May 13, 2003

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SUBJECT: SAXTON NUCLEAR EXPERIMENTAL CORPORATION — CORRECTIONS TO THE SAFETY EVALUATION REPORT FOR THE AMENDMENT APPROVING THE SAXTON LICENSE TERMINATION PLAN (TAC NO. MA8076)

Dear Mr. Kuehn:

Please find enclosed replacement pages with corrections for the NRC Safety Evaluation Report (SER) that was issued with Amendment No. 18 to Amended Facility License No. DPR-4 on March 28, 2003. Also enclosed is a short summary sheet which describes the changes. The changes either correct errors or clarifies the SER. None of these changes have any impact on the conclusions reached in the SER.

If you have any questions, please contact me at 301-415-1127.

Sincerely,

/RA/

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: As stated

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Summary of Changes to
SER for Amendment No. 18
Saxton Nuclear Experimental Facility

1. Page 6, Section 2.1, last paragraph, fourth sentence. Change “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.” to “The final classification designation will be included along with a summary of the results of the FSS, and a description of any changes in the initial survey unit assumptions in the FSS report for each survey unit.” This change corrects an error made in referencing material from the licensees’ license termination plan (LTP).

2. Page 9, Section 2.1.1.1, ninth paragraph, fifth line. Change “...up to the river discharge point is designated as a Class 2 because...” to “...up to the river discharge point is designated as a Class 3 because...” This corrects a typographical error.

3. Page 19, Section 2.1.1.5.4.4, first paragraph, third sentence. Change “960 pCi/L” to “760 pCi/L.” This corrects a typographical error.

4. Page 23, Section 2.1.1.9, “Exposure Rate Survey,” first Paragraph. All exposure rate values should be changed to “µ” (i.e, micro) to correct a typographical error. Change the last sentence from “Additionally, Figure 2-28 of the LTP indicates an exposure rate of 0.05 mSv/hr (5 mrem/hr) in the SSGS Intake Tunnel.” to “Additionally, Intake Tunnel dose rates ranged from 2 to 4 µR/hr. This change corrects an error made in referencing material from the licensees’ LTP.

5. Page 25, Section 2.2, third paragraph, second to the last sentence. Change “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)].” to “In addition, the licensee estimates the total radiation exposure to complete the remaining site dismantlement tasks to be 0.03 person-sievert (Sv) [3 person-roentgen-equivalent-man (rem)].” This change corrects an error made in referencing material from the licensees’ LTP.

6. Page 36, Section 2.5.3.2, last paragraph, first sentence. Change “The licensee also plans to apply volumetric DCGLs to residual radioactivity remaining in buried pipes that will be left at the site.” to “The licensee also plans to apply volumetric DCGLs to residual radioactivity remaining in buried contaminated pipes with a diameter greater than 30.5 cm (12 in) that will be left at the site.” This change clarifies the sentence.

7. Page 40, Section 2.5.4, second paragraph, fourth sentence. Change “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.” to “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 imbedded piping.” This change clarifies the sentence.

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 will be included along with a summary of the results of the FSS, and a description of any changes in the initial survey unit assumptions 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 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 3 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 hydraulic conductivity range of $1.13x10^{-4}$ to $4.94x10^{-5}$ cm/sec for the overburden wells and a range of $2.88x10^{-3}$ to $4.94x10^{-5}$ 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 760 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 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 µSv/hr (4 µrem/hr), at the interior of the PENELEC Switchyard Building, to 0.28 µSv/hr (28 µrem/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, Intake Tunnel dose rates ranged from 2 to 4 µR/hr.

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

[(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 0.03 person-sievert (Sv) [3 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 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 contaminated pipes with a diameter greater than 30.5 cm (12 in) 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 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 imbedded piping. 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.