March 28, 2003

Mr. G. A. Kuehn, Jr.
Vice President SNEC and
Program Director SNEC Facility
GPU Nuclear, Inc.
Route 441 South
P.O. Box 480
Middletown, PA 17057-0480

SUBJECT: SAXTON NUCLEAR EXPERIMENTAL CORPORATION — AMENDMENT RE: APPROVAL OF THE SAXTON LICENSE TERMINATION PLAN (TAC NO. MA8076)

Dear Mr. Kuehn:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 18 to Amended Facility License No. DPR-4 for the Saxton Nuclear Experimental Facility (SNEF). The amendment consists of changes to the amended facility license in response to your application of February 2, 2000, as supplemented on June 23, August 11, September 18 and December 4, 2000; January 30, February 14, March 15 and 19, June 20, July 2 and September 4, 2001; and January 11 and 24, February 4, May 22 and 28, July 11, August 20, September 17, 23, 24, and 26, October 10, and December 16, 2002.

The amendment revises Amended Facility License No. DPR-4 for the SNEF to annotate approval of the SNEF License Termination Plan.

A copy of the safety evaluation supporting Amendment No. 18 is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA by Patrick M. Madden for/

Alexander Adams, Jr., Senior Project Manager
Research and Test Reactors Section
Operating Reactor Improvements Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket No. 50-146

Enclosures:
1. Amendment No. 18
2. Safety Evaluation

cc w/enclosures:
See next page

Saxton Nuclear
Experimental Corporation

Docket No. 50-146

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March 28, 2003

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Middletown, PA 17057-0480

SUBJECT: SAXTON NUCLEAR EXPERIMENTAL CORPORATION — AMENDMENT RE: APPROVAL OF THE SAXTON LICENSE TERMINATION PLAN (TAC NO. MA8076)

Dear Mr. Kuehn:

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The amendment revises Amended Facility License No. DPR-4 for the SNEF to annotate approval of the SNEF License Termination Plan.

A copy of the safety evaluation supporting Amendment No. 18 is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,
/RA by Patrick M. Madden for/

Alexander Adams, Jr., Senior Project Manager
Research and Test Reactors Section
Operating Reactor Improvements Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket No. 50-146
Enclosures: 1. Amendment No. 18
2. Safety Evaluation
cc w/enclosures:
See next page

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SAXTON NUCLEAR EXPERIMENTAL CORPORATION

GPU NUCLEAR, INC.

DOCKET NO. 50-146

AMENDMENT TO AMENDED FACILITY LICENSE

Amendment No. 18
License No. DPR-4

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that

   A. The application for an amendment to Amended Facility License No. DPR-4 filed by the Saxton Nuclear Experimental Corporation and GPU Nuclear, Inc. (the licensees) on February 2, 2000, as supplemented on June 23, August 11, September 18 and December 4, 2000, January 30, February 14, March 15 and 19, June 20, July 2 and September 4, 2001, and January 11 and 24, February 4, May 22 and 28, July 11, August 20, September 17, 23, 24, and 26, October 10, and December 16, 2002, conforms to the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the regulations of the Commission as stated in Chapter I of Title 10 of the *Code of Federal Regulations* (10 CFR);

   B. The facility will be possessed in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance that (i) the activities authorized by this amendment can be conducted without endangering the health and safety of the public and (ii) such activities will be conducted in compliance with the regulations of the Commission;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. This amendment is issued in accordance with the regulations of the Commission as stated in 10 CFR Part 51, and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by renumbering the existing paragraph 2.C.(3) to 2.C.(4) and by adding a new paragraph 2.C.(3) to Amended Facility License No. DPR-4 to read as follows:

   (3) License Termination

   The licensee shall implement and maintain in effect all provisions of the approved SNEC Facility License Termination Plan as approved in the SER dated March 28, 2003. The licensee may make changes to the SNEC Facility License Termination Plan without prior approval provided the proposed changes do not:

   (i) Involve a change to the Technical Specifications or require NRC approval pursuant to 10 CFR 50.59;

   (ii) Violate the criteria of 10 CFR 50.82(a)(6);

   (iii) Reduce the coverage requirements for scan measurements;

   (iv) Increase the derived concentration guideline level (DCGL), developed to meet the requirements of 10 CFR 20.1402, and related minimum detectable concentrations for both scan and fixed measurement methods;

   (v) Use a statistical test other than the Sign test or Wilcoxon Rank Sum test for evaluation of the final status survey;

   (vi) Increase the radioactivity level, relative to the applicable DCGL, developed to meet the requirements of 10 CFR 20.1402, at which investigation occurs;

   (vii) Increase the Type I decision error; and

   (viii) Decrease an area classification (i.e., impacted to non-impacted; Class 1 to Class 2; Class 2 to Class 3; or Class 1 to Class 3).

3. This license amendment is effective as of the date of its issuance and shall be implemented no later than 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Patrick M. Madden, Section Chief
Research and Test Reactors Section
Operating Reactor Improvements Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Date of Issuance: March 28, 2003

SAFETY EVALUATION BY

THE OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

SUPPORTING AMENDMENT NO. 18 TO

AMENDED FACILITY LICENSE NO. DPR-4

SAXTON NUCLEAR EXPERIMENTAL CORPORATION

GPU NUCLEAR, INC.

DOCKET NO. 50-146

1.0 INTRODUCTION

By letter dated February 2, 2000, the Saxton Nuclear Experimental Corporation (SNEC) and GPU Nuclear, Inc. (the licensee), submitted a request to amend Amended Facility License No. DPR-4, for the SNEC facility. This letter was supplemented by letters dated June 23, August 11, September 18 and December 4, 2000; January 30, February 14, March 15 and 19, June 20, July 2 and September 4, 2001; and January 11 and 24, February 4, May 22 and 28, July 11, August 20, September 17, 23, 24, and 26, October 10, and December 16, 2002. In addition, the NRC staff met with the licensee on May 25 and 30, 2000; August 6, 2001; and April 8, May 8 and 22, June 21, July 31, August 29, and October 31, 2002, to discuss their application. These meetings were open to public observation. During the review of the licensee's application, the NRC staff also conducted site visits. In accordance with the requirements of Title 10, U.S. Code of Federal Regulations (10 CFR) Part 20, Subpart E, and 50.82(a)(9), the licensee submitted a license termination application, accompanied by the facility License Termination Plan (LTP), which has been designated as Supplement 1 to the SNEC Facility Updated Final Safety Analysis Report. Under the provisions of 10 CFR 50.82(a)(10), the U.S. Nuclear Regulatory Commission (NRC) approves LTPs by license amendment subject to such conditions and limitations as it deems appropriate and necessary.

2.0 EVALUATION

In accordance with 10 CFR 50.82(a)(9), the licensee submitted the LTP for the SNEC Facility to the NRC. This section of the regulations requires the LTP to contain the following information: (1) a site characterization; (2) identification of remaining dismantlement activities; (3) plans for site remediation; (4) detailed plans for the final radiation survey; (5) compliance with radiological criteria for license termination; (6) a description of the end use of the site, if restricted; (7) an updated site-specific estimate of remaining decommissioning costs; and (8) a supplement to the environmental report, pursuant to 10 CFR 51.53, describing any new information or significant environmental change associated with the licensee's proposed license termination

activities. In addition, the licensee requested the authority to make certain changes to the LTP once NRC has approved it. The following is the staff's evaluation of the licensee submission.

2.1 Site Characterization

The site characterization survey is the radiation survey conducted to determine the nature and extent of radiological contamination at the site. The purpose of the site characterization survey is: to permit planning for remediation activities; to demonstrate that it is unlikely that significant quantities of residual radioactivity have gone undetected at the site; to provide information to design the final site survey (i.e., identify survey unit classifications for impacted areas); and to provide input into dose modeling. Surveys and sampling conducted during site characterization are based on biased and judgmental measurements, the Historical Site Assessment (HSA), and the scoping survey. According to NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual" (MARSSIM) (Section 2.4.4), if an area could be classified as a Class 1 or Class 2 for the final status survey (FSS), based on the HSA and scoping survey results, a characterization survey is warranted.

Classification is a process by which a survey unit is described according to its radiological characteristics. The HSA for the SNEC site, and Section 2 and Table 5-2 of the LTP, provide an acceptable description of the classifications assigned to all structures and grounds for the site and adjacent areas. Characterization is an ongoing effort throughout the decommissioning process, and survey unit classifications may be modified based on new characterization information or impacts from decommissioning activities. The licensee is allowed to change the classification of a survey unit from a less restrictive classification to a more restrictive classification, based on new information; however, if the licensee decides to change the classification of a survey unit from a more restrictive classification to a less restrictive classification, the licensee will seek NRC approval before implementing this type of classification change (see Section 2.8 of this SER).

In general, the licensee has taken a conservative approach when applying MARSSIM guidance on survey unit classification by designating an area or structure with a more restrictive classification than the characterization data would warrant. Examples are the Saxton Steam Generating Station (SSGS) Intake Tunnel and the Weir Outfall. Additionally, in other cases, such as paved and unpaved roads and sub-pavement soils, the radiological characterization supports a non-impacted classification. However, the licensee has conservatively designated these survey units as impacted, based on HSA information pertaining to the use and history of the area.

In accordance with the provisions of MARSSIM, the licensee conducted an HSA that consisted of an investigative review and compilation of site historical records [e.g., 10 CFR 50.75(g) records, radiological incident files, operational and decommissioning records, and annual environmental reports to NRC]. Personnel interviews were conducted with present and former plant employees, to determine operational events that caused contamination in areas or systems not designed to contain radioactive or hazardous materials. The licensee's determination of the initial classifications of contamination potential was based on the HSA and the site characterization program.

The licensee plans on using a series of surveys to demonstrate compliance with the criteria specified in 10 CFR Part 20, Subpart E, for unrestricted use of the SNEC site, consistent with

the Radiation Survey and Site Investigation (RSSI) Process and the Data Quality Objectives (DQO) Process, as recommended by MARSSIM. Characterization is one of the steps in the RSSI Process that relies on the Data Life Cycle. The Data Life Cycle is the process of planning the survey, using the DQO Process, implementing the survey plan, and assessing the survey results before making a decision. It is the basis for the performance-based guidance in MARSSIM and it is used in an iterative fashion within the RSSI Process. Licensees may use data developed from site characterization as final site survey data, provided these data meet the same DQOs. NRC will review changes made to DQOs as part of NRC's ongoing inspection process.

The SNEC facility was built from 1960 to 1962 on a 4645.8-square meter (m^2) (1.148-acre) site. The pressurized water reactor was operated from 1962 to 1972 primarily as a research and training reactor. According to Section 2.2.1 of the LTP, the facility was shut down in 1972 and all fuel was removed from the containment vessel (CV) and shipped off-site to the Atomic Energy Commission facility at Savannah River, South Carolina, during the same year. After fuel removal, equipment tanks and piping located outside the CV were removed and the control rod blades and the superheated steam test loop were shipped off-site. The building and structures that supported reactor operations were partially decontaminated during 1972 through 1974. Radiological decontamination of reactor support structures and buildings including the Control and Auxiliary Building, the Radioactive Waste Disposal Facility, Yard Pipe Tunnel, Filled Drum Storage Bunker, and Refueling Water Storage Tank (including removal) was performed in 1987, 1988, and 1989, in preparation for demolition of these structures. These buildings and structures were demolished in 1992, after NRC accepted the final release survey, as incorporated in License Amendment No. 11, to Amended Facility License No. DPR-4.

The licensee states, in Section 2.2.1 of the LTP, that, in November 1994, the SNEC Soil Remediation Project was completed. This was a comprehensive project involving soil surveys, sampling, excavation, packaging, and shipment of slightly contaminated site soil. According to the licensee, this program reduced radioactive soil contamination levels over the majority of the site to levels less than the site cleanup criteria for unrestricted use. The project involved extensive surface [0-15 centimeters (cm) (0-5.9 inches)] and subsurface soil sampling in preparation for remediation. The report of this work, titled "1994 Saxton Soil Remediation Project Report," was forwarded to NRC during July 1995.

From 1996 through 1997, the licensee made site preparations to support full-scale efforts for the remaining decommissioning operations. The Decommissioning Support Facility (DSF) was erected south of the CV and was physically connected to the CV. The licensee indicates, in Section 2.2.1 of the LTP, that the site layout has not changed appreciably since that time. NRC approved the start of full-scale decommissioning efforts in April 1998 and operations began in May 1998. Up to that time, the licensee removed selected loose material, spare components, asbestos insulation, and electrical components, with NRC permission. After approval in April 1998, the licensee's main focus of decommissioning efforts was on making all necessary preparations for the removal of the nuclear steam supply system components, namely the reactor pressure vessel, the single steam generator, the pressurizer, and the main coolant pump.

The licensee completed the SNEC Large Component Removal Project on November 22, 1998. This involved the preparation, removal, packaging, shipment, and disposal of the SNEC Facility Pressurizer, Steam Generator, and Reactor Pressure Vessel. All permanent mechanical and

electrical systems and components were removed and shipped off-site for processing/disposal, in accordance with all applicable regulations. According to Section 2.2.1 of the LTP, the only remaining systems are small piping system sections, where they penetrate the CV steel liner and site storm drains. The licensee plans to remove and dispose of the CV piping system remnants before license termination. Site storm drains have been radiologically characterized as discussed in Section 2.2.4.1.8 of the LTP, and will be included in the FSS.

Since May 1999, the licensee has focused on CV concrete remediation work. As a result of extensive penetration of contamination into the various concrete surfaces in the CV, the licensee decided to remove all the interior CV concrete. This concrete was sent to approved off-site disposal facilities in accordance with current requirements. Consequently, the CV steel liner is the only portion of the CV structure remaining. Radiological characterization of the CV was described in Section 2.2.4.1.1 of the LTP.

The SNEC facility was built adjacent to the Pennsylvania Electric Company (PENELEC) -owned SSGS, a coal-fired station placed in operation in 1923. The SNEC facility was constructed without its own turbine-generator. From 1962 until 1972, when the SNEC facility produced usable steam, the steam was sent to the SSGS turbine-generator. The SSGS was deactivated at the end of 1974 and was subsequently demolished between 1975 and 1977. The SNEC facility site property, portions of the adjacent PENELEC property, and a section of the Raystown Branch of the Juniata River (hereinafter referred to as the “river”) have been radiologically impacted as a result of nuclear plant operations. All areas of the SNEC site and the adjacent PENELEC site were classified as either impacted (i.e., Class 1, 2, or 3) or non-impacted, in accordance with MARSSIM guidance. Regarding the SNEC site, all grounds, structures, systems, and components were classified as impacted, with the exception of the exterior section of the CV below elevation 243.11-meters (m) [797.6-feet (ft)] and the soil surrounding and below this elevation which was classified as non-impacted. Sections 2.2.4.1.1, 2.2.4.2, and 2.6 of the LTP provide adequate justification for this non-impacted classification. Impacted portions of the PENELEC site property include the SSGS, segments of the SSGS Intake Tunnel, SSGS Discharge Tunnel, storm drain system, pavement, warehouse, garage, line shack and switchyard buildings, and various open land areas. (After submission of the LTP, the warehouse and garage were demolished leaving only the foundation of the structures.) The intake tunnel from the river intake to the second clean-out [about 133.8-m (440-ft)] and the trash rack and intake screen areas are classified as non-impacted. Section 2.2.4.1.7.1 and Chapter 5 of the LTP provide adequate information to justify the non-impacted classification. According to Section 2.5 of the LTP, the river has been characterized to assess impacts by SNEC facility operations. The river, including surface waters and sediment, is classified as non-impacted, with the exception of the Weir Outfall (Class 2) and the Weir Outfall Buffer (Class 3). The information provided in Chapter 2 of the LTP is adequate to make these classifications.

The types of surveys and sampling that the licensee conducted for radiological characterization include: (1) surface activity measurements on interiors of buildings (surface structures); (2) extensive sampling and surveys to determine potential migration of radionuclide contamination in hard-to-reach or not readily accessible areas (e.g., cracks, crevices, areas beneath buildings, construction joints, etc.); (3) surface activity measurements on interior surfaces of embedded piping; (4) detailed surveys and sampling to supplement information developed during the initial site characterization (i.e., internal surfaces of secondary side systems, activated CV liner steel and concrete, direct radiation measurements, etc.); and 5) surveys and sampling of systems,

structures, and the environment that were not conducted during the characterization [e.g., open land areas, subsurface soil, sediment, groundwater, and structural surfaces potentially contaminated with transuranic (TRU) and difficult-to-measure/hard-to-detect (HTD) radionuclides]. The licensee also used radiological environmental monitoring program data to complement the characterization information, to assess the classification of off-site areas and the possible need for any remediation.

Because of the use of mixed oxide fuel at the SNEC Facility and the history of failed fuel experiments, the licensee has placed special emphasis on the detection of TRU and HTD radionuclides during characterization. Over 200 samples were analyzed for TRU and/or HTD radionuclides. Samples were taken from the SNEC facility, adjacent PENELEC property, and off-site background locations. The licensee used these results to determine the nuclide ratios and mix for the appropriate surrogate DCGL and to plan remediation activities. The extensive analysis performed for TRU and HTD radionuclides has enabled the licensee to focus on those nuclides present as a result of licensed operations, as discussed in Section 6.2.2.3 of the LTP. Table 2-29 of the LTP provides the results of the TRU and HTD radionuclide analysis.

The licensee committed, in Section 2.5 of the LTP, to follow the DQO Process and the SNEC Facility Decommissioning Quality Assurance (QA) Plan, to conduct any further site characterization work, and in Section 5.7, to document, in the survey unit release record, the characterization data, FSS data, and other information needed to demonstrate compliance with the site release criteria. This information and supporting data will be available on site for NRC review during inspections. Additionally in Section 5.7, the licensee commits to provide, in the FSS documentation for each survey unit, a description of any changes in initial survey unit assumptions, relative to the extent of residual radioactivity. NRC plans to review this information as part of its ongoing inspection effort. The NRC staff will review the licensee's FSS documentation to assess whether the licensee adequately used characterization data in the FSS design.

The licensee's characterization process for the site and adjacent areas focuses on structures, systems, and the environs, considering radiological, hazardous, and State regulated-materials. Groundwater and subsurface soil contamination are included in the assessment of the site environs. QA and Quality Control (QC) measures included the appropriate training and qualifications, instrumentation, procedures, records, audits and surveillance, and data collection, for the site characterization program, to ensure data quality in accordance with the SNEC QA Program and for compliance with 10 CFR Part 50, Appendix B.

In accordance with 10 CFR 50.82(a)(9)(ii)(A) and the guidance contained in NRC Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors," the radiological conditions of the site were provided in Chapter 2 of the LTP. Summaries of the most significant radiological events were described in Section 2.2.3 of the LTP. For the final decommissioning operations, most scoping, characterization surveys, and sample collections were performed from May 1994 through 1997. The purpose of this effort was to estimate the extent of contamination on the SNEC site and on adjoining PENELEC property that would require remediation to support decommissioning. The licensee used the results to develop initial decommissioning plans, schedules, and cost estimates. Since 1997, the licensee has continued to use the RSSI process to obtain additional characterization data needed to make decisions regarding remediation and FSS design.

In Section 2.1.1 of the LTP, the licensee notes that supplemental characterization continues to take place and will continue through remediation and during FSS activities. To date, the licensee has extensively characterized the nature and extent of radiological contamination of the SNEC site and, where appropriate, the immediate surroundings.

A list of the estimated total site radionuclide inventory, as of the date of the LTP submittal, was provided in Table 2-1. This table includes the following radionuclides: H-3; C-14; Fe-55; Ni-59; Co-60; Ni-63; Sr-90; Nb-94; Tc-99; Sb-125; Cs-134; Cs-137; Eu-152; Eu-154; Eu-156; U-234; U-235; U-238; Pu-238; Pu-239/240; Pu-241; Am-241; Pu-242; and Cm-243/244. These fission and activation products are typical of those found in pressurized-water reactor plants and are similar to those radionuclides described in NUREG/CR-0130, "Technology, Safety and Costs of Decommissioning a Reference Pressurized-Water Reactor Power Station." Additionally, 30 years of radioactive decay and remediation, since shutdown, have reduced the inventory of radioactive material. The remaining inventory (as of the date of the LTP submittal) consists mainly of contaminated and activated concrete in the CV, low-level contaminated soil in and near the site, and infiltrated water and sediment in two tunnels.

The FSS will be conducted, using guidance in MARSSIM, to demonstrate compliance with the criteria specified in Part 20, Subpart E, for unrestricted release of the SNEC site. The types of surveys and sampling described for complete characterization are acceptable, but will require further NRC staff validation, to ensure that the methodology and data are adequate, as this information becomes available. This validation will occur as part of NRC's ongoing inspection process.

The NRC staff finds the site characterization process acceptable based on the information provided above and because site characterization activities followed the guidance in MARSSIM, USNRC Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors," and NUREG-1727, "NMSS Decommissioning Standard Review Plan." As more characterization information is developed, it will be available on site for NRC staff review, as part of NRC's ongoing inspection process. The NRC staff will review the licensee's characterization plans and supporting reports, as part of NRC's ongoing inspection process, to ensure that the basis for the FSS design and implementation and supporting data are adequate for the licensee to ultimately demonstrate compliance with the requirements of 10 CFR Part 20, Subpart E. In Section 1.3.5 of the LTP, the licensee committed to discuss, with NRC, the planning of final survey activities, sufficiently in advance, to allow the scheduling of inspection activities.

The initial site classification for the SNEC site and adjacent areas began in 1997 and has expanded during subsequent site characterization activities, following the MARSSIM guidance. The licensee will evaluate area classifications throughout the dismantlement and decommissioning process as radiological conditions change and additional information and data are obtained. In accordance with the LTP, the licensee will finalize the classification of each survey unit during the development of the FSS package for that survey unit. The final classification designation and verification will be included, along with the characterization data, classification history of the unit, and a summary of the FSS design elements in the FSS report for each survey unit. NRC will examine the rationale for assumptions, classification designations, and characterization, before FSS implementation. The FSS report for each survey unit will be submitted to NRC for review. These reports will serve as the basis for NRC terminating Amended Facility License No. DPR-4.

2.1.1 Facility Radiological Status

As described in Section 2.2.1 of the LTP, the SNEC facility permanently shut down after approximately 10 years of operation. Operations ceased and all fuel assemblies were permanently removed from the CV in 1972 and shipped to the Atomic Energy Commission (now U.S. Department of Energy) facility at Savannah River, SC., which remains the owner of the fuel. After cessation of operations, the licensee began to decommission the SNEC facility. Air and liquid effluents were reported to NRC on a regular basis. The licensee reviewed operational events, corrective action system documents, and licensee event reports to determine the facility's radiological history. The licensee conducted oral interviews with plant personnel, to develop a clear understanding of the radiological status of the site. The HSA and various characterization reports were also developed to support the RSSI process. Further, the licensee maintains decommissioning records in accordance with 10 CFR 20.2103(a) and 10 CFR 50.75(g).

The NRC staff finds the facility radiological status acceptable based on the information described above and because such status was determined in accordance with the guidance provided in MARSSIM and NUREG-1727.

2.1.1.1 Structures

In Sections 2.2.4.1 through 2.2.4.1.7.1 of the LTP, the licensee discusses the extent and nature of contamination for structures on the SNEC site and adjacent PENELEC property. Structures were surveyed to determine general area and contact exposure rate values, as well as loose and total surface contamination. The licensee took core bores to evaluate the extent of penetration of contaminants into porous surfaces such as concrete and to assess activation levels. The surveys employed both systematic location selection and bias selection. Bias sample locations focused on those areas most likely to contain residual radioactivity.

The licensee states in Section 2.2.4.1.1 of the LTP that the SNEC facility CV is the only remaining original structure on the SNEC facility site. CV concrete decontamination began in 1999 and continued through 2000. As a result of extensive penetration of contamination into the various concrete surfaces in the CV, the licensee determined, in 2000, that complete removal of the interior CV concrete would be required, to comply with the release criteria. Additionally, the licensee plans to remove the upper CV dome and shell to approximately 2.1-m (7-feet) below grade, before license termination. Thus, the licensee plans to conduct an FSS on only the remaining below-grade portions of the CV steel liner.

Before deciding that complete removal of the interior CV concrete would be required, the licensee had characterized the general area and contact radiation, along with contamination levels in areas of the CV. Also, characterization was performed to evaluate the nature and extent of fixed and loose surface contamination on the CV concrete surfaces. As part of this effort, numerous concrete core bores were obtained and evaluated. However, based on the decision to completely remove the CV concrete, it is noted, in Section 2.2.4 of the LTP, that the characterization provided is no longer applicable to the FSS design.

Consequently, the licensee will continue to characterize the interior steel surfaces of the CV liner. With concrete removal complete, the only portion of the CV structure remaining will be the CV steel liner. Based on characterization and HSA information, FSS data for CV surface

contamination and activation of the CV steel liner (activation is discussed below in Section 2.1.1.3) will be important. Radionuclide analysis has identified cesium-137 (Cs-137) in excess of the DCGL for building surfaces as the principal surface contaminant on the inner CV steel liner. All surfaces within the CV liner are designated as Class 1 survey units. As more characterization information is developed, NRC will review it as part of NRC's ongoing inspection process. Furthermore, in reviewing the FSS reports submitted by the licensee, and NRC confirmatory results for the CV steel liner, NRC will evaluate the related results, and any concerns may be handled as inspection follow-up items.

The licensee has determined the exterior section of the CV below elevation 243.1-m (797.6-ft) is non-impacted. This portion of the CV exterior is below grade and has not been surveyed directly. The licensee states, in Section 2.2.4.1.1 of the LTP, that contamination of this area could only have resulted from contact with contaminated adjacent soil. A justification for the non-impacted classification of this area is provided in Section 2.2.4.2, "Soil," of the LTP. The staff's evaluation of this information is presented in Section 2.1.1.4, "Surface and Subsurface Soil," of this SER.

As part of the decommissioning process, the DSF (a "Butler-type" prefabricated building) was constructed on a concrete slab located adjacent to the south side of the CV. The DSF consists of three sections: the Decommissioning Support Building (DSB); the Material Handling Bay (MHB); and the Personnel Access Facility. An opening was cut between the CV and the MHB, to facilitate removal of components for packaging and shipping. Additionally, radwaste is handled and packaged in the DSF. Tables 2-6 and 2-6a of the LTP contain the characterization data for this structure, including exposure rate surveys, direct frisks, and beta/gamma and alpha smears. Based on this information and the DSF operating history, the licensee has designated: the floors and walls, up to 2-m (6.6-ft), as Class 1 impacted areas; upper walls and ceilings as Class 2 impacted areas; and the building exterior as Class 3 impacted areas. Table 5-2 of the LTP, providing initial classifications of site areas, also indicates that the Carport slab and roof/ceiling of the DSB will be designated as Class 2 and Class 3 areas, respectively. The licensee indicates, in Section 2.2.4.1.2 of the LTP, that, before licensee termination, this building will either be remediated and included in the FSS, or demolished, and removed from the site.

Buildings surrounding and impacted by the SNEC facility, on the adjoining PENELEC property, are the PENELEC Line Shack, Garage, Warehouse, and Switchyard Building. These structures were not directly associated with the operation of the SNEC facility, but the licensee used them for storage, staging, and other such activities. According to the licensee, these structures (except for the Switchyard Building) were surveyed during 1988, but further characterization was performed, since decommissioning activities since then may have impacted them. Direct and smear surveys for surface contamination were performed in each structure. No surface contamination greater than 1000 disintegrations per minute (dpm)/100 square centimeters (cm^2) beta/gamma or 20 dpm/100cm^2 alpha was detected. Laboratory analysis of building structural materials showed an elevated level of naturally occurring radioactive materials in these building materials. The licensee has identified areas of these structures as Class 2 or Class 3 for the FSS and does not expect to remediate these structures. The licensee indicates, in Section 2.2.4.1.3 of the LTP, that current plans are to demolish the PENELEC Garage and Warehouse, before conducting the FSS. (As confirmed during a NRC site visit, the demolishment of these buildings has been completed.)

According to Section 2.2.4.1.4 of the LTP, the SSGS Discharge Tunnel was contaminated as a result of radioactive liquid effluent discharges from the SNEC Facility. This concrete tunnel, approximately 213.4-m (700-ft) long, was the routine discharge point for liquid radioactive effluents. To adequately survey this below-ground structure for characterization and final release, the licensee removed groundwater that seeped in, and several centimeters (inches) of silt on the floor. Several piping sections were removed from this structure as Cs-137 and Co-60 levels exceeded their respective DCGLs. One of the removed pipes contained an internal surface deposition having 178 becquerel (Bq) per gram (g) [4800 picocuries (pCi)/g)] of Cs-137 and 1.1 Bq (30 pCi/g) of Co-60. Table 2-3 of the LTP lists some of the sediment and water sample results. Tables 2-3e through 2-3g of the LTP summarize more recent characterization information, including concrete rubble and core-bore information.

The first 45.6-m (150-ft) of the floor of the Discharge Tunnel from the point of entry near the SNEC CV is designated as a Class 1 area, because of elevated radionuclide concentrations in the sediment [1 Bq/g (27 pCi/g) of Cs-137] that were removed from the floor surface in this area. The middle section of the floor [about 71.6-m (235-ft)] is designated as Class 2 and the last section [about 96 m (315 ft)] up to the river discharge point is designated as a Class 2 because Cs-137 sediment and core-bore results were less than the DCGL. Based on a concrete rubble sample having 0.05 Bq/g (1.4 pCi/g) of Cs-137, taken from the ceiling near the Seal Chambers (which were SSGS and nuclear plant release points), the licensee designated the first 45.6 m (150 ft) of ceiling as a Class 2 survey area, however, the remainder of the ceiling is a Class 3 survey area. The walls in the first 45.6-m (150-ft) of the tunnel are Class 1 survey areas, because of elevated levels of Cs-137 contamination present, particularly at the Seal Chamber openings. Results of two core bores for Cs-137 were 0.7 Bq/g (18.4 pCi/g) and 1.2 Bq/g (31.5 pCi/g). Based on characterization survey results, the last 167.6-m (550-ft) of walls to the river discharge point are listed as a Class 3 survey area.

Regarding the Seal Chambers, sediment and debris samples indicated elevated levels of contamination. The highest activity found was 0.63 Bq/g (17.1 pCi/g) of Cs-137; however, Co-60 was not detected. Of four concrete core-bore samples taken at non-biased locations, the highest Cs-137 concentration was 0.01 Bq/g (0.3 pCi/g) and no Co-60 was detected. According to the licensee, Cs-137 did not penetrate greater than 1.27-cm (0.5-inch) into the concrete surface. Another sample, a piece of rubberized concrete, had an activity of 0.05 Bq/g (1.3 pCi/g), yet no detectable Co-60. All smears taken in this area indicated less than 1000 dpm/100 cm^2 (beta/gamma). In view of this characterization, the licensee has designated the Seal Chambers as a Class 1 impacted area. Other related structures that the licensee characterized, which will be included in the FSS design, are the 45.7-cm (18-inch) tie line between the Discharge and Intake Tunnels, and the Spray Pump Pit, which supplied Discharge Tunnel water to the Spray Pond area.

The SSGS Intake Tunnel, a 1.8-m (6-ft) by 2.4-m (8-ft) concrete structure about 198.1-m (650-ft) in length, provided river water to the SSGS, and no discharges to the river were made via this pathway. However, during freezing weather, warm water from the Discharge Tunnel was diverted into the Intake Tunnel, using the Spray Pond supply piping, to prevent ice formation on the intake tunnel screen wash and filtration system components. The licensee states, in Section 2.2.4.1.7 of the LTP, that this process would have provided a mechanism for low level radioactivity to enter the Intake Tunnel. Tables 2-26 and 2-29 of the LTP provide the results of an extensive characterization study, and TRU and HTD radionuclide analysis, respectively, for the Intake Tunnel.

Of the 174 sediment samples taken throughout the Intake Tunnel, 142 samples showed detectable concentrations of Cs-137. The highest sample was 0.07 Bq/g (1.8 pCi/g), taken at the 25.9-m (85-ft) Mid-Section of the North Wall. The average Cs-137 value was 0.02 Bq/g (0.46 pCi/g). None of these samples had detectable quantities of Co-60. Fourteen concrete core-bore samples obtained throughout the tunnel showed no detectable quantities of SNEC-plant-derived radionuclides. To assess loose surface contamination, 365 smears were obtained throughout the tunnel, one smear for every 9.3-m^2 [(100 square feet-(ft^2)] of surface area. All smears were less than 1000 dpm/100 cm^2 beta/gamma, and no alpha contamination was detectable.

The licensee states, in Section 2.2.4.1.7.1 of the LTP, that surface scans of the Intake Tunnel were conducted, using an E-140N with an HP-210/260 Probe. Scan measurements were obtained every 3-m (10-ft) of tunnel length, with each scan covering about 0.09-m^2 (1-ft^2) of surface area. All surface scan survey results were less than 100 cpm (counts per minute) (net). In addition, static measurements, using a Bicron MicroRem meter to assess dose rates, were obtained throughout the tunnel, approximately every 3-m (10-ft), at 0.9-m (3-ft) from the floor. Intake Tunnel dose rates ranged from 2 microroentgen per hour (μR/hr) to 4 μR/hr.

Based on the Intake Tunnel characterization results, the licensee concluded that the tunnel from the river intake to the second clean-out [about 121.9-m (400-ft)] is non-impacted. The licensee also concluded from these results that the trash rack and intake screen areas are non-impacted. The balance of the intake tunnel floors and walls are classified as a Class 2 area, whereas the ceiling is a Class 3.

The SSGS was also impacted by SNEC facility effluent discharges, as indicated in Section 2.2.4.1.5 of the LTP. Today, only the SSGS footprint remains; the above-grade structure was demolished from 1975 to 1977. The structure remaining, at and below-grade, is comprised of three levels that exhibit different amounts of contamination. The upper level concrete slab, the "Boiler Pad," was found to be free of residual contamination, based on surveys and sampling that included smears and core bores. However, the licensee conservatively designated this slab as a Class 3 impacted area because the slab is located at grade surface within a larger Class 1 land area. Additionally, an elevated Cs-137 activity was found in a sediment sample taken from a floor drain below the Boiler Pad. The next lower level, the "Firing Aisle," was found to have surfaces contaminated with detectable levels of SNEC-plant-derived radionuclides. Drain lines in this area had detectable levels of Cs-137 as high as 0.11 Bq/g (3.1 pCi/g). The licensee indicated that the majority of this piping will be removed and that any impacted piping remaining will be included in the FSS design. For implementation of the release criteria (i.e., DCGLs) for buried pipes that remain after the SSGS is backfilled, see Sections 2.5.3.2 and 2.5.4 of this SER. Surveys and sampling results for the lowest level, the "Basement Area," warranted that the floor and east end walls be designated as Class 1 survey units. Based on the characterization results, the licensee classified the remaining walls of the basement as Class 2 for areas up to 2-m (6.6-ft), and Class 3 for areas above 2-m (6.6-ft). Characterization results for the SSGS area are presented in Tables 2-3a through 2-3d and 2-29 of the LTP.

The NRC staff has determined that the approach the licensee used to characterize structures is acceptable because the licensee generally followed the guidance in MARSSIM and NUREG-1727. However, the NRC staff, through the inspection process, will review any additional characterization data, to ensure that they are adequate to design the FSS.

2.1.1.2 Systems

The licensee states, in Section 2.2.4.1.8 of the LTP, that only the complex Site Storm Drain System was characterized, because it is the one system that will remain and be included in the FSS. This system collects surface water and building drains from structures in the PENELEC property and directs it to the river. Segments of the SSGS supporting the Yard Drainage System were demolished, along with the SSGS, over 25 years ago. However, several sections of underground drainage piping still exist in the South and West sides of the SSGS in-ground structure, and continue to channel rain water and site run-off away from the site. According to the licensee, drainage systems surrounding the CV area have largely been removed as a result of the excavation of contaminated soils in the vicinity of the CV, including the Weir System to the river, in its entirety. Also, the licensee has excavated a Septic System Drain Field on the South side of the PENELEC Warehouse.

In Section 2.2.4.1.8.5 of the LTP, the licensee states that it believed the drain system was impacted based on the HSA and the soil contamination on-site. Inspection and sampling of the remaining segments of the SSGS Yard System Drainage piping were performed in two phases. The initial phase involved an investigation using robotics and video camera equipment to probe and examine piping segments and establish their interconnections. The investigation phase also located access points and established that the water flow patterns from these systems were away from the site and toward the river. Based on these findings, the licensee decided that a thorough investigation and sampling of remaining underground piping systems should be performed to rule out the possibility that these systems had introduced elevated levels of radionuclide contamination into the environs. Section 2.2.4.1.8.1 of the LTP discusses the specific locations of PENELEC property drainage lines, pipe sections, Yard Drain openings, and sumps, including their different points of connection along the Shoup Run Shunt Line. All the Yard Drain Piping sections are depicted in Figure 2A-1 of the LTP. This piping system is located under open land areas already classified as impacted Class 2 or Class 3.

The investigation phase also involved taking grab samples, at Yard Drain Piping access points, of materials that had collected in pipe sections since plant shutdown. Contamination levels for Cs-137 ranged from non-detectable up to 0.24 Bq/g (6.4 pCi/g) at the floor drain rim for Garage Bay Number 4 whereas the TRU analysis for the rim sample indicated no detectable activity. The licensee states, in Section 2.2.4.1.8.3 of the LTP, that this elevated level of Cs-137 may have been the result of radiological work performed in the PENELEC Garage during previous site remediation efforts. Co-60 results for all samples were below the minimum detectable activity.

The second phase involved a more rigorous sampling and measurement effort for the piping system. Measurements were made, over accessible lengths of pipe, using an in-situ sodium iodide (NaI) detector, and scale/sediment samples were taken from each piping system. Surface contamination levels ranged from non-detectable up to 910 dpm/100 cm^2 for gamma emitters, and sample results for Cs-137 contamination were found to be no greater than 0.05 Bq/g (1.4 pCi/g). Characterization results from this phase are summarized in Table 2-5b of the LTP. Information regarding the characterization of embedded and Yard Drain Piping is found in the investigation report referenced in Section 2.2.4.1.8.5 (Reference 2-38) of the LTP. Based on the HSA and characterization effort, the licensee has removed the majority of the Yard Drain Piping System.

According to the licensee, the results confirmed that the Yard Drain Piping System is below the DCGLs for releasing the site. As such, the licensee states, in Section 2.2.4.1.8.4 of the LTP, that no further actions are planned for the FSS since there were no significant findings in these systems. In other words, this piping will not be resurveyed as part of the FSS. For implementation of the release criteria (i.e., DCGLs) for buried pipes that remain after the SSGS is backfilled, see Sections 2.5.3.2 and 2.5.4 of this SER.

The NRC staff will review the FSS packages for the Yard Drain Piping System to ensure that the Data Quality Objectives/Assessment are consistent with the volumetric DCGLS for such systems as described in Section 6.0 of the LTP. Furthermore, in reviewing the FSS reports submitted by the licensee and NRC confirmatory results for this system, NRC will evaluate the related results and any concerns may be handled as inspection follow-up items through the inspection process.

The NRC staff has determined that the methodology the licensee used to characterize potential internal surface contamination of systems is acceptable because it is consistent with MARSSIM.

2.1.1.3 Activation

The licensee has committed to characterize areas of the CV steel liner that have the potential to have been exposed to neutron flux. The licensee indicates, in Section 2.2.4.1 of the LTP, that the activation zone of the CV steel liner is centered horizontally at the site of closest approach of the former SNEC reactor vessel. The licensee selected 21 locations on the CV steel liner, for the purpose of collecting samples. The area sampled starts at about the 241-m (791-ft) elevation and covers a 6.1-m by 3.1-m (20-ft by 10-ft) area.

Four steel samples had measurable quantities of Co-60, ranging from 0.08 Bq/g (2.06 pCi/g) to 0.13 Bq/g (3.43 pCi/g). Only one sample had contained measurable Ni-63 at 0.75 Bq/g (20.23 pCi/g). The sampling results are provided in Table 2-33 of the LTP. Figure 2-33 of the LTP shows the sample locations and Co-60 results. Additionally, the activation results reported in this table include the presence of Cs-137, which indicates that Cs-137 is present as a surface contaminant on the inner CV steel liner. Cs-137 surface contamination ranged from 0.05 Bq/g (1.37 pCi/g) to 6.7 Bq/g (180 pCi/g). With the completion of the CV concrete removal process, the licensee plans to take additional samples and measurements of this area for more extensive examination. Because characterization is still ongoing, the licensee is unable to provide specific radionuclide concentrations, at this time, that will be used in Equation 6-1 of the LTP, to demonstrate that exposure to the activated metal is less than the release limit of 0.25 millisievert (mSv) [25 milliroentgen-equivalent-man (mrem)] per year. Therefore, as part of the inspection process, NRC will review the licensee's activation assessment for inclusion in the FSS design for the CV steel liner.

The NRC staff has determined that the licensee's activation characterization strategy is acceptable because it is consistent with MARSSIM.

2.1.1.4 Surface and Subsurface Soil

The licensee notes, in Section 2.2.4.2 of the LTP, that, in addition to the CV, contaminated soil in and around the SNEC site will require remediation. Soil at the site is contaminated from licensed operations, accidental spills, and long-term accumulation of material in the soil from

effluent releases. It is also contaminated from subsurface concrete or other structural debris such as pavement, masonry, or structural steel. These events are discussed in the HSA.

The 1994 SNEC Soil Remediation Project (Reference 2-3 of the LTP) involved extensive surface (0-15 cm [0-5.9 inches]) and subsurface soil sampling, in an effort to reduce Cs-137 levels to less than 0.037 Bq/g (1 pCi/g) average. The licensee stated, in Section 2.2.4.2 of the LTP, that, although this project achieved its goal, contaminated soil near the CV and surrounding support tunnel could not be removed until these structures were removed. Also, soil and ground water conditions near the surface prevented an assessment of soil contamination below about 0.9-m (3-ft) deep, in these areas. Subsequently, during 1999, the licensee conducted a major surface and subsurface soil sampling program, to survey the areas not covered by the 1994 soil project, and to investigate potentially impacted areas identified by the HSA. Random and biased sample locations were selected, based on the HSA and previous survey results. Sampling locations covered both the SNEC facility and adjacent PENELEC property. According to the licensee, Cs-137 was the only nuclide, attributed to licensed operations, which was detected. The 55 surface sample results are reported in Table 2-14 of the LTP, and their locations are shown on Figures 2-13 and 2-14 of the LTP. Cs-137 activities ranged from non-detectable to 0.08 Bq/g (2.1 pCi/g).

For the 1999 program, the licensee also took 42 subsurface samples from predominantly biased areas where below-grade tanks, piping, ducts, spills, and structures were once present. To complement these samples, gamma bore logging was conducted at these subsurface locations. Based on the results of the subsurface sampling and gamma logging (Tables 2-15 and 2-16 of the LTP, respectively) the licensee chose to remediate soil to a depth of at least 3-m (10-ft) on the north side of the CV. The licensee has committed not to use gamma bore logging as a stand alone technique for characterization or FSS, but rather as a complement to sampling.

The CV Pipe Tunnel originally surrounded the CV, with the top of the concrete tunnel reaching a maximum below-grade depth of 3-m (10-ft). Leaking pipes inside the tunnel resulted in contaminated water penetrating the seam between the Tunnel floor and wall sections, and at other structural defects; this caused soil contamination at selected locations below and adjacent to the floor. The licensee has removed the majority of the Tunnel, to characterize and remediate the surrounding soil. Soil covering an area of about 1300-m^2 (14,000-ft^2), up to about 4.9-m (16-ft) below grade, has been excavated around the CV and Tunnel. Characterization information is provided in Tables 2-27, 2-29, 2-30, and 2-31 of the LTP, and Figures 2-29, 2-30, and 2-32 of the LTP show the excavated areas around the CV and sampling locations. According to Section 2.6.2 of the LTP, the CV Pipe Tunnel (i.e., the remaining section of the tunnel that supports the MHB floor) is scheduled to be completely removed before the FSS.

Based on the results of soil sampling, the licensee may need to remediate some additional soil around the CV. The licensee, in order to justify the classification of the backfill surrounding the CV below the 243.1-m (797.6-ft) elevation, and under the CV, as non-impacted, conducted an extensive characterization and sampling project in this area. The licensee obtained and analyzed 857 samples from 112 locations around the CV. Depths of these samples ranged from the surface to 45.6-m (150-ft) deep. Sample media included soil, soil-like materials, bedrock, groundwater, and concrete, from the exterior CV saddle. Of the 857 samples analyzed, 35 of those detected positive activity. Of those 35 positive results, only five indicated

Cs-137 above background. These five ranged from 0.02 Bq/g (0.6 pCi/g) to a high of 0.03 Bq/g (0.9 pCi/g), all well below the applicable DCGL. No positive results above background were detected greater than 3-m (10-ft) below the surface being sampled. A complete listing of the analysis results is given in Table 2-30 of the LTP, whereas Table 2-31 of the LTP provides a listing of all positive results. In addition, Figures 2-34 and 2-35 of the LTP indicate the boundary of the non-impacted region under the CV and the geophysical boundaries; both figures were drawn to scale. The NRC staff finds the licensee's extensive subsurface soil characterization (including the groundwater data for the CV angle well discussed in Section 2.1.1.5 of this SER) acceptable to classify the subsurface soil around and under the CV, and the exterior section of the CV below elevation 243.1-m (797.6-ft) as non-impacted because no radioactive material above background was detected 3-m (10-ft) below the surface being sampled.

TRU radionuclides and Sr-90 were positively identified by off-site laboratory analysis in several samples from the CV excavation area. SNEC sample number SX5SD99202 was taken at a depth of 1.2-m (4-ft) to 1.8-m (6-ft) within the CV North Yard area. This sample contained Am-241 at a concentration of 0.0004 Bq/g (0.012 pCi/g). Another North Yard area sample that was collected from soil bag number 34L (packaged for disposal), contained a combined TRU concentration of approximately 0.007 Bq/g (0.2 pCi/g) and exhibited an Sr-90 concentration of 0.01 Bq/g (0.27 pCi/g). Finally, a sample of sediment from within the CV Pipe Tunnel (before remediation), contained Sr-90 at a concentration of about 0.36 Bq/g (9.7 pCi/g). The latter two sample materials both contained measurable amounts of Cs-137 and Co-60 as well. Table 2-29 of the LTP provides a listing of all TRU and HTD radionuclide results.

The licensee also conducted investigations of soils at several locations in the vicinity of the SSGS Discharge and Intake Tunnels and the SSGS area. Section 2.2.4.1.6 of the LTP indicates that there was no evidence of elevated contamination above that which results from natural background radiation. Soils removed in the vicinity of the Discharge Tunnel during the investigations contained only background levels of radionuclides, associated with plant operations. The results of this investigation effort are reported in Table 2-3i of the LTP.

Throughout the SNEC soil characterization campaign, soil samples were analyzed on-site using gamma spectroscopy. Additionally, Section 2.2.4.2 of the LTP states that selected soil samples were routinely sent to an off-site laboratory for a more complete analysis, to support remediation efforts. At the off-site laboratory, samples were analyzed using radiochemical and liquid scintillation techniques, and gamma and alpha spectroscopy, under an audited QA program. The off-site analytical laboratory's minimum detectable concentrations (MDCs) for the alpha, beta, and gamma spectroscopy instruments used to quantify radionuclide concentrations were consistent with MARSSIM recommendations.

Based on the licensee's extensive characterization of site soils, the surface areas and subsurface to 1-m (3.3-ft) deep below the current excavation surrounding the CV have been classified as Class 1 survey areas. Table 5-2 of the LTP provides the survey classifications that result from the characterization data. The licensee acknowledges that continued remediation in selected areas will ensure that these areas satisfy the unrestricted release criteria before the FSS process begins.

The NRC staff finds the licensee's surface and subsurface soil characterization strategy and initial classification process acceptable because it is generally consistent with the approach in MARSSIM and NUREG-1727.

2.1.1.5 Groundwater

The NRC staff has evaluated whether groundwater on the SNEC Facility site contains plant-generated radionuclides. This evaluation is based on SNEC's LTP and supporting documents, and on NRC's independent assessment.

2.1.1.5.1 Background

The SNEC Facility and adjacent PENELEC property comprise approximately 60.7 hectares (150 acres) in Saxton, Pennsylvania. The SNEC site and PENELEC property lies in the Ridge and Valley physiographic province within the Appalachian highlands.[1] This province contains alternate successions of ridges and valleys that trend generally to the northeast. The harder, more-resistant-to-erosion, sedimentary rocks form the ridges surrounding the site, whereas the softer, less-resistant-to-erosion, sedimentary rocks form valleys. This SNEC site and PENELEC property comprises one of these valleys where surface water erosion along structural features has cut into the sedimentary rocks and where alluvial deposition has deposited sedimentary rocks during lower-energy (water-energy) periods. The ridges surrounding the SNEC site and PENELEC property (Allegrippis Ridge to the west and the Terrace Mountain to the east) and bedrock beneath the alluvial Pleistocene and recent deposits at the SNEC site and PENELEC property are of the Devonian age.[2] Thus, there is a significant erosional unconformity that represents millions of years between the upper alluvial materials and the underlying bedrock materials.

The SNEC site and PENELEC property occupies a valley that represents the flood plain and terraces of the Raystown Branch of the Juniata River. Approximately 54.4 kilometers (km) [34 miles(mi)] downstream from the site is Raystown Lake, which was formed when the river was dammed to provide recreational activities, flood control, and water-quality enhancements. The conservation pool level of the lake reaches 43.4 km (27 mi) upstream from the dam. The average elevation of the river near the SNEC site and PENELEC property is 242.0-m (794-ft) mean sea level (MSL), while the average elevation of the SNEC site and PENELEC property is about 247.1-m (811-ft) MSL. Another surface water feature impacting the SNEC site and PENELEC property is Shoup's Run, a small channelized stream that flows west into the Raystown Branch of the Juniata River. The SNEC site and PENELEC property is bounded on the north and west by the Raystown Branch of the Juniata River and on the south by Shoup's Run. The watershed of the Raystown Branch of the Juniata River extends approximately 1933.5 square kilometers (km^2) [756 square miles (mi^2)] upstream from the SNEC site and PENELEC property.

2.1.1.5.2 Groundwater Regime and Hydrogeology

The groundwater regime at the SNEC site and PENELEC property consists of an overburden and bedrock water-bearing units. The overburden water-bearing unit is comprised of an upper

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[1]GPU Nuclear, 2002. "Saxton Nuclear Experimental Corporation Facility Decommissioning Environmental Report," Revision 2.

[2]Ground/Water Technology, Inc., 1981. "Preliminary Hydrogeological Investigation Saxton Nuclear Experimental Station, Saxton, Pennsylvania."

layer of construction fill materials (i.e., silt, sand, and gravel or ash and cinders); a lower stream-deposited boulder layer where the boulders are quartzite and the interstices are filled primarily with silt and clay; and a weathered siltstone bedrock unit in some areas. The thickness of fill layer ranges from 0-m (0-ft) near the river to 0.43 to 1.22-m (1.4 to 4-ft) elsewhere, and the thickness of the boulder layer ranges from approximately 2.13 to 4.57-m (7 to 15-ft). Some of the overburden monitoring wells have been screened into the upper portion of the weathered bedrock.[3, 4] The NRC staff has estimated the thickness of this weathered bedrock unit from 0 to 1-m (0 to 3.28-ft). Two hydrogeologic cross-sections across the SNEC site and PENELEC property (Figures 5 and 6)[5] illustrate the variations in the thicknesses of construction fill materials and the boulder layers.

The bedrock water-bearing unit, which lies beneath the overburden units, is Upper Devonian marine deposits of red, grey, and olive-green siltstone, with variable amounts of fine sandstone and clay. The predominant lithology of the bedrock at the SNEC site and PENELEC property is a weathered and fractured siltstone. The coloration of the bedrock varies, based on oxidation and reduction conditions during deposition of the sediments. Groundwater movement in the bedrock is controlled by the fractures and bedding within the rock materials. There are two fracture patterns at the SNEC site and PENELEC property. The apparent dominant trend is northwest to southeast with dips steeply to the southwest, and a second fracture trend (which approximates the strike of the bedding) is northeast to southwest with dips moderately to the northwest.[1, 2] The bedrock elevation decreases from east to west across the site and from the CV to the Raystown Branch of the Juniata River.

2.1.1.5.3 Radiological Spills and Leaks

The licensee, in its LTP and HSA,[6] has acknowledged that spills and leaks of plant-generated radionuclides have occurred at the SNEC Facility. The licensee has responded to the radiological contamination of the soils and groundwater with remediation activities and additional characterization. For example, in 1994, the licensee installed additional bedrock monitoring wells to evaluate the impact of H-3 leaks observed in an overburden monitoring well near the former Waste Treatment Building.[7]

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[3]Haley & Aldrich, Inc., March 14, 2001. "Report of Field Investigation, Saxton Nuclear Experimental Station Saxton, Pennsylvania." Letter report to GPU Nuclear.

[4]GPU Nuclear, 2002. "Supplemental Response to RAI #3 Questions." Letter to NRC dated January 24, 2002.

[5]Haley & Aldrich, Inc. March 14, 2001. "Report of Field Investigation Saxton Nuclear Experimental Station Saxton, Pennsylvania." Letter report to GPU Nuclear.

[6]GPU Nuclear, 2000. Saxton Nuclear Experimental Corporation Historical Site Assessment Report.

[7]Geo Engineering, 1994. "Phase I Report of Findings - Groundwater Investigation Saxton Nuclear Experimental Station Saxton , Pennsylvania." June 7, 1994, letter report to General Public Utilities Nuclear Corporation.

2.1.1.5.4 Licensee's Response to the Groundwater Information

The licensee performed additional characterization of the groundwater, emphasizing potential plant-generated radionuclide contamination. The licensee's revised LTP adequately addressed the groundwater issues. The following material presents the licensee's groundwater characterization of the SNEC site and PENELEC property and additional NRC staff comments to clarify and supplement the characterization.

2.1.1.5.4.1 Radiological Monitoring Wells

The licensee has added additional monitoring wells screened in both the overburden and bedrock water-bearing units. The existing monitoring wells were the following:

(1) Eight overburden wells (GEO-1, GEO-2, GEO-3, GEO-4, GEO-5, GEO-6, GEO-7, and GEO-8) were installed during the fall of 1992.[8]

(2) Two angular bedrock wells (MW-1 and MW-2) were installed at 25 degrees to vertical along fracture trends, and a vertical piezometer (GEO-9) was installed in the bedrock in 1994. The angular wells were installed to intercept the maximum amount of groundwater from the fractures and bedding planes, whereas the piezometer was installed for water-level measurements of the angular wells. All wells were screened about 15.2-m (50-ft) below the land surface near the base of the CV.[7]

(3) Two vertical bedrock wells (MW-3 and MW-4) were installed about 8.5 m (28 ft) below the land surface near the former Waste Treatment Building. One overburden well (GEO-10) was installed down-gradient from the CV and outside the SNEC Facility.[9]

The licensee installed seven additional monitoring wells in December 2000. Three nests of wells (OW-3/3R, OW-4/4R, and OW-5/5R) were installed with an overburden well drilled to the top of the bedrock and with a bedrock well (denoted by R) drilled about 15.2-m (50-ft) below the land surface. Another overburden well (OW-6) was installed to the top of the bedrock. During the spring of 2001, another nest (OW-7/7R) was installed. Most of these nine wells (OW-3/3R and OW-6 were installed for other reasons) were installed down-gradient from the CV between the SNEC Facility and the Raystown Branch of the Juniata River. Wells OW-3/3R represent the background conditions of the groundwater in the overburden and bedrock water-bearing units. Well OW-6 was installed near the former spray pond.[3, 4]

In March 2002, the licensee installed an angular bedrock well, NRC Angle Well, at 25 degrees to the vertical near the CV. This well was screened about 1.5-m (5-ft) above and below the base of the CV, to intercept groundwater near the CV.

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[8]Geo Engineering, 1992 "Phase I Report of Findings - Groundwater Investigation Saxton Nuclear Experimental Station Saxton , Pennsylvania." November 18, 1992 letter report to GPU.

[9]Haley & Aldrich, 1998. "Summary Field Work Saxton Nuclear Experimental Station Saxton, Pennsylvania." Letter dated July 24, 1998 to GPU Nuclear.

2.1.1.5.4.2 Potentiometric Groundwater Surfaces and Groundwater Flow Patterns

The groundwater flow patterns within the SNEC site and PENELEC property are based on the potentiometric groundwater surfaces; the hydraulic gradients; hydraulic conductivities of the different rock types; structural features (i.e., trends of the fractures in the bedrock); and bedding planes in the bedrock. The groundwater flow patterns in both the overburden and bedrock water-bearing units are toward the Raystown Branch of the Juniata River. Figures 3.3-1 and 3.3-2[1] represent the potentiometric surfaces for the overburden and bedrock water-bearing units, respectively, for high- and low-water levels. The April 12, 2001, water levels represent wet climatic conditions, whereas the November 6, 2001, water levels represent dry climatic conditions. Table 2-34 in the LTP lists the groundwater levels for the radiological monitoring wells discussed above for measurement periods approximately every 2 to 4 weeks from September 6, 2000, through April 16, 2002. An examination of the April 12, 2001, and November 6, 2001, water levels indicates that these do represent high- and low-water level periods. The overburden water-bearing unit ranges from less than 0.3 to 1.5-m (1 to 5-ft) lower in November, whereas the bedrock water-bearing unit ranges from 1m to about 2.7-m (a few feet to about 9-ft) lower in November.

The groundwater flow directions can be developed by constructing flow lines at right angles to the potentiometric contours on Figures 3.3-1 and 3.3-2. The groundwater flow directions or pathways are similar for high- and low-water levels. Furthermore, the groundwater flow patterns, as stated above, are somewhat similar for the overburden and bedrock water-bearing units. The flow patterns for these units are controlled by the surface-water elevation of the Raystown Branch of the Juniata River, their discharge area. Figures 7 and 8,[3] the potentiometric surfaces for the overburden and the bedrock water-bearing units, delineate the flow paths for groundwater originating from the SNEC Facility by the shaded blue areas. The dominant fracture trend, northwest to southeast, coincides with the groundwater flow patterns (northwest toward the river) in the bedrock (Figures 3.3-2[1] and 8[3]). The other bedrock fracture trend, northeast to southwest, does not appear to impact the groundwater flow patterns at the SNEC site and PENELEC property. Also, the discharge tunnel, which runs from the SSGS to the river, may intercept groundwater flow in the overburden water-bearing unit. The discharge tunnel acts as a barrier and a drain because of the coarse-grained materials backfilled around the tunnel.

2.1.1.5.4.3 Aquifer Parameters

The aquifer parameters for the SNEC site and PENELEC property are based on packer tests, laboratory analyses, and slug tests. The licensee performed laboratory grain-size analyses (sieve analyses) on two fill samples, and packer tests on discrete intervals of three bedrock boreholes (B-3, B-4, and B-5). The sieve analyses indicate that the ash and cinders (fly ash) appear to be somewhat coarser-grained than the sand and silt fill materials. However, the usefulness of this data is limited by the limited number of samples analyzed. The packer tests provided a range of hydraulic conductivity, in the bedrock, of 8.46×10^{-3} to 6.85×10^{-5} cm/sec, with some borehole zones exhibiting no flow.[2]

Slug tests were performed on three overburden wells (OW-3, OW-5, and OW-6) and four bedrock wells (OW-3R, OW-4R, OW-5R, and OW-7R). The tests were conducted by adding water to the wells and measuring the decline in water levels over time. The slug tests produced

hydraulic conductivity range of 1.13×10^{-4} to 4.94×10^{-5} cm/sec for the overburden wells and a range of 2.88×10^{-3} to 4.94×10^{-5} cm/sec for the bedrock wells.[4]

2.1.1.5.4.4 Groundwater Sampling and Analysis for Radionuclides

Historically, plant-generated radionuclides in the groundwater within the SNEC Facility appeared infrequently. The overburden well GEO-5 near the former Waste Treatment Building was the only existing (or new) radiological monitoring well to contain any plant-generated radionuclide (in this case H-3) significantly above background or Minimum Detectable Activity (MDA) levels. The highest H-3 level in groundwater obtained from well GEO-5 was 960 pCi/L (October 1995 sample). This concentration of H-3 was significantly below the U.S. Environmental Protection Agency Primary Drinking Water Standard for H-3 of 20,000 pCi/L.

The available radionuclide data for the initial LTP (submitted in February 1999 but returned to the licensee by NRC without review) was limited because only one of 13 existing radiological monitoring wells extended outside the SNEC Facility.

The licensee instigated quarterly groundwater sampling of the new and existing radiological monitoring wells. The licensee collected and analyzed quarterly groundwater samples from April 2001 through April 2002, until it could be verified that the likelihood of plant-generated radionuclides in either the overburden or bedrock groundwater was extremely low.

The licensee performed synoptic groundwater level measurements approximately every 2 to 4 weeks on these same wells, from September 6, 2000, through April 16, 2002. Table 2-34 in the LTP lists the groundwater levels for these measurement events.

The groundwater samples were analyzed for H-3 and gamma spectroscopy for each quarterly sampling event (Table 2-17a in the LTP). The results of these analyses indicate that H-3, Co-60, Cs-137, and Cs-134 are not present in these groundwater samples above the MDA levels.

Groundwater samples from the new radiological monitoring wells were initially analyzed for TRU and HTD radionuclides (Table 2-17c in the LTP). The only TRU or HTD radionuclides above the MDA were the U-234 and U-238 radionuclides for a few wells samples. None of these groundwater samples was significantly above the background levels for these radionuclides observed in wells OW-3 and OW-3R.

To resolve NRC staff uncertainty about the licensee's past and current sample preservation procedures, the NRC staff collected split or confirmatory groundwater samples during an April 1 -2, 2002, sampling event. The following monitoring wells were sampled: OW-3; OW-6; GEO-5 and GEO-8 (overburden); OW-3R; OW-4R; OW-5R; OW-7R; MW-4; and NRC Angle Well (bedrock). Two samples were collected from each well. One sample was preserved with dilute hydrochloric acid, and the other sample was not preserved, which represents the licensee's sampling procedure.

The analytical results of NRC split or confirmatory groundwater samples indicated that there were no statistically significant differences between the preserved and unpreserved samples. Usually, the preserved samples had slightly higher concentrations than the unpreserved samples. However, there were exceptions where the unpreserved samples had slightly higher

concentrations.[10] NRC's independent laboratory, Oak Ridge Institute for Science and Education, analyzed these samples for the following radionuclides: H-3; Cs-137; Co-60; C-14; Sr-90; I-129; Pu-242; Pu-239/240; Pu-241; Am-241; U-234; U-235; and U-238. The NRC's and the licensee's sample results are similar.

The NRC staff has concluded that the licensee's analytical procedures (i.e., not preserving the sample with dilute acid) has not significantly impacted its analytical results.

The analytical results for the NRC Angle Well indicate that all the radionuclides except radionuclides U-234 and U-238 are below the MDC. Even the U-234 and U-238 are only slightly above the background groundwater levels (well OW-3R).[10] These results support the licensee's determination that the bedrock beneath and adjacent to the base of the CV is non-impacted with respect to plant-generated radionuclides.

2.1.1.5.4.5 Rate of Groundwater Movement

The groundwater movement or travel times of plant-generated radionuclides at the SNEC site and PENELEC property is limited to H-3, because H-3 has been the only radionuclide observed in groundwater samples above MDA or background levels. However, recently, H-3 has not been observed in the groundwater above MDA or background levels.

H-3 is not absorbed by the soils or rock materials; therefore, it acts as a tracer for groundwater movement within the SNEC site and PENELEC property. The average velocity for groundwater can be calculated using the formula:

$$v = Ki / \theta.$$

Where: v = average velocity (dimension - length per time);
K = hydraulic conductivity (dimension - length per time);
i = hydraulic gradient or change in head per unit length (dimensionless); and
θ = effective porosity of the flow medium (dimensionless).

The licensee has calculated H-3 travel times based upon the above average velocity formula for each water-bearing unit for high-and low-water levels. The NRC staff agrees with the licensee's estimates for H-3 travel times. The H-3 travel times indicate that plant-generated H-3 in the groundwater will be discharged into the Raystown Branch of the Juniata River after approximately 17 to 30 years for the overburden water-bearing unit and 5 to 6 years for the bedrock water-bearing unit.

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[10]ORISE, 2002. "Analytical Results for Water Samples Collected April 1 and 2, 2002, from the Saxton Nuclear Experimental Corporation (SNEC)," Saxton, Pennsylvania (Docket No. 50-146)[RFTA NO. 02-005].

Table 1. Travel Times for H-3

Water-Bearing Unit	High-Water Level (April 2001)	Low-Water Level (November 2001)
Overburden	range: 11-26 yrs	range: 20-46 yrs
Overburden	average: 17 yrs	average: 30 yrs
Bedrock	range: 0.6-33 yrs	range: 0.08-45 yrs
Bedrock	average: 5 yrs	average: 6 years

2.1.1.5.5 Conclusion

The current levels of plant-generated radionuclides in the groundwater at the SNEC site, both in the overburden and the bedrock water-bearing units, have been adequately addressed. The remaining remediation and decontamination activities are not likely to increase the level of plant-generated radionuclides dissolved in the groundwater. The NRC staff is aware that some groundwater monitoring wells may be abandoned because of remediation and decontamination activities. However, it is the staff's understanding that the remaining groundwater monitoring wells at this site will be maintained until license termination.

The staff has determined that the licensee's groundwater characterization with respect to plant-generated radionuclides is acceptable. The results of the radiological analyses indicate that no plant-generated radionuclides are currently present in the groundwater samples above the MDA levels or significantly above background levels. Also, the groundwater flow patterns and H-3 travel times indicate that mobile radionuclides dissolved in the groundwater will be discharged into the river after 5 to 30 years where they will be diluted by the much larger volume of surface water.

2.1.1.6 Surface Water

The SNEC site is bounded by the Raystown Branch of the Juniata River and Shoup's Run. The water quality of the Raystown Branch of the Juniata River has not been environmentally impaired by conditions upstream of the SNEC site. However, Shoup's Run has acidic waters that are caused by upstream materials derived from former mining activities.

The licensee has collected quarterly grab samples from two river locations, one upstream from the site and the other near the former site discharge tunnel to the river. The surface waters are analyzed for H-3 and gamma-emitting radionuclides. No radionuclides attributed to the SNEC site have been detected above the MDC. Because of the groundwater flow patterns in both the overburden and bedrock water-bearing units, it is unlikely that Shoup's Run has been contaminated with plant-generated radionuclides. Therefore, the NRC staff has concluded that the surface waters of the Raystown Branch of the Juniata River and Shoup's Run have not been impacted by plant-generated radionuclides.

The staff has determined that the licensee's surface water characterization is acceptable. The information presented in this section and in the previous Groundwater Section indicates that the impact of plant-generated radionuclides on the site's surface water is insignificant.

2.1.1.7 Sediment

The licensee discussed the collection of bottom sediments obtained from near-field river sites, in the vicinity of the SNEC Weir Discharge and Discharge Tunnel, and standing waters near the former Spray Pond in Sections 2.2.4.7.3 through 2.2.4.7.8 of the LTP. The selection of these sediment sample sites was based on areas of interest and field reconnaissance activities performed by the licensee to identify likely radionuclide depositional zones. At each site, one of the following methods was used to collect samples: core sampling; ponar sampling; suction sampling; or scoop sampling. In addition, background sediment samples were taken from three locations in the river upstream of possible contamination from the SNEC facility. Table 2-23 of the LTP lists the sample locations and type.

The licensee selected 10 near-field river sites, located in depositional zones that could have been impacted by SNEC facility discharges, including five locations downstream of all discharge points, for sediment sampling. These sampling locations, and others in the immediate vicinity of the Discharge Tunnel and Spray Pond, all had results that were at or below environmental background detection levels and are thus classified as non-impacted. On the other hand, samples taken from two locations at the Weir Line site had Cs-137 concentrations ranging from 0.04 Bq/g (1.2 pCi/g) to 0.09 Bq/g (2.55 pCi/g). These samples were taken from an area of approximately 25-m^2 (269-ft^2) surrounding the location where the mouth of the weir pipe used to be located. Since the Cs-137 contamination in this area is about 25 percent of the site Cs-137 DCGL, it was classified as an impacted Class 2 area. Surrounding this Class 2 area, the licencee has also designated a Class 3 buffer zone, as a transition to the non-impacted areas of the river. Tables 2-24 and 2-25 of the LTP provide the river sediment gamma spectroscopy and TRU and HTD radionuclide results, respectively. U-234, U-235, and U-238 activity was also detected in some samples and these results are indicative of natural background uranium.

Sediments will be evaluated against the volumetric soil DCGLs. Through the inspection process, the NRC staff will verify that the LTP requirements that apply to surface soil sampling and analysis and data assessment were applied to sediments.

The NRC staff has determined that the licensee's sediment characterization strategy is acceptable because it is consistent with the MARSSIM approach.

2.1.1.8 Pavement

Paved and unpaved site access roads extend over the SNEC facility property as well as the adjacent PENELEC property. Also, there are two main paved areas at the site. One area lies between the PENELEC Warehouse and Garage; the other area is by the DSF. Figures 2-11, 2-12, and 5-1 of the LTP show these features in detail. The licensee performed scan surveys of these surfaces using 5-cm (2-inch) -diameter by 5-cm (2-inch) -long NaI detectors. Near-site background scans of 7200 cpm to 13,400 cpm were comparable with measurements of 8400 cpm to 13,400 cpm taken at the access roads and the main paved areas. Detectable concentrations of Cs-137 for soil and pavement samples ranged from 0.0037 Bq/g (0.1 pCi/g) to 0.022 Bq/g (0.6 pCi/g). Additionally, the pavement area south of the DSF had subsurface

sampling and gamma logging performed. Results of sampling and gamma logging showed no activity related to licensed operations. The characterization results are summarized in Tables 2-15, 2-16, and 2-28 of the LTP.

In Section 2.2.4.3 of the LTP, the licensee states that the radiological characterization of these areas indicates they are likely to be non-impacted. However, the licensee has designated these areas as Class 1 and Class 2 survey units, based on the HSA information as to the use and history of these areas and a very conservative application of such classifications from MARSSIM guidance. Areas adjacent to Class 2 roadways have been classified as Class 3 impacted areas, serving as buffer zones to the surrounding non-impacted open-land areas. NRC, through the inspection process, will review the FSS design for these areas. Furthermore, in reviewing the FSS reports submitted by the licensee and NRC confirmatory results for paved areas, NRC will evaluate the related results.

The NRC staff has determined that the licensee's pavement characterization strategy is acceptable because it is consistent with the MARSSIM approach.

2.1.1.9 Exposure Rate Survey

The NRC staff reviewed exposure rate data provided in the SNEC LTP for structures and areas. Table 2-6 of the LTP provided a summary of exposure rates for the PENELEC Garage, Line Shack, Switch Yard Building, and Warehouse, the DSF, and SSGS Discharge Tunnel. General area measurements ranged from 0.04 mSv/hr (4 mrem/hr), at the interior of the PENELEC Switchyard Building, to 0.28 mSv/hr (28 mrem/hr), at the interior of the Material Handling Bay section of the DSF. Figure 2-18 of the LTP shows the SSGS Discharge Tunnel in detail and contains the general area exposure rate results. Figures 2-19 through 2-22 of the LTP show the PENELEC buildings in detail and give the general area-exposure-rate measurements taken in each building. Additionally, Figure 2-28 of the LTP indicates an average exposure rate of 0.05 mSv/hr (5 mrem/hr) in the SSGS Intake Tunnel.

General area-exposure-rate measurements were taken for the CV before the licensee's decision to totally remove the internal CV concrete. Figures 2-1 through 2-5 of the LTP provide information on general area-exposure-rates for locations inside the CV before concrete removal, as well as the exterior surfaces. However, the licensee states, in Section 2.2.4 of the LTP, that this characterization will no longer be applicable to the FSS design because the licensee changed its plans to remove all CV concrete. Removal of the CV concrete has been completed and all that remains inside the CV is the steel liner.

General area exposure rate measurements for the SNEC Facility and PENELEC property were taken using various instrumentation during characterization and as part of routine operational decommissioning surveys. Different survey instrumentation was used depending on the range of exposure rates to be measured. Low-level dose-rate measurements were taken using a Bicron MicroRem meter, a scintillation-based dose-rate instrument. When exposure rates exceeded the measurement range of the Bicron MicroRem meter, the licensee used an Eberline RO-2, or equivalent, ion-chamber instrument.

The licensee references the conduct of a radiological scan survey for open land areas at the SNEC site and adjoining PENELEC property in Section 2.2.4.2 of the LTP. Approximately 60,703-m^2 (15-acres) were surveyed with 100 percent coverage, using NaI scintillation

spectrometers. Based on this survey, the average concentration, site-wide, of Cs-137 in soil, was estimated to be 0.01 ± 0.006 Bq/g (0.3 ± 0.15 pCi/g) (1 standard deviation). According to the licensee, this survey constituted the first phase of a two-phase effort to perform an FSS. The licensee has indicated that all survey results are available at the SNEC facility for NRC inspection. NRC will review the licensee's exposure rate data throughout the RSSI process, as part of the inspection process, to determine whether it is acceptable, and if instrument use and data analysis are consistent with MARSSIM guidance.

The licensee used the Bicron MicroRem meter to take background exposure rate measurements of land areas about 16.1 km (10 mi) from the site. A list of these measurements was provided in Table 2-22 of the LTP. The background exposure rate varied from 5 μR/hr to 11 μR/hr, with the average being about 7 μR/hr.

The NRC staff has determined that the licensee's exposure rate survey strategy is acceptable because it is consistent with the MARSSIM approach.

2.1.2 Site Characterization – Summary Finding

The NRC staff has reviewed the information provided by the licensee in the SNEC LTP, according to Section B.2 of NRC Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors." Based on this review, the NRC staff has determined that the licensee has met the objectives of providing an adequate site characterization, as required by 10 CFR 50.82(a)(9)(ii)(A), because the licensee followed the guidance in MARSSIM and NUREG-1727 and provided adequate characterization information to justify the classification of each survey unit as non-impacted or impacted (i.e., Class 1, Class 2, or Class 3).

2.2 Remaining Dismantlement Activities

In accordance with the requirement of 10 CFR 50.82(a)(9)(ii)(B), the licensee provided a listing of the remaining dismantlement activities necessary for license termination as of September 2002. Also, in accordance with the guidance provided in Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors," the licensee provided a radioactive waste characterization estimate: an estimate of the quantity of radioactive material that will be shipped for off-site burial; an estimate of personnel exposures; and the methods that will be used to control the spread of contamination, while performing these dismantlement activities – and they are acceptable.

In Section 3.2.4 of the LTP, the licensee states that the SNEC Facility License requires the interior of the CV to be maintained as an exclusion area and the ventilation system to be operating during activities that have the potential to cause a measurable release. Thus, before removal of the upper dome of the CV and the CV Ventilation System, the SNEC Technical Specifications will need to be revised to allow these dismantlement activities. NRC is in the process of approving these changes to the technical specifications, which will be incorporated in Amended Facility License No. DPR-4 .

The licensee estimated, as of September 2002, that the remaining low-level radioactive waste generated as a result of decommissioning activities will be: (1) 22,679 kilograms (kg) [50,000 pounds (lb)] of metal; (2) 14.2 cubic meters (m^3)[(500 cubic foot (ft^3)] of soil; (3) 6056 liters (L)

[(1600 gallons (gal.)] of water; (4) 1.4 m^3 (50 ft^3) of sediment; (5) 2,608,154 kg (5,750,000 lb) of concrete; (6) 56.6 m^3 (2000 ft^3) of dry active waste; and (7) 6803 kg (15,000 lb) of debris. In addition, the licensee estimates the total radiation exposure to complete the remaining site dismantlement tasks to be 0.38 person-sievert (Sv) [38 person-roentgen-equivalent-man (rem)]. The remaining dismantlement activities will continue to be conducted under the existing SNEC Radiation Protection Program and Radioactive Waste Management Program.

The NRC staff has reviewed the information in the LTP for the SNEC site according to Section B3 of Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors." Based on this review the NRC staff has determined that the licensee has identified the remaining dismantlement activities necessary to complete decommissioning of the facility, as required by 10 CFR 50.82(a)(9)(ii)(B). Further, the NRC staff has determined that these activities can be completed in accordance with 10 CFR 50.59.

2.3 Plans for Site Remediation

In accordance with the requirement of 10 CFR 50.82(a)(9)(ii)(C), the licensee provided its plans for completing the radiological remediation of the site. The licensee plans to remediate the site, including structures that remain on site, to the criteria specified in 10 CFR Part 20, for unrestricted use. To meet these criteria, the licensee plans to use typical remediation methods, which include abrasive blasting, concrete sectioning, needle gunning, concrete scabbling, and manual removal of building and structural materials. The licensee states, in Section 3.4 of the LTP, that greater than 99 percent of the (curies) present at the SNEC facility at the beginning of decommissioning, in 1998, have been safely removed from the site and shipped for proper disposition (as of September 2002).

Included in the criteria specified in 10 CFR Part 20 for release for unrestricted use is that, in addition to the remaining residual radioactivity being less than or equal to 0.25 mSv (25 mrem) per year above background, the remaining residual radioactivity must be reduced to levels that are as low as is reasonably achievable (ALARA). The licensee also provided its ALARA analysis strategy in Section 6.4 of the LTP. The licensee stated in Section 6.4 of the LTP that remediation of all structural surfaces to or below the DecontaminationandDecommissioning (DandD) code screening levels would ensure that any residual radioactivity remaining at the site would not result in any significant impact on public health and safety. Through the inspection process, NRC will review the licensee's ALARA evaluation as required by 10 CFR 20.1402. For volumetric contamination, the remediation plans are to remove residual contamination to below the DCGL values provided in Table 6-2 of the LTP. Based on the guidance provided in NUREG-1727, "NMSS Decommissioning Standard Review Plan," the licensee indicated, in Section 6.4 of the LTP, that in the case of the SNEC Facility, no further ALARA evaluation is required, after the removal of soil contamination, to reach the DCGLs. The licensee's strategy for determining the remediation levels regarding application of the ALARA process conforms to the guidance provided in NUREG-1727.

The NRC staff has reviewed the information in the LTP for the SNEC facility according to Section B.4 of Regulatory Guide 1.179. Based on this review, the NRC staff has determined that the licensee has adequate processes to identify all site areas that may require remediation and has in place an organization to safely perform the remediation, as required by 10 CFR 50.82(a)(9)(ii)(C). Further, the NRC staff has determined that there is reasonable certainty that

the licensee can complete remediation at this site and meet the criteria specified in 10 CFR Part 20 for unrestricted use.

2.4 Final Site Survey

The FSS is the radiation survey performed after an area has been characterized, remediation has been completed, and the licensee believes that the area is ready to be released for unrestricted use. The purpose of the FSS is to demonstrate that the area meets the radiological criteria for license termination. The FSS design entails an iterative process that requires appropriate site classification—based on the potential residual radionuclide concentration levels relative to the DCGLs—and formal planning, using DQOs. An integrated design is developed that addresses selection of appropriate survey and laboratory instrumentation and procedures and a statistically based measurement and sampling plan for collecting and evaluating the FSS data. Sections 5.4, "Survey Design"; 5.5, "Survey Data Collection"; 5.6, "Survey Data Assessment"; and Appendices 5-1 through 5-3 of the LTP were reviewed to ensure that the design was appropriate and all applicable variables had been addressed. In some cases, all information was not available for the licensee to complete all aspects of the design. The licensee will have to gather this information as the decommissioning progresses. For those survey design aspects that could not be completed, the licensee has committed to following appropriate guidance contained in NUREG-1575, or, alternatively, preparing a technical-basis document for deviations from the guidance. This approach to final survey design is acceptable because it is consistent with the approach in MARSSIM and NUREG-1727.

NRC recognizes that all the information required to properly design the FSS may not be available when the licensee submits the LTP for review, and is therefore conducting performance-based in-process inspections of the licensee's FSS program at various stages in its decommissioning process. The purpose of the inspections is to verify the implementation of the commitments the licensee made in the LTP, and to review the procedures, methodology, equipment, training and qualifications, and QC. Implementation of DCGLs, embedded piping surveys, detection sensitivities, instrument calibration, reference background areas, area factors, QA/QC, and other areas may be subjects for future inspections.

The licensee used site historical, scoping, and characterization data, together with process knowledge and operational and routine surveillance survey records, as the principal means for initially classifying site areas as non-impacted or impacted. Table 5.2 of the LTP lists each survey unit, the size of the survey unit, and the initial classification as either Class 1, 2, or 3. Survey unit sizes that the licensee designated as Class 1, 2, or 3 were evaluated relative to the recommendations provided in MARSSIM and were within the recommended limits. For Class 3 structures and land areas, the licensee has committed to limiting the survey unit size to 10,000-m^2 (107,639-ft^2), in contrast to the "no limit" for survey unit size suggested in MARSSIM.

MARSSIM does not directly address FSS design for subsurface soils. As explained in Section 5.5.3.4.7 of the LTP, for survey units where subsurface contamination may be present, the licensee has committed to use site procedures and a systematic process for collecting subsurface information. The licensee has identified the Spray Pond, the 4645.8-m^2 (1.148-acre) SNEC Facility site, and any suspect surface areas that have shown contamination levels approaching the DCGL as locations where subsurface surveys will be planned. Section 5.5.3.4.7 of the LTP describes the subsurface sampling process. NRC plans to review this

information as part of NRC's ongoing inspection effort. Furthermore, in reviewing the FSS reports submitted by the licensee and NRC confirmatory results for subsurface soil contamination, NRC will evaluate the related results consistent with dose-modeling assumptions, and any concerns may be handled as inspection follow-up items through the inspection process.

The LTP described the input parameters for the DQOs. These DQOs included the selection of an appropriate statistical test, limits on decision errors, and variables for calculating sample size. The statistical tests discussed in MARSSIM are the Wilcoxon Rank Sum (WRS) test and the Sign test. The WRS test is typically selected when the radionuclides of concern are present in background, or gross measurements are to be made. The WRS test also requires the identification of appropriate background reference areas from which the same sample or measurement type is collected as was collected from the survey unit. Alternatively, the Sign test may be selected if the radionuclides of concern are not present in background or are present at a small fraction of the DCGL. Finally, consistent with NUREG-1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions," the Sign test may be used to evaluate gross activity measurements from survey units containing multiple materials, by subtracting the appropriate background, using paired measurements. Sections 5.4.1.3 and 5.6.4.3 of the LTP indicate that the licensee may compare the total radionuclide concentrations, including background, to the release criteria.

However, in Section 5.4.1 of the LTP, the licensee has committed not to subtract environmental background concentrations of Cs-137 from nuclide-specific measurement results (e.g., gamma-ray spectroscopy measurements of sample materials, etc.) for any open-land area. In addition, Sr-90 will be handled in a similar manner. Regarding the uranium isotopes, the licensee provided an analysis showing that uranium detected at the site is likely natural uranium, as opposed to originating from site operations. Thus, uranium concentrations will not be assessed when demonstrating regulatory compliance for unrestricted release. Should conditions require application of the WRS test, Appendix 5-3 of the LTP establishes the methods that the licensee would use to select background reference areas. The NRC staff finds this acceptable because it is consistent with the approach in MARSSIM and NURGEG-1505, "A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys."

The input parameters for sample size calculations include the DCGL, the Lower Bound of the Gray Region (LBGR) — which generally provides an estimate of the mean concentration in the survey unit, but may be adjusted to optimize design — and an estimate of the radionuclide variability. These parameters, together with decision errors, are used to calculate the required number of statistical samples, and the information is usually available from planning or from preliminary surveys (scoping, characterization, remedial action support). For initial planning purposes, the licensee has set the LBGR at 50 percent of the DCGL, established default decision errors at 0.05, and in Section 5.2.3.2.3 and Appendix 5-2 of the LTP, discussed how it will estimate the radionuclide variability. The principal decision error of concern to NRC for survey design inputs is the Type I or α error. This error occurs when a survey unit is determined to meet the release criteria when in fact it does not. The default value of 0.05 for the Type I or α error is acceptable. Section 5.4.3.1 of the LTP indicated that the number of samples that will be collected from each survey unit will be determined by the method described in Section 5.5.2 of MARSSIM. The licensee has further committed, in Section 5.6.1 of the LTP, to evaluate the acceptability of the selected input parameters when assessing final status data.

The statistical survey planning strategy discussed in the LTP is acceptable because it is consistent with the MARSSIM approach.

MARSSIM allows for the use of advanced survey technologies as long as these techniques meet the applicable requirements for data quality and quantity. The decision to use advanced technologies and other methods may not be a matter of convenience for a licensee, but a genuine attempt to use the best instruments, which can minimize the potential for residual radioactivity to exceed the release criterion. For example, by using a system that can sample the entire population or surface, a licensee can determine if there exists any combination of areas and activity that could potentially exceed the dose limit.

In Chapter 5 of the LTP, there are several references to advanced survey technologies and methods such as *in-situ* gamma spectroscopy and semi-automated equipment for possible use during the conduct of the FSS. Use of these technologies and methods necessitates that they meet the DQOs specific to the FSS design. Consequently, when such is used, the licensee has committed to include in each survey design package a description of the instrument calibration, operational checks, sensitivity, and sampling methods, with a demonstration that the instruments and methods have adequate sensitivity. In Section 5.4.3 of the LTP, the licensee indicates that it is currently evaluating whether to use semi-automated, large-area, position-sensitive, radiation-measurement equipment for scanning structural surfaces and surface soils. With the ability to provide measurements covering 100 percent of the area, in conjunction with an appropriate MDC, results using this equipment could be substituted for static measurements. However, the licensee has committed that soil samples will still be collected in open-land areas, in addition to these semi-automated scan survey or *in-situ* gamma spectroscopy special measurement techniques. NRC, through the inspection process, will review the procedures for using advanced survey technologies and methods, the acceptability for use in specific survey units, and the statistical techniques for evaluating the data. Furthermore, in reviewing the FSS reports submitted by the licensee, NRC will evaluate the results from using these technologies and methods.

The LTP discussion of the reference system that will be used for structural surfaces and land areas is provided in Section 5.4.1.2 and in Table 5-5. Overall, the proposed grid sizes are appropriate for the survey unit classification and the type of survey unit (i.e., structures or land area). Grid coordinates will then serve as the basis for the random-start systematic sample-location selection. The recommended guidance has been adapted for this portion of the LTP and is, therefore, acceptable.

FSS-meter/detector selection, calibration, and evaluation are discussed in Sections 5.2.7.3 and 5.5.2.2, and Tables 5-9 and 5-10 of the LTP. The meter/detector selection process must ensure that the instrumentation used for the FSS will adequately respond to the radiations emitted from the various radionuclides of concern, is sufficiently sensitive to detect these radiations at levels less than the DCGLs, and is calibrated in a manner that accounts for the radionuclide mixture, the expected radiation energies of the mixture, and surface efficiencies. The NRC staff finds that the instrumentation presented in Table 5-9 of the LTP is appropriate for the primary detectable radionuclides, for surveys of structures and land areas. Section 5.5.2.2 provides a discussion of the expected calibration sources that will be used and accounts for other modifying factors, specifically surface efficiency. The licensee will use National Institute of Standards and Technology traceable calibration sources that are similar in energy to the primary radionuclides of concern. The methods used to determine the beta-gamma and

alpha scan MDCs for structural surfaces in Sections 5.5.2.4.1 and 5.5.2.4.2, respectively, and the gamma scan MDC for land areas in Section 5.5.2.4.3, are appropriate. In addition, the methods discussed in Sections 5.5.2.4.4 and 5.5.2.5 of the LTP to determine the MDC for static measurements are appropriate. Table 5-10 provides the typical detection sensitivities (i.e., scanning and static) for the principal instruments that are expected to be used for alpha and beta-gamma direct-surface-contamination measurements. However, for the conduct of the FSS, the MDCs for both scanning and static measurements will be derived specific to the instrumentation used and conditions of performance present at that time. The licensee has committed to include in each survey unit design package a description of the instruments, methods, calibration, operational checks, coverage, and sensitivity for each media and radionuclide. NRC will review the licensee's determination of specific MDCs for both scanning and static measurements, and other instrumentation-related parameters, for inclusion in the FSS design packages as part of NRC's ongoing inspection process.

Based on the review of Section 5.5.2.3 of the LTP, which discusses instrument response checks, the NRC staff has determined that the licensee's plans are acceptable because they are consistent with the MARSSIM approach. The licensee will perform an instrument response check before use each day. Portable instruments will also be checked after instrument use each day. Should a response check fail, the instrument is removed from service and any data collected since the last acceptable check are evaluated and may be discarded.

FSS procedures contained in Section 5.5.3 of the LTP and supplemented with information found in Sections 5.4.2 and 5.4.3 and Table 5-3 discuss methods for performing surface scans, surface activity measurements, soil and bulk material sampling, and special measurements. Surface-scan descriptions recognize the importance of surface-to-detector distance and scan speed to achieve an adequate scan sensitivity. The licensee provided a discussion in Appendix 5-2 for evaluating the required-scan MDC against the actual-scan MDC, to ensure adequate sensitivity is achieved. If adequate sensitivity is not achieved, additional measurements or samples are required. The scan coverage is based on survey unit classification, with Class 1 survey units receiving 100 percent scan coverage and Classes 2 and 3 receiving coverages of 10 to 50 percent (or greater) and up to 10 percent, respectively. The licensee states, in Section 5.4.3.2 of the LTP, that measurement/sampling locations are to be determined based on a random-start systematic pattern for Classes 1 and 2 survey units and randomly for Class 3 survey units. Additional measurements or samples will also be collected from areas of elevated radioactivity detected while scanning, and from judgmental locations. The staff finds these proposed methodologies for surveys are acceptable because they generally follow MARSSIM guidance.

MARSSIM does not apply to non-structural systems and components. Free release of non-structural systems and components, except for buried piping, will be done in accordance with current free-release guidance. As appropriate, those non-structural systems and components not meeting the release criteria will be disposed of as radioactive waste.

For facility systems, many of the survey design criteria applicable to structures and land areas do not apply or are dictated by the physical system layout and the accessibility to the system piping and components. To accommodate these factors, the licensee has integrated both non-systematic (random) and judgmental (biased) methods of data collection in the FSS design. The licensee states, in Section 5.4 of the LTP, that survey design will be performed in accordance with SNEC procedures E900-ADM-4500.59, "Final Site Survey Planning," and

E900-ADM-4500.58, "Treatment of Embedded Piping and Components." Survey procedures in the latter, for embedded piping and components, have not been completely described – although the LTP provides specific commitments the licensee intends to satisfy. These commitments include: surface scan coverage according to the class of survey unit, depending on the accessible surface area; taking 30 static measurements at accessible points; assessing scale and sediment samples as appropriate; and, when practical, filling embedded pipes with grout or concrete. When remaining sections of system piping allow less than 30 measurement points, the licensee will assess whether the embedded piping can easily be removed or prepared for FSS. In this case, the licensee will use the RSSI process as a basis for biased scanning and sampling of the radionuclide concentration present on/in the embedment, to ensure that the release criteria are met. For implementation of the release criteria (i.e., DCGLs) for buried pipes that remain during the FSS, see Sections 2.5.3.2 and 2.5.4 of this SER.

In Section 5.5.3.4.3 of the LTP, the licensee states that internal video inspections of piping have been performed to assist in determining the viability of internal measurements, and also to map out or diagram the extent of some system piping. Video inspections were also used to determine the interconnections and level of sediment in pipe sections. To measure the internal activity of piping, the licensee indicates the use of NaI detection systems attached to multichannel analyzers to perform *in-situ* analysis of gamma-emitting radionuclides. In this case, gamma emitters will be used as surrogate radionuclides. Other types of detection systems that can measure internal surface beta and/or alpha emissions directly (in properly prepared piping sections) may also be used when appropriate. The survey methods for systems and components, including embedded piping, as presented in the LTP, are recognized by the NRC staff as an acceptable general approach. NRC plans to review the licensee's procedures for FSS planning and treatment of embedded piping and components as part of NRC's ongoing inspection effort. Furthermore, in reviewing the FSS reports submitted by the licensee and NRC confirmatory results for embedded piping and components, NRC will evaluate the related results, and any concerns may be handled as inspection follow-up items through the inspection process.

During 2002, the licensee performed the FSS for several steel surface areas of the CV steel shell, in support of installation of steel I-beams, which were designed to stabilize the shell during removal of concrete. The FSS had to be done then, since the I-beams remain in place permanently, and therefore the CV steel surface behind them is no longer accessible for surveying. Cleaning of the steel surface involved methods such as surface grinding to remove surface oxides, paint, and any residual concrete. According to the licensee, the vast majority of surface contamination was removed. However, no information was provided in the LTP on the radiological status of the steel surface as the result of post-cleaning surveys that were performed to verify that the cleaning effort was successful. As stated in Section 5.5.3.4.9 of the LTP, these areas have been surveyed "at risk," in that they have been surveyed before NRC approval of the SNEC LTP. NRC, through the inspection process, will review the post-remediation characterization data and the FSS to ensure that: (1) data quality, investigations of anomalies, and survey design are adequate; (2) the appropriate DCGLs for surface contamination were applied; (3) areas of elevated activity were evaluated against the appropriate DCGL elevated measurement comparison ($DCGL_{emc}$); (4) any scanning factors used were acceptable; (5) the entire steel surface was scanned; (6) static measurements were properly located; and (7) the release criteria are satisfied for each survey unit.

Section 5.5.4 of the LTP discusses the process for sample handling and analysis. The licensee controls sample tracking, using a chain of custody record for samples that are shipped off-site for analysis. On-site sample analysis capabilities include: gamma spectroscopy of bulk sample materials; tritium analysis by liquid-scintillation counting; and gross beta/gamma and alpha counting of smears by a Geiger-Meuller thin-window detector; and zinc-sulfide scintillation-based scalar, respectively. The NRC staff finds that these analyses are appropriate for the primary radionuclides of concern at the site. Other radionuclides, such as Sr-90 and TRU, will require wet-chemistry analysis; an offsite laboratory does this, as needed. The off-site laboratory also performs sample analysis for HTD radionuclides. The licensee states, in Section 5.5.4 of the LTP, that the sensitivity of the analytical procedures will be 10 to 50 percent (or less) of the DCGLs, which is acceptable to the NRC staff.

Data management requirements are discussed in Section 5.5.5 of the LTP. Specifically, the section defines which data may be used as FSS data. Some of the elements of the data management system are described in the LTP. However, the methodology by which the licensee will integrate the data and demonstrate regulatory compliance is not described in the LTP. This aspect will be the focus of future NRC in-process inspections.

The survey data assessment process and investigation levels are discussed in Sections 5.6 and 5.4.4.1 through 5.4.4.5 of the LTP. The licensee's data assessment, as described in the LTP, involves data validation, graphical data reviews, basic statistical evaluations, and statistical data testing, when applicable. The data validation is presented as a nine-step process, to ensure data quality and defensibility. Any discrepancies identified must be investigated. The graphical data reviews that the licensee intends to use serve to identify spatial patterns and potential anomalies that would indicate additional investigation is required. The basic statistical evaluation serves as a method to evaluate the adequacy of survey design and lists specific acceptance criteria. Finally, data testing would be performed when necessary, using either the WRS or Sign test or advanced survey technologies and methods, as discussed above. The approaches are acceptable because they are consistent with the methods described in MARSSIM.

The licensee has established investigation-level requirements, a process for evaluating the results of the investigation, and follow-up actions. Requirements for investigation are related to the survey unit classification and the DCGL. Elevated activity detected while scanning, or from measurements, is investigated. The licensee states that the investigation may result in remediation, reclassification of a given survey unit to a higher level, and/or evaluation of the elevated area to a $DCGL_{EMC}$. The staff finds these processes are acceptable as a means to ensure data quality and adequate investigations of anomalies, evaluation of the FSS design, and assurance that the release criteria are satisfied for each survey unit.

When reviewing FSS results, NRC will examine whether the licensee has measured and/or accounted for each of the radionuclide contaminants (Table 6-1 of the LTP) when presenting dose compliance information for each survey unit. Additionally, whenever the licensee accounts for HTD radionuclides through surrogate analyses, NRC will examine if the licensee has verified that the activity ratios (activity ratio for difficult to detect radionuclides to easy-to-detect radionuclides) used during the FSS were valid in accordance with Sections 5.2.3.2.3 and 5.2.7.6 of the LTP. In Sections 5.2.3.2.3 and 5.2.3.2.4 of the LTP, the licensee has committed to documenting in each survey package a radionuclide mix justification.

Section 5.7 of the LTP provides a brief description of the FSS documentation. The information that is to be compiled for each survey unit includes a history file and release record. At project completion, the licensee will prepare an FSS report summarizing the ALARA evaluations, survey data results, and overall conclusions, as they relate to the radiological criteria for release for unrestricted use. The NRC staff concludes that the planned presentation strategy of the final radiological status for the SNEC site and adjacent PENELEC property is acceptable because it is consistent with the approach in MARSSIM and NUREG-1727.

The NRC staff has reviewed the information in the LTP for the Saxton Nuclear Plant, provided by the licensee, according to Section B.5 of Regulatory Guide 1.179. Based on this review, the NRC staff has determined that the licensee has conformed to 10 CFR 50.82(a)(9)(ii)(D) in that the final radiation survey plan in the LTP provides assurance that residual radioactive contamination levels will meet the criteria specified in 10 CFR Part 20, for unrestricted use.

2.5 Compliance with Radiological Criteria for License Termination

The development of residual radionuclide concentration levels that will be used to demonstrate compliance with the regulations for releasing the site is discussed in Section 6 of the LTP. Two primary scenarios were considered in developing radionuclide-specific DCGLs for the SNEC Facility — a building occupancy scenario and a residential farming scenario. DCGLs have been developed as acceptable levels of residual radioactivity that can be left at the site in compliance with the unrestricted release criteria. Development of DCGLs that will be used to demonstrate compliance with the regulations is provided in Sections 5.2 and 6.3 of the LTP.

2.5.1 Site Release Criteria

For the SNEC facility, the licensee proposes an unrestricted release of the site per 10 CFR 20.1402. In accordance with 10 CFR 20.1402, the residual radioactivity that is distinguishable from background must not cause the total effective dose equivalent (TEDE) to an average member of the critical group to exceed 25 mrem/yr. The residual radioactivity must also be reduced to levels that are ALARA.

Section 6 of the LTP was reviewed using the following guidance documents: NUREG-1700, “Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans—Final Report”; MARSSIM; Draft NUREG-1549, “Decision Methods for Dose Assessment to Comply with Radiological Criteria for License Termination” and NUREG-1727, “NMSS Decommissioning Standard Review Plan.”

As required under 10 CFR 20.1402, expected doses are to be evaluated for the average member of the critical group, which is not necessarily the same as the maximally exposed individual. The use of the “average member of the critical group” acknowledges that any hypothetical “individual” used in the dose assessment is based, in some manner, on the statistical results from data gathered from groups of individuals.

Calculating the dose to the critical group is intended to bound the individual dose to other possible exposure groups because the critical group is a relatively small group of individuals, because of habits, actions, and characteristics, who could receive among the highest potential dose at some time in the future. By using the hypothetical critical group as the dose receptor, it

is unlikely that any individual would actually receive doses in excess of that calculated for the average member of the critical group.

2.5.2 Building-Surface Residual Radioactivity

DandD Version 1.0 was used to develop DCGLs for residual radioactivity remaining on surfaces of buildings and structures that will remain standing at the time of license termination. DandD was developed to allow a screening analysis with a high probability of assurance of ensuring that the true dose from exposure to residual radioactivity remaining at a site will not result in a dose exceeding the dose limit. In accordance with guidance provided in NUREG-1727, DCGLs developed through a screening analysis using DandD will be acceptable, if the following conditions are met:

(a) The assumptions of the screening analysis are not violated: specifically, (1) the residual radioactivity is on the wall surface (i.e., non-volumetric); and (2) the residual radioactivity on the surface is mostly fixed, with the fraction of loose radioactivity not to exceed 10 percent of the total activity.

(b) Only default parameters within the code are used in the analysis.

2.5.2.1 Contaminant Characteristics

DCGL values have been developed by the licensee for 11 radionuclides listed in Table 2 of this SER. The DCGLs were determined for those radionuclides found contributing to site contamination, based on a screening analysis that considered half-life, detection of radionuclides at the site, and the potential contribution from each radionuclide to the overall dose. In screening out radionuclides, the licensee initially developed a comprehensive list of 71 radionuclides that could be expected at the site, based on the type of reactor fuel used. From this initial list of 71 radionuclides, 30 radionuclides were screened out, based on having a very short half-life (i.e., less than 3 years). All of these radionuclides will have gone through 10 half-lives since the facility shutdown. With the exception of Cm-242, none of these short-lived radionuclides was detected above its MDA in any site samples. Cm-242 has been reported in a few samples above the MDA; however, the licensee has concluded that this is likely the result of a misidentification in the laboratory analysis because of interference from another isotope. As a result of the NRC staff's own assessment, the NRC staff agrees that detection of this isotope is likely the result of a misidentification. Because of the large number of half-lives Cm-242 would have undergone since the facility shutdown (i.e., 67 half-lives), it is unlikely that reported activities truly represent Cm-242 activities.

The licensee screened out an additional 23 radionuclides with half-lives greater than 3 years, based on their not being detected above their MDA in any site samples. Of the 18 remaining radionuclides, seven were screened out based on an analysis showing they would have only a small contribution to the total dose (i.e., less than 10 percent). This is consistent with NRC guidance in Appendix C of NUREG-1727. The licensee also provided an analysis showing that uranium detected at the site is likely natural uranium, as opposed to originating from site operations. As a result of this screening process, the licensee identified 11 radionuclides for which DCGL values should be developed.

Table 2. Radionuclides for Which DCGLs
Were Developed for the SNEC Site.
(Table taken from LTP)

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

The NRC staff reviewed the approach the licensee used for screening out radionuclides and the resulting suite of radionuclides for which DCGL values have been established, to ensure that an appropriate suite of radionuclides has been established. NRC staff performed independent confirmatory calculations for screening out radionuclides and compared the resulting suite of 11 radionuclides with the nuclides identified in the following documents:

- *Generic Environmental Statement, Mixed Oxide Fuel Recycle Plutonium in Light Water-Cooled Reactors*, Volume 3, Chapter 4, "Environmental Impact due to the Implementation of Plutonium Recycle" (AEC, 1974); and

- *Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities*, NUREG-0586 (NRC, 1988).

Based on the NRC staff assessment, it is concluded that the licensee has included radionuclides that would be expected to be present in significant quantities in a reactor that used mixed oxide fuel and had been in safe storage for 28 years.

2.5.2.2 Scenario Definition and Exposure Pathways

The building occupancy scenario is used to evaluate potential exposure to fixed and removable surface radioactivity within structures that will be left on the site after license termination. It is assumed that a light industrial worker occupies the structure in a passive manner, without deliberately disturbing the residual radioactivity on building surfaces. The worker is assumed to be exposed to penetrating radiation from surface sources, inhalation of resuspended surface contamination, and inadvertent ingestion of surface contamination.

A light industrial worker, who is assumed to be an adult, is considered the critical group. The internal designs of the buildings that could possibly be left standing at the Saxton site make consideration of reusing the buildings as residential houses or converting them to apartment complexes unlikely.

2.5.2.3 Application of DandD

Surface-contamination DCGL values were developed through the use of DandD Version 1.0, using all default parameters. The results from the DandD runs for each of the 11 radionuclides in units of mrem per dpm/100 cm^2 were scaled to the 25-mrem TEDE limit to determine an acceptable DCGL value. Table 3 lists DCGLs that will be used for residual radioactivity on building surfaces.

Table 3. SNEC DCGLs for Building Surfaces
(Table Taken from LTP)

Radionuclide	DCGL (dpm/100 cm^2)	Radionuclide	DCGL (dpm/100 cm^2)
Am-241	2.7E+01	Ni-63	1.8E+06
C-14	3.7E+06	Pu-238	3.0E+01
Co-60	7.1E+03	Pu-239	2.8E+01
Cs-137	2.8E+04	Pu-241	8.8E+02
Eu-152	1.3E+04	Sr-90	8.7E+03
H-3	1.2E+08		

The NRC staff has determined that a screening approach consistent with guidance in Appendix C of NUREG-1727 was used to develop DCGLs for building surfaces. NRC staff has confirmed that only default parameters were used in the analysis. Further, the LTP states that the building surface DCGLs will be applied to areas of the site where residual radioactivity is limited to the surface and the residual radioactivity is mostly fixed, with less than 10 percent of the total activity being loose.

2.5.3 Volumetric Residual Radioactivity

RESRAD Version 6.1 was used to determine DCGLs for residual radioactivity remaining in soils, sediments, and other volume sources.

2.5.3.1 Contaminant Characteristics

Eleven radionuclides were identified as relevant to the decontamination activities at the SNEC site. The selection of these radionuclides was judged by the NRC staff to be acceptable, and the reasons are discussed in detail in Section 2.5.2.1 of this SER.

2.5.3.2 Scenario Definition and Exposure Pathways

The residential farming scenario is used to evaluate potential exposure to volumetric residual radioactivity that will be left on the site after license termination. The LTP states that DCGL values derived from this scenario will be applied to open-land areas at the SNEC site. It will be also applied to sediments, buried pipes (no above-surface pipes are expected at the site after

decommissioning), and construction debris. Specific survey units for which these DCGL values will be applied are listed in Table 5-2 of the LTP.

For the residential farming scenario, the residual radioactivity is assumed to be distributed in a surface soil layer. The licensee also evaluated different scenarios for residual radioactivity located below 1-m (3.3-ft). These included scenarios involving residual radioactivity remaining in bedrock overburden material being excavated and brought to the surface, and residual radioactivity in bedrock. The scenario involving possible exposure to residual radioactivity remaining in bedrock was found to be bounding for evaluating subsurface radioactivity. The residential farming scenario in general assumes light farming activities resulting in continuous exposure to residual radioactivity remaining at the site via multiple exposure pathways. Potential exposure pathways considered include direct external exposure from residual radioactivity in soil material, internal exposure from inhalation of airborne radionuclides, and internal exposure from ingestion of: (1) plant foods grown in the soil with residual radioactivity and irrigated with contaminated water; (2) meat and milk from livestock fed with contaminated fodder and water; (3) drinking water from a contaminated well; (4) fish from a contaminated pond; and (5) soil with residual radioactivity. The choice of a resident farmer scenario is considered to be reasonable and consistent with the generic scenario used for screening analyses described in Appendix C of NUREG-1727. Because the exposure pathways considered for the resident farmer scenario cover all the likely routes of exposures, it is unlikely that any other set of reasonably plausible human activities postulated for the site would result in a dose exceeding that calculated for the hypothetical farmer.

Elevated concentrations (i.e., above background) of Cs-137 have been measured in the Raystown Branch of the Juniata River, at one of the outfalls (near weir No. 1 and No. 6). To evaluate the need for remediating river sediment with residual radioactivity, the licensee performed a dose analysis assuming a recreational scenario involving exposure through external gamma radiation, and aquatic food and water ingestion. Use of a recreational scenario is considered appropriate, given the location of the sediments.

The licensee also plans to apply volumetric DCGLs to residual radioactivity remaining in buried pipes that will be left at the site. Volumetric DCGLs will be used (as opposed to surface DCGLs) because the licensee has committed to grouting these pipes with concrete. Thus, exposure to residual radioactivity will only result as the pipe and grout degrades. Accordingly, NRC staff considers exposure resulting from residential farming activities bounding for this type of material, as it is an unlikely growing medium.

2.5.3.3 Application of RESRAD

As previously stated, RESRAD, Version 6.1, was used to determine DCGL values for the residential farming scenario. The results from RESRAD for each of the 11 radionuclides in units of mrem per pCi/g were scaled to the 25-mrem TEDE limit, to determine an acceptable DCGL value. Table 4 lists DCGLs that will be used for residual radioactivity in soils, sediments, and other volume sources.

Table 4. SNEC DCGLs for Volumetric Sources
(Table taken from LTP)

Radionuclide	DCGL (pCi/g)	Radionuclide	DCGL (pCi/g)
Am-241	9.9E+00	Ni-63	7.5E+02
C-14	2.0E+00	Pu-238	1.8E+00
Co-60	3.5E+00	Pu-239	1.6E+00
Cs-137	6.6E+00	Pu-241	8.6E+01
Eu-152	1.0E+01	Sr-90	1.2E+00
H-3	1.3E+02		

Although default parameter values are provided in the RESRAD code, site-specific parameter values must be used to obtain meaningful results. Specific values for physical parameters (as defined in NUREG/CR-5512) usually vary widely from site to site and have the potential to greatly affect calculated doses. On the other hand, behavioral and metabolic parameters are representative of the scenario and critical group, and thus are less likely to vary from one site to another.

The licensee used an assumed contaminated area of 10,000 m^2 in the assessment. This bounds the size of the survey unit for which the proposed DCGL values can be used, ensuring that the dose limit will not be exceeded. NRC staff considers use of a contaminated area of 10,000 m^2 to be reasonable and acceptable because: (1) it is a site-specific value; (2) dose results are generally not sensitive to the size of the contaminated area when the area is large; and (3) dose results from the meat and milk pathways would not be affected by the choice of the contaminated area because it is assumed that 100 percent of the consumed meat and milk are contaminated. Even though NRC staff considers use of a 10,000 m^2 area to be appropriate, the volumetric DCGLs should not be applied to any survey units that exceed 10,000 m^2 without an appropriate demonstration that the dose limit will not be exceeded.

The depth of contamination used in the analysis was set to coincide with the assumed configuration of the contamination for the assumed land-use activity. For residual radioactivity remaining in surface soils above the water table, an assumed contaminated zone thickness of 1-m (3.3-ft) was used to coincide with the average depth to the maximum water table elevation. This assumed contaminated zone thickness should be appropriate because the dose from radionuclides most affected by surface-exposure pathways (e.g., gamma radiation) are not sensitive to the depth of contamination. For residual radioactivity in the subsurface [i.e., below 1-m (3.3-ft)], a contaminated zone thickness of 2-m (6.6-ft) was used, assuming the contamination remained in place and extends down into the bedrock. Therefore, the thickness of contamination in the subsurface should not exceed 2-m (6.6-ft) for the derived DCGLs to be applicable.

Because of uncertainties involved in selecting appropriate input parameter values, the potential effect that this variation could have in developing DCGLs was investigated. The licensee

selected a specific value for key parameters, to ensure an overall conservative estimate of potential doses, in consideration of whether the parameter is positively or negatively correlated with dose. In general, the approach the licensee used for selecting parameter values consisted of:

(a) The licensee performed probabilistic analyses to identify sensitive parameters. In the probabilistic analysis, default behavioral and metabolic parameter values that NRC established for screening proposes, to represent attributes of the average member of the critical group, were used as constants. Values for behavioral and metabolic parameters were primarily taken from NUREG/CR-5512. When values were not available from NUREG/CR-5512, RESRAD default values were used. Using NUREG/CR-5512 values is consistent with the guidance in NUREG-1727 and using the RESRAD default values should result in conservative dose estimates. Statistical distributions for parameters (i.e., physical parameters) treated stochastically in the analysis were established based on either site information or default statistical distributions established for use with the RESRAD code.

A site-specific range of distribution coefficients (Kds) was developed for a broad spectrum of material, so that the established DCGLs can be applied to a range of materials at the site. Specifically, site-specific Kds were developed for sediments, construction debris, fly ash, soil, clay material, and bedrock.

(b) Once the sensitive parameters were identified for each radionuclide, a deterministic analysis was used to determine the DCGL value for each radionuclide. The 25th or 75th quantile values of the parameter statistical distribution were used to represent sensitive parameters, depending on whether the parameter was determined to be negatively or positively correlated with dose. For sensitive parameters determined to be positively correlated with dose, the mean was used, if the mean was found to be greater than the 75th quantile of the distribution.

The NRC staff reviewed the approach used by the licensee for justifying parameters and found the approach to be acceptable and consistent with guidance provided in Appendix C of NUREG-1727. The NRC staff also conducted independent confirmatory analyses to ensure that the licensee's analysis was carried out as documented. Although the NRC staff found the overall licensee application of RESRAD for developing volumetric DCGLs to be acceptable, several items should be noted:

(a) For the excavation scenario for subsurface residual radioactivity, the licensee used a dilution factor of one-half, based on an assumption that a future hypothetical basement built on the site would intrude into equal parts (1-m) [3.3-ft] of soil, with residual radioactivity and clean soil. Given that a typical house basement is 3-m (10-ft), the licensee should have assumed that the basement would intrude into 2-m (6.6-ft) of soil with residual radioactivity. Accordingly, a dilution factor of one-third should have been used, instead of one-half. Use of a dilution factor of one-third would have resulted in lower or more restrictive DCGLs for the scenario; however, because the composite set of DCGLs for subsurface radioactivity is taken from a more conservative exposure scenario (i.e., involving residual radioactivity in the bedrock) use of the wrong dilution factor for this particular scenario is not a concern.

(b) With the exception of H-3 and C-14, K_d was not treated as a stochastic parameter in identifying sensitive parameters for scenarios involving subsurface radioactivity. Instead of treating K_d as a stochastic parameter to identify its importance and effect in developing DCGLs, the licensee used the lowest derived K_d value. This implicitly assumes that the K_d is negatively correlated with dose (i.e., the lowest K_d value will result in the highest dose). This assumption is not always valid when water-independent pathways are the primary contributor to dose. Through its own independent assessment, the NRC staff found the use of the lowest derived K values to be appropriate in this case, because the water-dependent pathways are the principal exposure pathways for the subsurface radioactivity.

(c) A uniform statistical distribution was assumed for the hydraulic conductivity for the saturated zone in conducting sensitivity analyses for scenarios involving subsurface radioactivity. Because of the widespread (i.e., several orders of magnitude) range of potential hydraulic conductivity values, a loguniform distribution is more appropriate. However, because the licensee modeled subsurface radioactivity by assuming no transport within the saturated zone (i.e., by using the Mass Balance model within RESRAD, which conservatively assumes all radioactivity reaching the water table is captured by the well), DCGLs for the subsurface will not be sensitive to the hydraulic conductivity in the saturated zone.

(d) In conducting its sensitivity analysis for subsurface radioactivity, the licensee set the maximum number of time integration points the code used to 1, in an effort to speed up the analysis. The maximum number of time integration points that the code can use is 17. Although use of a single time integration point may be appropriate for conducting sensitivity analysis, it may not be appropriate for carrying out the final analysis, as it affects the accuracy of the results. Thus, in carrying out its final analysis to develop subsurface DCGLs, the licensee should have set the maximum number of time integration points to 17. However, based on the NRC staff's own assessment, use of a higher number of points would have increased the accuracy by only a small amount, and, in each case, would have resulted in less restrictive (i.e., less conservative) DCGLs.

(e) Based on the approach the licensee proposed for developing DCGLs, mean or median parameter values were supposed to be used for those parameters that were found not to be sensitive to dose. In the deterministic analysis used to develop DCGLs for radioactivity in surface soils, the licensee used, for some parameters, a value other than the mean or median. The NRC staff found that use of these other parameter values did not affect the developed DCGLs, as they are insensitive to dose.

From its derivation of DCGLs for residual radioactivity in surface soils and the subsurface, the licensee developed a single composite list of DCGL values for the site, using the most limiting value calculated for each radionuclide. Table 5 shows the source of the analysis used to establish each DCGL. Thus, the DCGL for any two radionuclides could be based on two different and mutually exclusive scenarios. This provides greater assurance that the composite suite of DCGL values, when properly applied, is not likely to result in exposures exceeding the dose limit.

Table 5. Source of Analysis Used to Develop Volumetric DCGLs

Radionuclide	Assumed Location of Contamination	Radionuclide	Assumed Location of Contamination
Am-241	Subsurface	Ni-63	Surface
C-14	Subsurface	Pu-238	Subsurface
Co-60	Surface	Pu-239	Subsurface
Cs-137	Surface	Pu-241	Subsurface
Eu-152	Surface	Sr-90	Surface
H-3	Subsurface		

In review of the licensee's analysis to establish volumetric DCGLs, the NRC staff found that use of sensitive parameter values at the 25th and 75th quantile of their distribution provides high confidence that the dose limit will not be exceeded. DCGLs developed assuming surface contamination were found to result in calculated doses above the 75th quantile of the peak dose distribution. DCGLs developed assuming subsurface contamination were found to result in calculated doses above the 95th quantile of the peak dose distribution. For either surface or subsurface residual radioactivity, the peak mean dose (i.e., the peak of a plot of mean doses over time) was less than, or equal to, 20 mrem/year. Thus, the NRC staff concludes that the DCGLs in Table 4 of this SER should provide a high confidence that exposure to residual radioactivity in soils, sediments, and rubble remaining at the site will result in a dose below the dose limit.

2.5.4 Operational DCGLs

The LTP (Section 6.2.1) discusses the method to develop operational DCGLs for survey units where someone could be exposed to residual radioactivity in multiple media. Equation 6-1 of the LTP will be used to adjust DCGLs to ensure that the 25 mrem/yr dose limit is not exceeded in consideration of exposure to residual radioactivity from multiple media.

Application of surface DCGLs (i.e., Table 3) and/or volumetric DCGLs (i.e., Table 4) within the SNEC CV and the SSGS area need to be adjusted downward to account for potential exposure to the activated metal liner and concrete with volumetric contamination that is not addressed in the development of the DCGLs. For survey units within the CV where there is activated metal, DCGLs will need to be reduced to account for an estimated dose of 7.2 mrem attributable to the activation. Thus, the applicable DCGLs must be reduced by a factor of 28.8 percent. For survey units within the SSGS with concrete with volumetric contamination, the DCGLs will need to be reduced by a factor of 2.44 percent to account for an estimated dose of 0.611 mrem attributable to the volumetric contamination. In addition, for survey units within the SSGS with buried pipes, surface DCGLs will need to be reduced to account for the dose attributable to the pipes. This approach is acceptable because it appropriately accounts for exposure to all media within each survey unit and, thus, ensures that the total dose will not exceed the dose limit.

2.5.5 Gross-Activity DCGLs

The licensee proposes in the LTP the use of gross-activity DCGLs ($DCGL_{GA}$), generally for alpha or beta surface activity, for demonstrating compliance for areas with multiple radionuclides, rather than determining individual radionuclide activities. The approach provided in the LTP for developing $DCGLs_{GA}$ is consistent with the approach recommended by MARSSIM (NUREG-1575). Therefore, it is considered acceptable to the NRC staff. Although the method proposed for deriving the $DCGLs_{GA}$ is acceptable, the NRC staff will determine the acceptability of the relative fractions derived for each survey unit, as part of its review of the FSS to ensure compliance with the dose limit. Further, if $DCGLs_{GA}$ are not used, the NRC staff will ensure, using the equation below, that for survey units with multiple radionuclides the sum of fractions of the dose contribution for each radionuclide does not exceed 1.

$$\sum_{i=1}^{N} \frac{r_i}{DCGL_i} \leq 1$$

where:

$r_i \equiv$ concentration of radionuclide i

$DCGL_i \equiv$ DCGL for radionuclide i

$N \equiv$ total number of radionuclides in the survey unit

2.5.6 Surrogate Ratio DCGLs

The licensee in the LTP proposes the use of surrogate ratio DCGLs in areas where difficult-to-detect radionuclides may be present. The surrogate ratio DCGLs are computed on the basis of the activity ratio between difficult-to-detect radionuclides and the easy-to-measure radionuclides. Use of this approach can potentially save time and resources by enabling the licensee to measure the concentration of one radionuclide and relate the concentration of other radionuclides based on this ratio. This procedure also allows the DCGLs specific to HTD radionuclides, in a mix, to be expressed in terms of a single radionuclide that is more readily measured. The approach provided in the LTP for developing surrogate ratio DCGLs is consistent with the approach recommended by MARSSIM; therefore, NRC staff finds the approach to be acceptable.

The use of surrogate ratio DCGL values requires that a sufficient number of measurements, spatially separated throughout the survey unit, be made, to establish a consistent ratio. The key is establishing a consistent ratio; therefore, the NRC staff will ensure, as part of its review of the FSS, that appropriate ratios are used for each survey unit where surrogate ratio DCGLs are applied.

2.5.7 Elevated Measurement Comparison DCGLs

Area factors are needed for elevated measurement comparisons during scanning in Class 1 areas. The number of static measurements needs to be adjusted if the sensitivity of the scanning technique is not adequate for detecting levels of residual radioactivity below the

DCGLs. Area factors are also needed to identify small areas with elevated residual radioactivity that may require further investigation.

The licensee calculated area factors for both the resident farmer and the building occupancy scenarios. Area factors for the resident farmer scenario were computed by running the RESRAD computer code repeatedly with changing areas of contamination and other parameters (such as the fraction of plant, meat, and milk assumed to be taken from the site) affected by the area of contamination. Area factors were computed for all radionuclides of concern at the site, considering all potential pathways of exposure. The area factors for all radionuclides of concern at the site for the resident farmer scenario are listed in Table 5-15 of the LTP.

DCGLs for building surfaces were developed, using a version of the DandD code (i.e., Version 1), that did not allow a consideration of the effect of area on dose (the contamination source is conservatively assumed to be an infinite plane). Therefore, RESRAD-BUILD was used to compute area factors for the building occupancy scenario for all radionuclides of concern at the site. DCGLs developed using DandD were converted to an equivalent concentration for input into RESRAD-BUILD, and then RESRAD-BUILD was run adjusting the size of contamination source and using all default parameters. This approach was considered to be acceptable, because the main point was to determine a relationship between the size of the contamination and the dose. The area factors for the building occupancy scenario are listed in Table 5-15A of the LTP.

The NRC staff independently verified area factors, using RESRAD Version 6.2 for the resident farmer scenario, and RESRAD-BUILD Version 3.1 for building surfaces, and found no discrepancies. Therefore, the area factors proposed in the LTP are acceptable.

2.6 Site End Use

Section 50.82(a)(9)(ii)(E) requires a licensee to provide a description of the planned end use of the site if the licensee proposes to have its license terminated under restricted conditions. The licensee has proposed to have its license terminated with no restrictions on the use of the site, under the provisions of 10 CFR 20.1402 (unrestricted use criteria). Therefore, the licensee is not required to provide a description of the planned end use of the site.

2.7 Decommissioning Cost Estimate

The LTP, Revision 0, February 2000, Section 7.1, contained a total cost estimate of $35.5 million for the SNEC Facility Decommissioning Project. According to the LTP, unit cost factor approaches for the estimate, collateral costs, and costs for packaging, transporting, and decontaminating or direct disposal of radioactive waste at a licensed facility were used in the calculations.

The licensee indicated in a response to a NRC staff request for additional information that, based on then current expenditures, $37.5 million would be spent on the decommissioning project by December 31, 2000. NRC reviewed and accepted the licensee's response, in conjunction with the merger between FirstEnergy Corp. and GPU, Inc. NRC documented the adequacy of decommissioning funding assurance for the SNEC Facility in the "Order Approving Application Regarding Proposed Merger of GPU, Inc., and FirstEnergy Corp. - Saxton Nuclear

Experimental Facility (TAC No. MB0215)," dated March 7, 2001. SNEC provided an estimate of $14.8 million (in year-2000 dollars, without contingency) for the remainder of the work at that time. This included remediation of the SSGS and CV. This would have made the total decommissioning cost about $52.3 million. SNEC estimated that the cost associated with the removal of concrete in the CV and the necessary extension of the project schedule would add about 90 percent of the projected $14.8 million, and the balance of 10 percent would be required, to complete the decommissioning work for the SSGS footprint. SNEC also indicated the commitment of the Operating Companies (the SNEC owners) to fund any increase in the SNEC decommissioning costs out of their general funds.

In Revision 1 (dated September 26, 2002) and Revision 2 (dated December 16, 2002) to the LTP, SNEC indicated that the cost associated with the CV concrete removal project has exceeded previous assumptions. Revision 2 to the LTP shows the current overall project cost estimate is approximately $63 million. As of July 31, 2002, approximately $51 million has been spent and thus the remaining cost to complete decommissioning is approximately $12 million. Table 7-1 in Revision 2 to the LTP provides a breakdown of the $12 million. Revision 2 to the LTP indicates that the licensee's (GPU Nuclear's) Letter E910-01-004, dated February 19, 2001, "Parent Guarantee for Decommissioning Funding" committed the SNEC owners to carry out the required activities or setup a trust fund in favor of the NRC in the event the licensee failed to perform the required decommissioning activities. The amount of this guarantee is $20 million. Thus adequate funding exists to complete the SNEC Facility decommissioning project.

2.8 Changes to License Conditions

The licensee has requested the addition of a new license condition to the NRC Amended Facility License No. DPR-4. The new license condition would incorporate the NRC-approved "License Termination Plan, Revision 1, dated September 26, 2002, and Decommissioning Environmental Report, Revision 2, dated September 23, 2002, supplemented by Revision 2 to the LTP, change pages dated December 16, 2002," (referred to as the SNEC Facility LTP) into Amended Facility License No. DPR-4 and allow the licensee to make certain changes (to the NRC-approved LTP) without prior NRC review or approval. The first paragraph of the new license condition as approved by the NRC staff differs from that proposed by the licensee. The NRC staff added the words to the license condition "and maintain in effect all provisions of" to clarify that the licensee's responsibility goes beyond the implementation of the LTP by maintaining it in effect. During a telephone conversation on January 30, 2003, between the NRC project manager for the SNEC facility and Mr. G. A. Kuehn, Jr., Vice President SNEC and Program Director SNEC Facility, this change was discussed and the licensee agreed to the additional wording. The new license condition would appear as follows:

> <u>License Termination</u>
>
> The licensee shall implement and maintain in effect all provisions of the approved SNEC Facility License Termination Plan as approved in the SER dated March 28, 2003. The licensee may make changes to the SNEC Facility License Termination Plan without prior approval provided the proposed changes do not:
>
> (i) Involve a change to the Technical Specifications or require NRC approval pursuant to 10 CFR 50.59;

(ii) Violate the criteria of 10 CFR 50.82(a)(6);

(iii) Reduce the coverage requirements for scan measurements;

(iv) Increase the derived concentration guideline level (DCGL), developed to meet the requirements of 10 CFR 20.1402, and related minimum detectable concentrations for both scan and fixed measurement methods;

(v) Use a statistical test other than the Sign test or Wilcoxon Rank Sum test for evaluation of the final status survey;

(vi) Increase the radioactivity level, relative to the applicable DCGL, developed to meet the requirements of 10 CFR 20.1402, at which investigation occurs;

(vii) Increase the Type I decision error; and

(viii) Decrease an area classification (i.e., impacted to non-impacted; Class 1 to Class 2; Class 2 to Class 3; or Class 1 to Class 3).

These restrictions on the changes that the licensee can make to the LTP without prior NRC approval will help to ensure that the licensee does not make changes to the LTP that could possibly affect the ability to demonstrate that the radiological release criteria for license termination were met. The NRC staff concludes that authorizing the licensee to make certain changes during the final site remediation is acceptable, subject to the above listed conditions.

2.9 Summary of Areas Requiring Further Validation

As a result of the NRC staff's review of the LTP, the NRC staff has determined that there are several areas related to the LTP that will need to be validated, either as part of the staff's ongoing inspection effort at the SNEC site and/or during the FSS. These areas are:

- The NRC staff has determined that the licensee has identified the remaining dismantlement activities necessary to complete decommissioning of the facility, as required by 10 CFR 50.82(a)(9)(ii)(B). Further, the NRC staff has determined that these activities can be completed in accordance with 10 CFR 50.59. However, the NRC staff will confirm, during in-process inspections, that such activities are conducted in accordance with approved procedures and NRC regulations under Parts 20 and 50. (Section 2.2 of this SER)

- The licensee has committed to follow the DQO Process and the SNEC Facility Decommissioning Quality Assurance Plan to conduct any further site characterization work, and to document in the survey release record the characterization data, the FSS data, and other information needed to demonstrate compliance with the site release criteria. Additionally, the licensee committed to provide in the FSS documentation for each survey unit a description of any changes in the initial survey unit assumptions

relative to the extent of residual radioactivity, and verification of the final classification designation, with classification history. This information and supporting data will be available on-site for NRC review during inspections. NRC expects to review this information for each survey unit to assess the adequacy of the FSS design and conduct. This review will be performed as part of the inspection process and evaluation of the FSS. (Section 2.1of this SER)

- The FSS will be conducted using guidance in MARSSIM to demonstrate compliance with the criteria specified in Part 20, Subpart E, for unrestricted release of the SNEC site. The types of surveys and sampling described for complete characterization will require further NRC staff validation, as part of the inspection process, to ensure that the methodology and data are adequate when this information becomes available. (Section 2.1 of this SER)

- The licensee determined that complete removal of the interior CV concrete would be required before the FSS for the interior of the CV liner. Consequently, the licensee characterized the interior steel surfaces of the CV liner as the concrete was removed. Cs-137 in excess of the DCGL for building surfaces was identified as the principal surface contaminant, and activation of the liner has occurred where the reactor vessel was once located. The licensee has committed to fully characterize the CV liner. NRC will review the characterization and activation analysis of these areas, including the determination of specific radionuclide concentrations, as part of NRC's ongoing inspection efforts. (Sections 2.1.1.1 and 2.1.1.3 of this SER)

- Survey procedures for embedded piping and components have not been completely described – although the LTP provides specific commitments the licensee intends to satisfy. These commitments include: surface-scan coverage, according to the class of survey unit, depending on the accessible surface area; taking 30 static measurements at accessible points; assessing scale and sediment samples as appropriate; and when practical, filling embedded pipes with grout or concrete. When remaining sections of system piping allow less than 30 measurement points, the licensee will assess whether the embedded piping can easily be removed or prepared for FSS. In this case, the licensee will use the RSSI process as a basis for biased scanning and sampling of the radionuclide concentration present on/in the embedment, to ensure that the release criteria are met. Additionally, for surveying embedded and buried piping, the LTP proposes the use of *in-situ* NaI detectors. The survey methods for piping and components, as presented and referenced in the LTP, are recognized as an acceptable general approach by the NRC staff, but will require further validation during the FSS, to confirm that residual contamination levels meet the release criteria. (Sections 2.1.1.2 and 2.4 of this SER)

- Surface and subsurface soil at the SNEC site is contaminated from licensed operations, accidental spills, and long-term accumulation of material in the soil from effluent releases. The licensee has committed not to use gamma bore-logging as a stand-alone technique for the characterization or FSS of soil but rather as a complement to sampling. NRC will review the licensee's characterization and FSS for survey units involving soil to ensure that sampling was performed and that gamma-bore logging was not used primarily for demonstrating regulatory compliance. Additionally, NRC will

review the FSS results for subsurface soil sampling and determine if the results were evaluated consistent with dose modeling assumptions. (Sections 2.1.1.4 and 2.4 of this SER)

- NRC will review the licensee's sediment data in the FSS report, as part of the inspection process, to determine whether it is acceptable, and to verify that the LTP requirements that apply to surface-soil sampling and analysis and data assessment were applied to sediments. (Section 2.1.1.7 of this SER)

- NRC will review the licensee's exposure rate data during the RSSI process and in the FSS report as part of the inspection process to determine whether it is acceptable and whether instrument use and data analysis are consistent with MARSSIM guidance. (Section 2.1.1.9 of this SER)

- The licensee stated, in Section 6.4 of the LTP, that remediation of all structural surfaces below the DandD code screening levels would ensure that any residual radioactivity remaining at the site would not result in any significant impact to public health and safety. Through the inspection process, NRC will review the licensee's ALARA evaluation, as required by 10 CFR 20.1402. (Section 2.3 of this SER)

- In Chapter 5 of the LTP, there are several references to advanced survey technologies and methods such as *in-situ* gamma spectroscopy and semi-automated equipment, for possible use during the conduct of the FSS. NRC, through the inspection process, will review the procedures for using such technologies and methods; the acceptability for use in specific survey units; and the statistical techniques for evaluating the data. (Section 2.4 of this SER)

- For the conduct of the FSS, the MDCs for both scanning and static measurements will be derived specific to the instrumentation used and conditions of performance present at that time. The licensee has committed to include, in each survey unit design package, a description of the instruments, methods, calibration, operational checks, coverage, and sensitivity for each medium and radionuclide. NRC will review the licensee's determination of specific MDCs, for both scanning and static measurements, and other instrumentation-related parameters, for inclusion in the FSS design packages, as part of NRC's ongoing inspection process. (Section 2.4 of this SER)

- During 2002, the licensee performed the FSS for several steel surface areas of the CV steel shell, in support of installation of steel I-beams, which were designed to stabilize the CV shell during concrete removal. The FSS had to be done then, since the I-beams remain in place permanently, and therefore the CV steel surface behind them is no longer accessible for surveying. According to the licensee, the vast majority of surface contamination was removed before the FSS. However, no information was provided in the LTP on the radiological status of the steel surface as the result of post-cleaning surveys that were performed to verify that the cleaning effort was successful. The licensee stated, in Section 5.5.3.4.9 of the LTP, that these areas have been surveyed "at risk," in that they have been surveyed before NRC approval of the SNEC LTP. NRC, through the inspection process, will review the post- remediation characterization data and the FSS to ensure that: (1) data quality, investigations of anomalies, and survey

design are adequate; (2) the appropriate DCGLs for surface contamination were applied; (3) areas of elevated activity were evaluated against the appropriate $DCGL_{EMC}$; (4) any scanning factors used were acceptable; (5) the entire steel surface was scanned; (6) static measurements were properly located; and (7) the release criteria are satisfied for each survey unit. (Section 2.4 of this SER)

- Data management requirements are discussed in Section 5.5.5 of the LTP. Specifically, the section defines which data may be used as FSS data. Some of the elements of the data management system are described in the LTP. However, the methodology by which the licensee will integrate the data and demonstrate regulatory compliance is not described in the LTP. This aspect will be the focus of future NRC in-process inspections. (Section 2.4 of this SER)

- When reviewing FSS results, NRC will examine whether the licensee has measured and/or accounted for each of the radionuclide contaminants (Table 6-1 of the LTP), when presenting dose-compliance information for each survey unit. Additionally, whenever the licensee accounts for HTD radionuclides through surrogate analyses, NRC will examine if the licensee has verified that the activity ratios (activity ratio for difficult to detect radionuclides to easy-to-detect radionuclides) used during the FSS were valid, in accordance with Sections 5.2.3.2.3 and 5.2.7.6 of the LTP. In Sections 5.2.3.2.3 and 5.2.3.2.4 of the LTP, the licensee has committed to documenting in each survey package a radionuclide mix justification. NRC, when reviewing these packages, will confirm that such a justification was provided. (Section 2.4 of this SER)

- For survey units within the CV where there is activated metal, NRC staff will ensure that the applicable DCGLs are reduced by a factor of 28.8 percent. For survey units within the SSGS concrete with volumetric contamination, NRC staff will ensure that the applicable DCGLs are reduced by a factor of 2.44 percent.

- For survey units in which $DCGLs_{GA}$ are applied, the NRC staff will determine the acceptability of the relative fractions derived for each survey unit. If $DCGLs_{GA}$ are not used, NRC staff will ensure that for survey units with multiple radionuclides the sum of fractions of the dose contribution for each radionuclide does not exceed 1.

- For survey units in which surrogate ratio DCGLs are applied, the NRC staff will ensure that appropriate ratios are used for each survey unit.

3.0 STATE CONSULTATION

In accordance with NRC regulations, the Pennsylvania State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATIONS

In accordance with the requirements of 10 CFR 50.82(a)(9)(ii)(G), a licensee is required to provide a supplement to the environmental report, pursuant to 10 CFR 51.53, describing any new information or significant environmental changes associated with the licensee's proposed license termination activities. The licensee, by letter dated February 2, 2000, submitted its

SNEC Facility LTP. In the LTP, the licensee incorporated, by reference, a report entitled "Decommissioning Environmental Report, Revision 1," dated February 2, 2000. The licensee also submitted, by letter dated September 23, 2002, "Decommissioning Environmental Report, Revision 2." Consistent with the decommissioning rule that appeared in the *Federal Register* notice dated July 29, 1996 (Vol. 61, No. 146, pp. 39283-39284), and pursuant to 10 CFR 51.21, 51.32, and 51.35, the NRC has prepared an environmental assessment (EA) to determine the adequacy of radiation release criteria and the FSS presented in the LTP.

On the basis of this EA, NRC has concluded that this licensing action would not adversely affect the environment and does not warrant the preparation of an Environmental Impact Statement. Accordingly, it has been determined that a Finding of No Significant Impact is appropriate.

A "Notice of Issuance of Environmental Assessment and Finding of No Significant Impact" was published in the *Federal Register* on March 20, 2003 (68 FR 13733).

5.0 CONCLUSIONS

The NRC staff has concluded, on the basis of the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed activities; (2) such activities will be conducted in compliance with the Commission's regulations; and (3) the issuance of the amendment will not be inimical to the common defense and security nor the health and safety of the public.

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