids from the reactor building.
- Systems or equipment with indications of contamination, as determined in FSV historical information.

Equipment will be characterized on a system by system basis. Specific items within each system will be selected to represent the overall condition of the system. When multiple components are to be selected to represent an entire system, components from different plant elevations and lateral locations will be chosen to avoid possible anomalous results that could be created by localized conditions.

In order to determine the distribution of radioactive contaminants within some systems various components will be selected to represent the "worst case", "average case" and "least case" conditions. Components in BOP systems that have the highest probability of being contaminated will be selected to represent the worst case contamination levels for that system. The following are generic types of components that would likely be considered as worst case:

- Items that impinge flow, such as pumps, turbines, or blowers.
- Items that restrict or divert flow, such as valves, elbows or pipe tees.
- Items such as tank bottoms or filter housings that provide horizontal surfaces upon which materials deposit or settle.

Specific sample locations will be selected to represent the worst case condition. Worst case components and sub-components may be selected, as applicable, based upon external contact gamma exposure rate measurements. Components of each generic type (e.g., pump, valve, orifice, etc.) with the highest contact exposure rate may be chosen to represent the worst case. If there are no components within the system with contact exposure rates greater than background, sound engineering judgement will be used to select the sample locations for worst case conditions.

If radioactive contamination is detected under the worst case investigation, additional sample locations may be selected in the system to bound and define the average case contamination levels throughout the system. These locations will be selected on surfaces with little or no crud trap potential, such as pipe or duct walls that are parallel to material flow paths.

Contamination determinations will be made on the internal side wall and on the internal bottom surface where contamination levels could differ because of gravitational settling. Typically, contamination levels on the bottom surfaces represent the worst case system surface contamination levels. The side wall surfaces typically have the average case or least case contamination levels in a system.

The methodology for characterization of BOP contaminated systems includes the following:

- Development of the work packages for characterization of the BOP contaminated systems. Work packages will identify the survey locations, type of surveys, external and internal sample points and type of sample analysis.
- Identification of equipment and components requiring disassembly.

- Pipe systems will be characterized by analyzing liquid samples and taking smear samples from the inside surfaces of pipes, valves, pumps, tanks, and low spots along pipe runs.

- The drainage system tanks, sumps, and pumps will be evaluated for internal contamination.

- The ventilation system will be characterized by entering the duct segments to smear internal duct surfaces, blower fan blades, and filter housing.

The data collected as part of the system and equipment characterization will be analyzed to quantify the following parameters:

- Determination of the mean and range of total surface contamination and removable activity on internal surfaces of systems.

- Data for estimating the total radionuclide inventory within each system.

- Determination of the deposition pattern of radioactive contaminants within systems.

- Determination of the gamma isotopic identity and concentration of the radionuclides within a system.

- Evaluation of characterization data to determine decommissioning work scope.

- Comparison of characterization results with the information provided in the proposed decommissioning plan.

4.3 PCRV and Internals Characterization

This section of the SCP deals with the characterization of the Prestressed Concrete Reactor Vessel (PCRV) and the PCRV internals. The radioactivity of primary concern in these areas is that induced by neutron activation during operation of the reactor and the radioactivity that has plated out on the surfaces of the PCRV internal components.

The following information is available and will be considered during PCRV characterizations:

- Calculated concentrations and total activity of induced radioactivity present in the PCRV and its internal components.

- Calculated exposure rates from activated components.
- Calculated depth at which PCRV activated concrete is less than 5 micro R/hr above background one meter from the surface.
- Calculated and measured levels of activity plated out on the surfaces of core internals.

When access to the PCRV internals becomes available, the activation and plate out results will be confirmed. The confirmation will include:

- Collecting borings of the PCRV to determine the depth of the induced radioactivity.
- Measuring radiation levels of PCRV internal components.
- Sampling selected PCRV internal components and determining the quantity and identity of induced radioactivity.
- Sampling selected PCRV internal components and determining the quantity and identity of radioactivity plated out on the surface of the component.

The measured data will be compared to the calculated data for similar items. Based on a comparison of the data points, scaling factors will be developed. These scaling factors will be utilized to determine the exposure rate and the activity levels contained in other PCRV internal components. During dismantlement of the PCRV, actual measurements of selected components will be taken and scaling factors will be adjusted as necessary.

4.4 Environmental Characterization

The purpose of this portion of the SCP is to describe the methods that will be used to determine the current radiological status of the FSV site. This data will be used to validate historical data and to identify additional areas requiring remediation during decommissioning.

This portion of the SCP will demonstrate, by comparison with offsite locations that the site meets the criteria for unrestricted use, and is, in fact, statistically equivalent to the local background. The characterization will further serve to demonstrate, by later comparison to final survey results, that decommissioning did not adversely affect the site.

Environmental media to be collected during the SCP includes soil, sediment, surface water and ground water samples. The Radiological Environmental Monitoring Reports

(REMP) for 1988 and 1989, the first and second quarter 1990 REMP Tables (Reference 7.8), and a report dated February 26, 1990, from Industrial Compliance, Inc. (Reference 7.9) evaluating the results of samples collected from 13 ponds associated with FSV, were all reviewed as a prelude to determining the types and location of specific samples. The following conclusions were contained in these reports:

1) Gross beta concentrations in the Gilcrest Well have routinely been greater than the reference supply in Fort Collins, Colorado. However, this difference cannot be due to reactor effluent activity for the following reasons:

- None of the individual fission or activation product radionuclides measured were significantly higher in the Gilcrest drinking water.

- Tritium concentrations measured at Gilcrest were only slightly statistically greater than those at Fort Collins.

- The city of Gilcrest does not treat its water to the same degree as Fort Collins. Gross beta concentrations are due to the elevated concentration of the naturally occurring U-238 and Th-232 decay products in the suspended solids.

2) Elevated concentrations of tritium have been routinely detected in the outlet of the Farm Pond which correlates well with release schedules of blowdown effluent from FSV.

3) I-131 has been detected in milk and ditch surface water. Neither the milk or the water contains any other fission products. This indicates a source other than FSV. The source of the I-131 was determined to be due to nuclear medicine use and release upstream of the sample point.

4) There is no evidence that the two onsite ground water wells (REMP location 251 and 1352) have been contaminated.

5) Cs-137 was observed in many environmental samples due to the Chernobyl fallout.

6) Co-60 was identified in a sample from the Goosequill Stub and Co-60, Fe-59 and Mn-54 were identified in a sample from the Farm Pond Inlet. It was reported by Industrial Compliance, Inc. that these results are considered by the analyst as false positive.

In summary, tritium is the only radionuclide reported in any effluent pathway. Other reported radionuclides have been dismissed for various reasons. However, it is incumbent that verification of these conclusions be obtained during the SCP. There is

no basis for assuming deposition of radionuclides as having occurred from plant airborne emissions.

Based on the aforementioned review, sampling and analysis will include samples of soil, sediment and surface water. Both biased and unbiased sampling will be performed which will provide a reasonable assurance that all areas of potential contamination have been surveyed. There will be approximately 40 unbiased sample locations selected. These sample locations will not include any paved areas or areas covered with concrete.

4.4.1 Unbiased Environmental Samples

Unbiased samples are those which are randomly distributed about the area to be sampled without regard to potential contamination pathways. In selecting the sample locations, a "stratified random" approach is used whereby the area to be sampled is divided into sectors and random locations are selected within each sector. There will be approximately 40 unbiased sample locations. These sample locations will not include any paved areas or areas covered with concrete.

The basis for the above number of unbiased samples is as follows:

1) It is assumed that the samples from the unbiased locations will be representative of a normal or lognormal distribution.

2) It is desired to characterize the site such that the values of soil/sediment contamination (in units of pCi/gm) for each detected isotope be known within plus or minus 20%, at a 95% confidence level.

3) For the precision and confidence level described above, the number of measurements used will be verified by use of the following equation:

$$N=\left(\frac{ts}{r\bar{x}}\right)^2$$

where

N = required number of samples

n = number of samples actually taken

t = the "student 't' statistic" (for example; the value for 't' at 95% confidence and 29 degrees of freedom is 1.699)

s = estimated population standard deviation of the average soil concentration of each nuclide

r = acceptable relative error (0.20)

$\bar{x}$ = average soil concentration of any non-naturally occurring and potentially facility related nuclide

When sample analysis is completed, the aforementioned equation will be applied to the isotopes detected which are deemed not naturally occurring. If the required number of samples (N) is less than or equal to the number actually taken (n), no further samples are required. If not, then (N-n) additional evenly distributed samples will be taken.

The unbiased sample locations and the method of sampling will be prescribed in the work packages and indicated on survey maps. This will be supplied to the characterization specialist before sample collection. These maps, will indicate a target sample area approximately 50 square feet. A sample may be taken anywhere within the area shown. If, for any reason, the sample cannot be taken within the designated area, an alternate location near the indicated location will be identified. The actual sample locations with sample coordinates will be identified in a work package. Samples will also be collected from selected "control points" that are removed from the FSV site.

4.4.2 Biased Environmental Samples

Within the scope of this program, "biased samples" are those for which there is a reasonable scenario or basis for suspecting that radiological contamination could have occurred. As an example, samples will be obtained from Goosequill Stub, the Farm Pond inlet and outlet, the Farmers Independent ditch, the Goosequill ditch and the Gilcrest well. As previously noted, samples from these locations have contained elevated levels of radioactive material. In addition, samples will be collected along all known or potential release paths from the facility. Examples of areas where samples will be collected include areas where radioactive waste materials have been stored, the entry into the Waste Compacting Building and areas where roof drains empty onto the ground.

The sample locations and a description of the manner in which biased samples are to be collected will be provided in SEG implementing procedures. The actual sample locations and a description of the sample will be identified in the work package. Samples will also be collected from selected "control points" that are removed from the FSV site.

4.4.3 Environmental Analytical Methods

Environmental samples will be collected and sent to an outside vendor laboratory for analysis. The laboratory will be selected on the basis of its ability to achieve the desired lower limits of detection (LLD) for the specific analyses requested. The specific analyses will include gamma isotopic and gross beta for all samples. Selected samples will be analyzed for additional specific isotopes such as Sr-89/90, Ni-63 and Fe-55.

A portable survey instrument will be used by a characterization specialist to measure the exposure rate at one meter above the surface of the sampling location. The instrumentation used will be calibrated with NIST traceable sources and the survey will be performed in accordance with the SEG implementing procedures.

4.4.4 Environmental Data Reporting and Analysis

A number of isotopes, either naturally occurring or not attributed to FSV operations have been detected in the past and are expected to be detected in the analyses. However, as reported in the PSC Radiological Environmental Monitoring Program, there appears to be no individual measurement exceeding reporting levels. Nonetheless, the Chernobyl fallout effects are still observable. The isotopes Sr-90 and Cs-137 are the result of atmospheric nuclear weapons testing and/or the April 1986 accident at the Chernobyl plant in the Soviet Union. The isotopes K-40, Ra-226, and Th-228 are naturally occurring and are in concentrations which are consistent with the Colorado Front Range area. As long as the measured concentrations are within the same range as the control samples, it will be assumed that their presence in the samples is naturally occurring or consistent with local background.

If the only isotopes detected are naturally occurring or consistent with the concentrations of the local background, then it will be concluded that there is no soil contamination on the site or its environs due to FSV operations. Samples containing isotopes other than expected, will be re-analyzed on a case-by-case basis. If the analyses provides a reasonable basis for the isotope to be consistent with natural background or to be a statistical anomaly, then the same conclusion of no contamination will be drawn.

If the identified isotopes are attributed to FSV operations, then the same statistical test as the one for the appropriate number of biased samples will be applied. If more samples are required, they will be collected. If the number of samples is deemed acceptable, then the concentrations will be compared to the proposed acceptance criteria for unrestricted use and the appropriate remedial action will be determined.

5.0 CHARACTERIZATION DOCUMENTATION AND REPORT

5.1 Documentation (General)

Documentation of site characterization survey results will be specified in the SCP implementing procedures. This will ensure that all documentation is accurately recorded in a format that can be used for future reference.

5.2 Data Reporting And Analysis

The unbiased results for the total surface contamination (fixed and removable) determinations will be converted into units of $dpm/100cm^2$. Data for each type of determination from a Survey Area or a Survey Unit will be presented in a tabular format and cross referenced to the specific location at which it was taken. The data from each Survey Area or Survey Unit will be evaluated and compared to determine the extent of remediation required. The mean, standard deviation and standard error of the mean will be calculated for each Survey Area or Survey Unit to establish the radiological status for specific FSV areas, structures and systems.

Surface contamination results of biased surveys will be converted to units of $dpm/100cm^2$ and presented in tabular form by Survey Area and Survey Unit. Location of these measurements will be similar to those used for the unbiased results.

The data will be sorted into the following categories:

- Category 1 - Equivalent to background.
- Category 2 - Greater than background, but meeting the release criteria for unrestricted use.
- Category 3 - Greater than the release criteria for unrestricted use.

Structures and systems with results in Category 1 are equivalent to background and will be considered free of contamination.

Structures and systems with results in Category 2, will be considered to be slightly contaminated as a result of FSV operations and will be treated accordingly.

Structures and systems with results in Category 3, will be classified as contaminated. Items in this category will either be decontaminated or disposed of as radioactive waste during the decommissioning project.

5.3 Final Report on Initial Site Characterization

The data will be presented in such a manner that: first, the radiological condition of the site is accurately depicted; second, the radiological condition of the site can be determined without further manipulation of the data; and third, types and locations of conditions exceeding decommissioning guidelines are easily identified. In order for these goals to be met, the radiological survey report will be written on two levels. The first level consists of a written description of the radiological condition of the site supplemented with figures illustrating significant radiological conditions. The second level consists of a detailed presentation of data in the form of tables, figures and maps. All areas, systems or structures requiring remedial action will be identified. Complete information concerning contamination and radiation levels will be given in tables so that data compilation and conclusions can be verified.

All original data will be available for examination on request. The original data will be stored by Westinghouse SEG in accordance with the SEG-QAP-107, "Quality Plan, Initial Radiological Site Characterization FSV/PSC" (Reference 7.6).

5.4 Structural Data

Data by building or facility location for unbiased samples will be compiled by Survey Units into a tabular presentation. A graphic depiction will be utilized for the area and location maps or diagrams with directions and elevations clearly noted for easy identification of survey locations. A sample population will be calculated for these locations by computing the mean, the standard deviation and a comparison with the calculated background for the surveyed material (See Appendix C and D). Those readings that exceed the established criteria for release for unrestricted use, will be listed.

The biased survey data will also be compiled by building or facility location and by Survey Area into a tabular presentation. A graphic depiction will be utilized. Maps or diagrams with directions and elevations clearly noted will facilitate identification of survey locations. Sample results will be compared to background to determine if data represents a contaminated area. Results will also be compared with the established criteria for release for unrestricted use.

The data will be reviewed to identify any trends and to determine its impact on the decommissioning project and final release of the facility.

5.5 System and Equipment Data

Data tabulated from each system surveyed will be summarized into a report. This report will include gross total and removable surface contamination levels, all isotopes detected, concentration of each isotope, and the ratio to the total activity. In addition, each survey location will be identified. The data will be summarized and its impact on the decommissioning project and the final release of the facility will be presented.

5.6 PCRV Activation Data

The results of PCRV activation analysis will be compared to calculated data. Based on this comparison, scaling factors will be developed such that calculated data can be converted to probable true exposure rates and radioactivity concentrations. This data will be utilized during decommissioning activities to determine potential exposure rates, shielding needs and disposal criteria.

5.7 Environmental Data

The results of the analyses of biased and unbiased environmental samples will be used to produce a location map or grid depicting locations of known or potential contaminated areas. Results of the analysis will include isotopes present and the concentration of each. These results will be compared with the known background. The results will be summarized into a narrative form identifying trends and assessing the impact on the decommissioning project.

6.0 QA/QC REQUIREMENTS

6.1 Data Collection

The Westinghouse SEG QA Program, as defined in the SEG-QAP-107, "Quality Plan, Initial Radiological Site Characterization FSV/PSC," is applicable to the conduct of the SCP. Samples and measurements at all locations will be collected using accepted and proven techniques and methodologies to ensure accurate results and consistent collection methods. Onsite sampling and measurement instrumentation will be subject to daily operational checks and periodic calibration to ensure both accuracy and precision of results. Records and calculations will be checked for errors and the use of appropriate recording and calculational techniques will be employed. Vendor laboratories will be required to verify data quality in accordance with SEG implementing procedures.

6.2 Measurement Quality Control

To ensure measurement quality, the following controls will be implemented:

- Five percent of soil, silt and liquid samples will be split and analyzed by an outside vendor laboratory.
- Approximately five percent of swipes will be reanalyzed (gross beta).
- Approximately five percent of the direct measurements taken for fixed surface contamination and exposure rates will be conducted and verified independent of the original characterization survey.

6.3 Records Retention And Control

The storage, maintenance, transmittal and retention of SCP records will be performed in accordance with the SEG-QAP-107, "Quality Plan, Initial Radiological Site Characterization FSV/PSC" (Reference 7.6).

Documentation collected during the characterization program are required to have a document reference number for purposes of information tracking, reference, and future retrieval. Instrument measurements and analytical results will be documented and survey parameters will be recorded.

The document reference number shall be unique to those used during operational phases of FSV. In general, any information generated in the form of memorandum, reports, surveillance functions, analytical results or instrument measurements, etc. in support of the characterization program will be traceable.

NOTE: The following is a list of references considered during program development and is not intended to imply full compliance with the material listed.

7.0 REFERENCES

7.1 NDG-89-1526; "Decommissioning and Conversion of the Fort St. Vrain Nuclear Generating Station." Public Service Company of Colorado Request for Proposal, December 8, 1989. Also Addendum-A, February 12, 1990, and Addendum-B, February 16, 1990.

7.2 USNRC, "Below Regulatory Concern; Policy Statement," Effective Date, July 3, 1990.

7.3 NRC letter from Mr. P. Erickson to PSC Mr. C. Crawford, dated October 4, 1989, (Docket Number 50-267). Subject; "Fort St. Vrain Decommissioning Financial Plan and Preliminary Decommissioning Plan - Request for Additional Information."

7.4 USNRC, Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors," June 1974.

7.5 NUREG/CR-5512, "Residual Radioactive Contamination from Decommissioning: Technical Basis for Translating Contamination Levels to Annual Dose," Draft Report for Comment, January 1990.

7.6 SEG-QAP-107, "Quality Plan, Initial Radiological Site Characterization FSV/PSC."

7.7 FSV-QA-001, "FSV Quality Assurance Interfaces."

7.8 Colorado State University, "Radiological Environmental Monitoring Reports (REMP)." 1988 and 1989 and REMP Tables, First and Second Quarter, 1990.

7.9 Industrial Compliance, Inc., report dated February 26, 1990.

7.10 "Decommissioning and Conversion of the Fort St. Vrain Nuclear Generating Station," Proposal to Public Services Company of Colorado, Reference Volume 6, Decommissioning Technical Information April 2, 1990.

7.11 USNRC, "Control of Radioactively Contaminated Material," IE Circular No. 81-07, May 14, 1981.

7.12 NUREG/CR 2082, "Monitoring for Compliance with Decommissioning Termination Survey Criteria," June 1981.

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

7.14 USNRC, Draft Regulatory Guide, DG-1005, "Standard Format and Content for Decommissioning Plans for Nuclear Reactors," September 1989.

7.15 USNRC, Draft Regulatory Guide, DG-1006, "Records Important for Decommissioning of Nuclear Reactors," September 1989.

7.16 NUREG-0613, "Residual Radioactivity limits for Decommissioning," Draft Report, October, 1979.

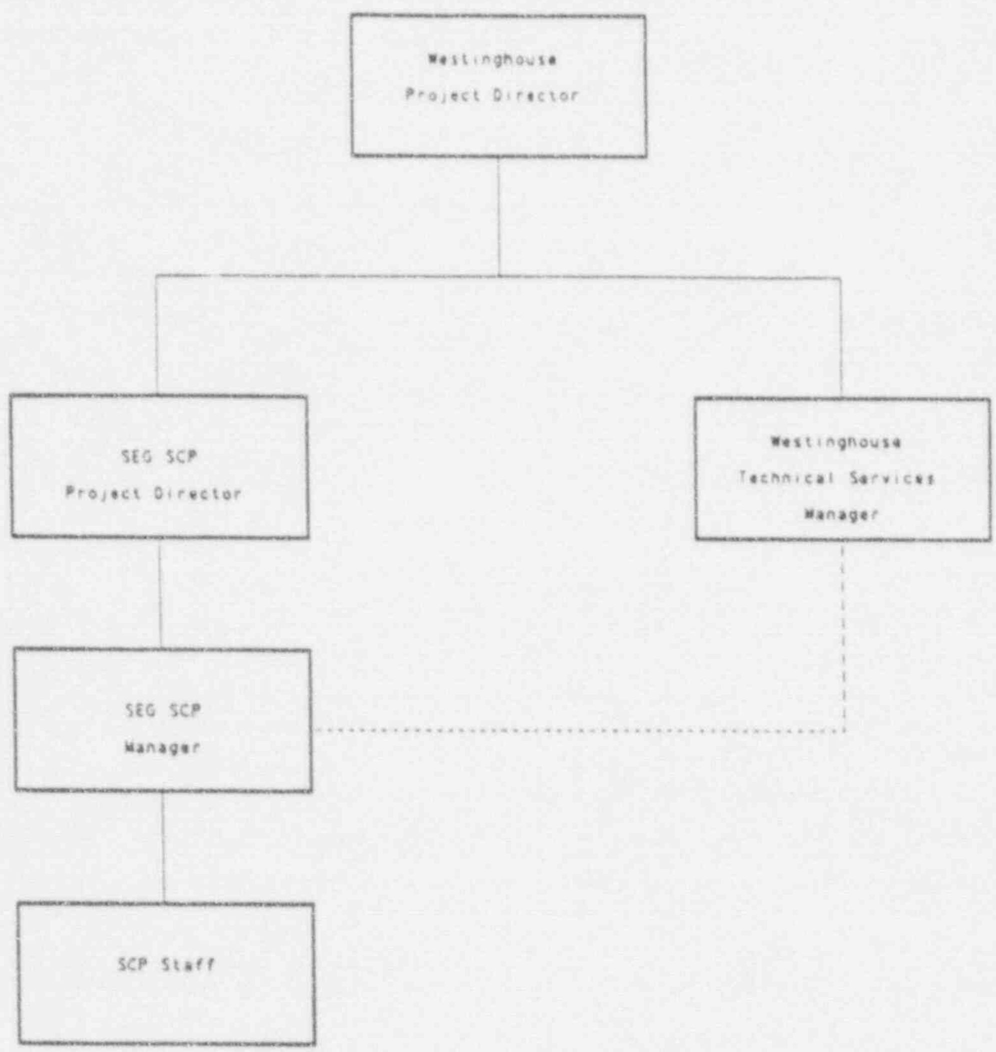
7.17 "Basic Statistics with Business Applications," Second Edition, authors, R.C. Clelland, J.S. deCani and F.E. Brown, published by John Wiley and Sons, Inc., 1966/1973.

7.18 "Handbook of Radiation Measurement and Protection," Allen Brodsky, Editor, published by CRC Press, Inc., 1982.

APPENDIX A

SITE CHARACTERIZATION PROGRAM
ORGANIZATION

SITE CHARACTERIZATION PROGRAM ORGANIZATION



APPENDIX B

SCP IMPLEMENTATION SEQUENCE

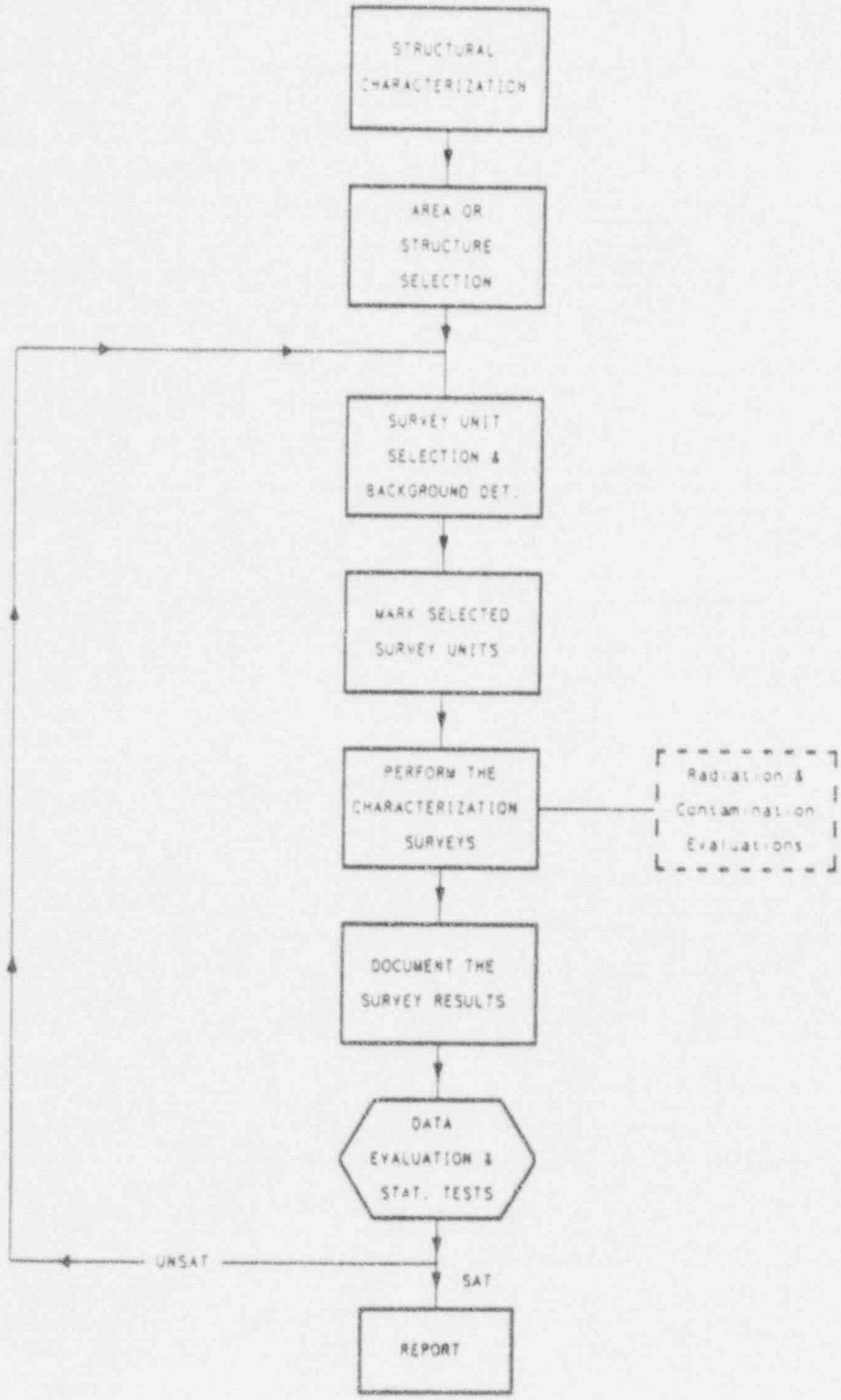
B-1 STRUCTURAL CHARACTERIZATION

B-2 SYSTEM AND EQUIPMENT CHARACTERIZATION

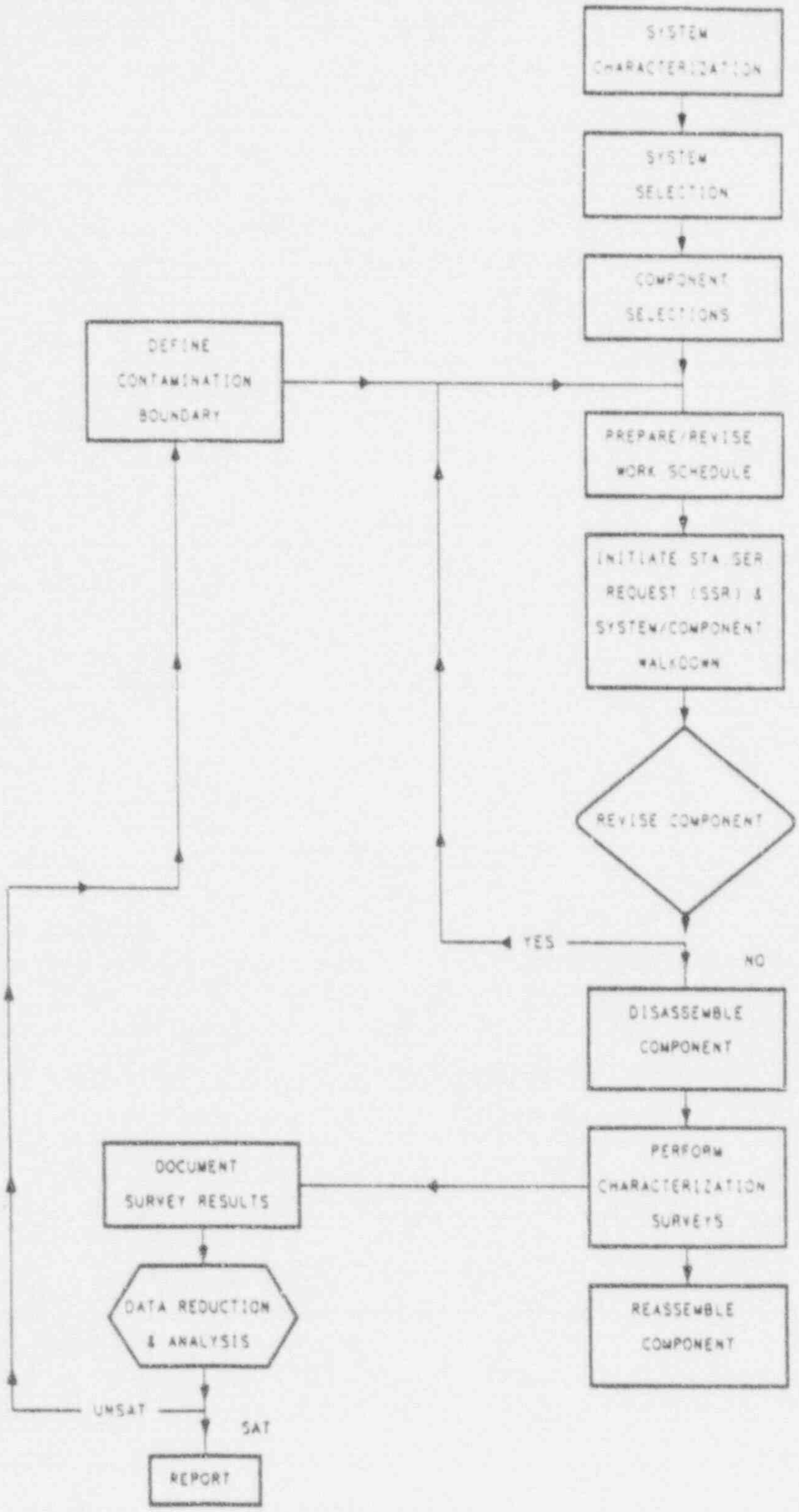
B-3 REACTOR VESSEL AND INTERNALS CHARACTERIZATION

B-4 ENVIRONMENTAL CHARACTERIZATION

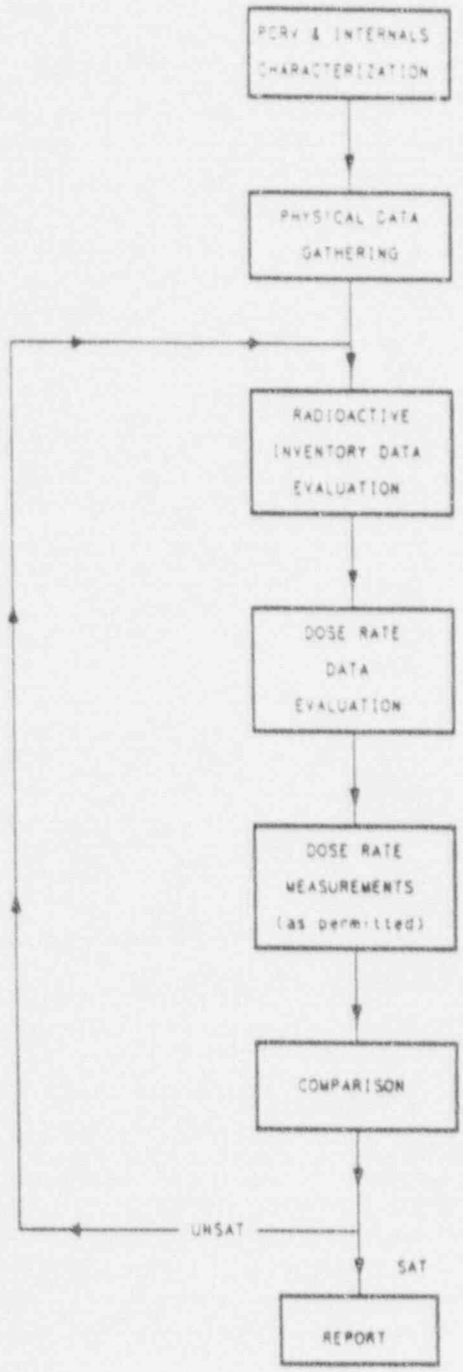
STRUCTURAL CHARACTERIZATION



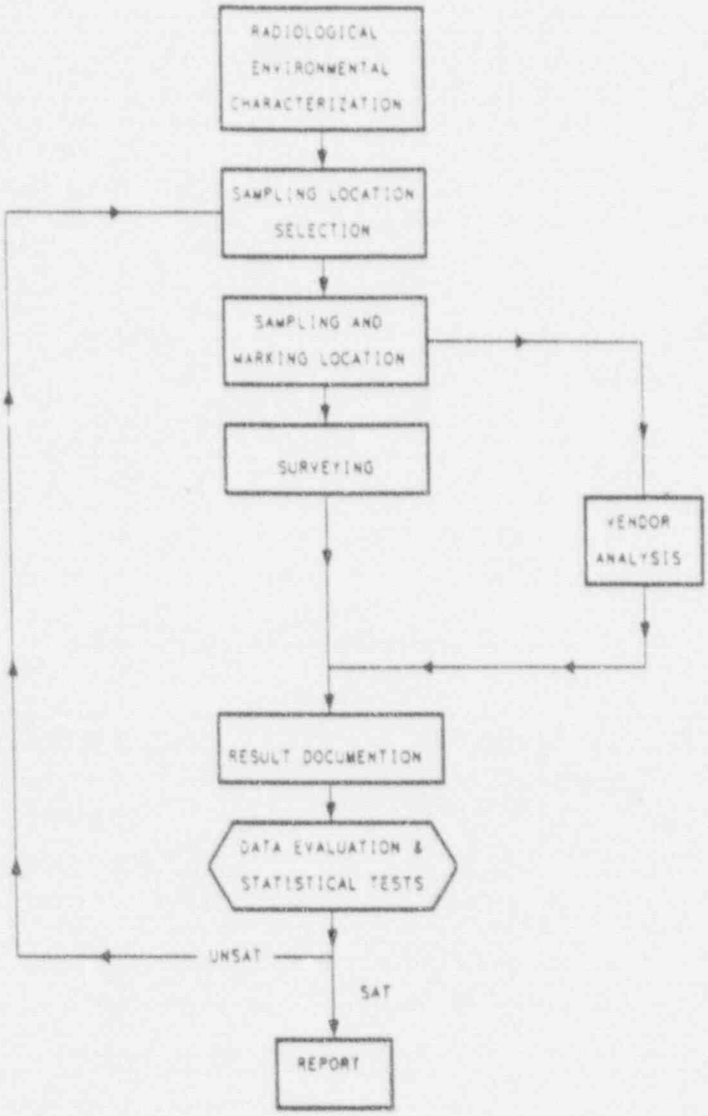
SYSTEM & EQUIPMENT CHARACTERIZATION



PCRV & INTERNALS CHARACTERIZATION



ENVIRONMENTAL CHARACTERIZATION



APPENDIX C

BACKGROUND DETERMINATION

APPENDIX C

BACKGROUND DETERMINATION

Site characterization data will be compared to site background levels. Background will include both "instrument background" and background due to naturally occurring radioactive materials, including enhanced background radiation due to technology (e.g., nuclear weapons tests). Therefore, reliable background data will be obtained for each type of measurement or determination. This background determination will use:

- Gamma radiation measurements
- Direct surface beta/gamma contamination measurements
- Removable beta/gamma and alpha contamination measurements, as applicable (activity in smear surveys)
- Soil sampling

Background data will be obtained by using:

- Unbiased sampling
- Proper number of samples (>30)
- Selection of background sample areas least likely to be affected by the FSV plant
- Use of calibrated and stable instrumentation
- Quality assured results

Of primary importance will be the selection of background sampling areas which closely resemble the materials to be sampled or measured, but which have not been affected by FSV operation. Background data will also be obtained from existing background data from the FSV radiological environmental monitoring program.

Removable surface contamination determinations do not typically have a clean material background other than the counting instrument's background. Removable surface contamination background will be determined by counting an unused smear.

A measurement of background for the total surface contamination determinations will be made for bare concrete, painted concrete, concrete block and painted steel surfaces. Background will be determined from data obtained for at least 30 measurements on each type of material. Surfaces used for the background determinations will be chosen to simulate the construction materials encountered during the characterization as closely as possible.

Materials used for the total surface background determinations will be chosen from site structures at locations that are not likely to have been affected by FSV operation.

These locations will be upwind, upstream and up elevation, as appropriate, to avoid any contamination due to reactor operations. In addition, multiple background measurements may be taken from structures to better obtain representative background levels.

If detectable results are obtained, a "background level," (B), will be determined for each type of measurement. The definition of a "background level" will be based on the assumption that the distribution of background data is either lognormal (the logarithms fit a normal [Gaussian] distribution) or normal. Because the standard deviation of the distribution (either normal or log normal) is not known, the t statistic is used rather than the normal distribution and the coefficient, k, is taken from the attached table (Table 1, Percentage Points of the t Distribution with m Degrees of Freedom (Two-Tailed Probabilities), (References 7.17 and 7.18).

All individual site characterization results less than or equal to B will be considered background and all measurements greater than B will be interpreted as reflecting contamination. B will be determined so that the probability that x (the random variable for the given radiological determination) is less than or equal to B is 95%. Symbolically, this will be expressed as $(x \leq B) = 0.95$. Some measurements less than B could be due to contamination, but there will be background measurements at the same levels. Measurements that are above B will have a small probability ($< .05$) of being background measurements or, equivalently, a large probability of reflecting contamination. However, the land elevation in the Denver-Platteville area is such that "background levels" are higher in this part of the country than elsewhere. In addition, consideration will be given to assessing levels of "background" within and without the FSV plant site.

Once the sample background measurements are made, the natural logarithms of each will be determined and the sample mean, $(\overline{lnx})$, and sample standard deviation, s will be computed:

$$\overline{\ln x} = \frac{\sum \ln x_i}{n} \tag{1}$$

$$s = \sqrt{\frac{\sum [\overline{lnx} - lnx_i]^2}{n-1}} \tag{2}$$

The "maximum likelihood" estimate of ln B is then:

$$lnB_{.95} = [\overline{lnx} + ks] \tag{3}$$

so that B can be estimated from the formula:

$$B_{95} = \exp[\overline{\ln x} + ks] \tag{4}$$

The preceding equation will be used to obtain an estimate of the background level B_{95} for each radiological determination to be made. Where a normal distribution is assumed, the above equations become:

$$\bar{x} = \frac{\sum x_i}{n} \tag{5}$$

$$s = \sqrt{\frac{\sum [\bar{x} - x_i]^2}{n-1}} \tag{6}$$

$$B_{95} = \bar{x} + ks \tag{7}$$

For the 95% single tailed test, we will use the 0.1 Percentage Point Column from Table 1 because we are using a single-tailed test. Thus, if the value of m is 29 because we based our estimate on 30 samples, we would choose k = 1.699 and our equation for B_{95} becomes:

$$B_{95} = \bar{x} + 1.699s, \text{ normal distribution} \tag{8}$$

$$B_{95} = \exp[\ \overline{\ln x} + 1.699s\ ], \text{ log normal distribution} \tag{9}$$

TABLE 1
PERCENTAGE POINTS OF THE t DISTRIBUTION WITH m DEGREES OF FREEDOM (TWO-TAILED PROBABILITIES)*

(Reference 7.17)

m	Percentage Point												
	0.9	0.8	0.7	0.6	0.5	0.4	0.3	0.2	0.1	0.05	0.02	0.01	0.001
1	0.158	0.325	0.510	0.727	1.000	1.376	1.963	3.078	6.314	12.706	31.821	63.657	636.619
2	0.142	0.289	0.445	0.617	0.816	1.061	1.386	1.886	2.920	4.303	6.965	9.925	31.598
3	0.137	0.277	0.424	0.584	0.765	0.978	1.250	1.638	2.353	3.182	4.541	5.841	12.941
4	0.134	0.271	0.414	0.569	0.741	0.941	1.190	1.533	2.132	2.776	3.747	4.604	8.610
5	0.132	0.267	0.408	0.559	0.727	0.920	1.156	1.476	2.015	2.571	3.365	4.032	6.859
6	0.131	0.265	0.404	0.553	0.718	0.906	1.134	1.440	1.943	2.447	3.143	3.707	5.959
7	0.130	0.263	0.402	0.549	0.711	0.896	1.119	1.415	1.895	2.365	2.998	3.499	5.405
8	0.130	0.262	0.399	0.546	0.706	0.889	1.108	1.397	1.860	2.306	2.896	3.355	5.041
9	0.129	0.261	0.398	0.543	0.703	0.883	1.100	1.383	1.833	2.262	2.821	3.250	4.781
10	0.129	0.260	0.397	0.542	0.700	0.879	1.093	1.372	1.812	2.228	2.764	3.169	4.587
11	0.129	0.260	0.396	0.540	0.697	0.876	1.088	1.363	1.796	2.201	2.718	3.106	4.437
12	0.128	0.259	0.395	0.539	0.695	0.873	1.083	1.356	1.782	2.179	2.681	3.055	4.318
13	0.128	0.259	0.394	0.538	0.694	0.870	1.079	1.350	1.771	2.160	2.650	3.012	4.221
14	0.128	0.258	0.393	0.537	0.692	0.868	1.076	1.345	1.761	2.145	2.624	2.977	4.140
15	0.128	0.258	0.393	0.536	0.691	0.866	1.074	1.341	1.753	2.131	2.602	2.947	4.073
16	0.128	0.258	0.392	0.535	0.690	0.865	1.071	1.337	1.746	2.120	2.583	2.921	4.015
17	0.128	0.257	0.392	0.534	0.689	0.863	1.069	1.333	1.740	2.110	2.567	2.898	3.965
18	0.127	0.257	0.392	0.534	0.688	0.862	1.067	1.330	1.734	2.101	2.552	2.878	3.922
19	0.127	0.257	0.391	0.533	0.688	0.861	1.066	1.328	1.729	2.093	2.539	2.861	3.883
20	0.127	0.257	0.391	0.533	0.687	0.860	1.064	1.325	1.725	2.086	2.528	2.845	3.850
21	0.127	0.257	0.391	0.532	0.686	0.859	1.063	1.323	1.721	2.080	2.518	2.831	3.819
22	0.127	0.256	0.390	0.532	0.686	0.858	1.061	1.321	1.717	2.074	2.508	2.819	3.792
23	0.127	0.256	0.390	0.532	0.685	0.858	1.060	1.319	1.714	2.069	2.500	2.807	3.767
24	0.127	0.256	0.390	0.531	0.685	0.857	1.059	1.318	1.711	2.064	2.492	2.797	3.745
25	0.127	0.256	0.390	0.531	0.684	0.856	1.058	1.316	1.708	2.060	2.485	2.787	3.725
26	0.127	0.256	0.390	0.531	0.684	0.856	1.058	1.315	1.706	2.056	2.479	2.779	3.707
27	0.127	0.256	0.389	0.531	0.684	0.855	1.057	1.314	1.703	2.052	2.473	2.771	3.690
28	0.127	0.256	0.389	0.530	0.683	0.855	1.056	1.313	1.701	2.048	2.467	2.763	3.674
29	0.127	0.256	0.389	0.530	0.683	0.854	1.055	1.311	1.699	2.045	2.462	2.756	3.659
30	0.127	0.256	0.389	0.530	0.683	0.854	1.055	1.310	1.697	2.042	2.457	2.750	3.646
40	0.126	0.255	0.388	0.529	0.681	0.851	1.050	1.303	1.684	2.021	2.423	2.704	3.551
60	0.126	0.254	0.387	0.527	0.679	0.848	1.046	1.296	1.671	2.000	2.390	2.660	2.460
120	0.126	0.254	0.386	0.526	0.677	0.845	1.041	1.289	1.658	1.980	2.358	2.617	3.373
∞	0.126	0.253	0.385	0.524	0.674	0.842	1.036	1.282	1.645	1.960	2.326	2.576	3.291

* This table is taken by consent from *Statistical Tables for Biological, Agricultural, and Medical Research*, by Prof. R. A. Fisher and F. Yates, published by Oliver and Boyd, Edinburgh.

APPENDIX D

INTERPRETATION OF UNBIASED DATA

APPENDIX D

INTERPRETATION OF UNBIASED DATA

The unbiased structural surface contamination data will be used to determine which areas of the FSV site have or have not been affected by operation of the plant. It is currently believed that the majority of the FSV structural surfaces may not be contaminated; that is, these surfaces will not exhibit total surface beta/gamma levels, gross removable beta/gamma and alpha levels or gamma exposure rates in excess of those encountered on similar unaffected materials. Therefore, it will be necessary to decide if each set of data from each survey unit differs sufficiently from the accepted background levels to reflect a slight contamination resulting from FSV Plant operations.

1. Comparison of Sample Mean to Background Mean

It is expected that several sets of 30 background observations will not produce the same results. According to the Central Limit Theorem, if a series of observations is made on a population of unknown distribution, with a large enough sample ($n, \geq 30$) the series of means (x's) for each set of observations will be normally distributed (even though the population sampled may not be normal). Therefore, the mean of a background set of observations may be compared to a mean of a similar set of observations made during the characterization process using normally distributed statistics.

Background measurements will be made for each type of determination, on each type of surface material, as described in Appendix C. The mean of the sample data obtained and used for Appendix C background determinations will be used for comparison with the mean of the data obtained for each survey unit. Comparison of the background data mean to each survey unit's data mean will use the following analytical process:

A. Set up null and alternate hypotheses:

- Null hypothesis (H_o): Sample mean = background mean
- Alternate hypothesis (H_a): Sample mean > background mean

B. Calculate standard deviation and standard error for the sample distribution:

Estimated population standard deviation s : $s=\left[\frac{\Sigma(x_i-\bar{x})^2}{n-1}\right]^{1/2}$

n = sample size

$\bar{x}$ = mean of n observations

x_i = each individual observation

Standard error of the mean, $\bar{x}$: $s_{\bar{x}} = \frac{s}{n^{1/2}}$

C. Accept or reject the null hypothesis:

- Test at 95% confidence level, with a one-tail alpha level of 0.05

  Reject H_o if: $\bar{x} \geq 1.699(s_x) + \text{background mean}$

  (1.699 = number of standard deviations corresponding to a one-tailed alpha level of 0.05 for the Student t statistic with 29 degrees of freedom).

If the null hypothesis is rejected, the survey unit will be considered to be contaminated.

2. Examination of Individual Sample Results

Each individual sample result (direct contamination reading, smear, or soil sample) shall be compared to Regulatory Guidance as follows:

A. Subtract medium maximum background, B_{95}.
B. Compare to "average" values in the appropriate Regulatory Guide.
C. For areas where the Regulatory Guide criteria are exceeded, determine extent of the area which is contaminated.

P-91240
Attachment 2

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