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July 16, 2001

Mr. Larry W. Camper
Decommissioning Branch Chief
U.S. Nuclear Regulatory Commission
Mail Stop 7 F27
Washington, DC 20555-0001

PROJECT NUMBER: 689

Dear Mr. Camper:

The Nuclear Energy Institute is submitting the enclosed questions and answers in support of the agency's license termination guidance document consolidation project. The enclosure includes questions one through ten, along with proposed answers with basis, developed in an effort to clarify existing guidance associated with demonstrating compliance with NRC's license termination rule.

These questions were developed by NEI's License Termination Task Force, which contains representatives with health physics expertise from actively decommissioning, commercial U.S. reactors. The submittal was developed using the procedure discussed at the NRC guidance consolidation workshop held June 1, 2001. Copies of this procedure were provided at the public workshop.

Each question has an identified industry sponsor. Contact information for each sponsor is included. Should you or your staff need additional information or clarification, please contact me at (202-739-8034 or phg@nei.org).

Sincerely,

Paul H. Genoa

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Enclosure

Draft LTP Questions and Answers

7-16-01

S: Final Status Survey
C: Characterization
M: Dose Modeling

QUESTION 1 - (C): During the process of developing an initial radionuclide profile for characterizing commercial light water reactor sites, which nuclides are typically considered?

ANSWER: The nuclides that need to be considered are listed below:

Contamination Suite		Activation Suite
H-3	Sb-125	H-3
C-14	Cs-134	C-14
Mn-54	Cs-137	Fe-55
Fe-55	Eu-152	Ni-63
Co-57	Eu-154	Co-60
Co-60	Ce-144	Cs-134
Ni-59	Pu-238	Cs-137
Ni-63	Pu-239/240	Eu-152
Sr-90	Pu-241	Eu-154
Nb-94	Am-241	Eu-155
Tc-99	Cm-243/244	

BASIS: NRC guidance found in NUREG/CR-3474, "Long-Lived Activation Products in Reactor Material", and NUREG/CR-0130, "Technology, Safety and Cost of Decommissioning", identifies a suite of radionuclides which should be considered for initial characterization. Industry experience, based on Historical Site Assessment, radioactive waste profiles, and activation analysis, has identified a subset of those radionuclides that must be considered. The identified list of radionuclides was determined to be significant in the license termination plans submitted to date.

Other radionuclides might be present, but in such low relative abundances due to low production or radioactive decay, that they may not be detectable at the time characterization surveys are performed.

For additional details on this subject refer to the 2001 Annual Health Physics Society presentation: "Nuclide Suites for the Decommissioning of the Maine Yankee Atomic Power Plant"[1].

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[1] J. Darman, "Nuclide Suites for the Decommissioning of the Maine Yankee Atomic Power Plant," Health Physics, Vol. 80, No. 6, S142, June 2001.

QUESTION 2 - (C): When developing gross DCGLs for the Final Status Survey, which detected radionuclides can be de-selected from further consideration?

ANSWER: For radionuclides that are detectable, it is acceptable to de-select those that collectively contribute less than 10% of the total dose[2].

BASIS: In order to avoid expending inordinate resources on insignificant dose contribution during implementation of the Final Status Survey, a reasonable de-selection criteria is needed. A criterion that meets this objective and is both reasonable and fully protective of the public is 10% of the total dose from the mixture. In developing this criterion, the following regulatory guidance was evaluated:

1. Provisions for disregarding radionuclides in dose assessment are contained in 10 CFR 20.1204(g) for internal dose assessment:

   When a mixture of radionuclides in air exists, licensees may disregard certain radionuclides in the mixture if–

   (1) The licensee uses the total activity of the mixture in demonstrating compliance with the dose limits in §20.1201 and in complying with the monitoring requirements in §20.1502(b), and

   (2) The concentrations of any radionuclide disregarded is less than 10 percent of its DAC, and

   (3) *The sum of these percentages for all of the radionuclides disregarded in the mixture does not exceed 30 percent.*

2. 10 CFR 20.1502 entitled "Conditions Requiring Individual Monitoring of External and Internal Occupational Dose":

   Each licensee shall monitor exposures to radiation and radioactive material at levels sufficient to demonstrate compliance with the occupational dose limits of this part. As a minimum –

   (a) *Each licensee shall monitor occupational exposure to radiation from licensed and unlicensed radiation sources under the control of the licensee and shall supply and require the use of individual monitoring devices by—*

      (1) *Adults likely to receive, in 1 year from sources external to the body, a dose in excess of 10 percent of the limits in §20.1201(a),*

      (2) Minors likely to receive, in 1 year, from radiation sources external to the body, a deep dose equivalent in excess of 0.1 rem (1 mSv), a lens

[2] 10 CFR 20.1402, " Radiological criteria for unrestricted use"

dose equivalent in excess of 0.15 rem (1.5 mSv), or a shallow dose equivalent to the skin or to the extremities in excess of 0.5 rem (5 mSv);

(3) Declared pregnant women likely to receive during the entire pregnancy, from radiation sources external to the body, a deep dose equivalent in excess of 0.1 rem (1 mSv); and

(4) Individuals entering a high or very high radiation area.

(b) Each licensee shall monitor (see §20.1204) the occupational intake of radioactive material by and assess the committed effective dose equivalent to–

(1) Adults likely to receive, in 1 year, an intake in excess of 10 percent of the applicable ALI(s) in table 1, Columns 1 and 2, of appendix B to §§20.1001-20.2402;

3. NRC Regulatory Guide 1.109 requires an assessment of population dose from radiation exposure pathways. The Regulatory Guide notes that in addition to the generic pathways identified in the guide, other pathways must be considered if they are "significant." A "significant" pathway is one in which the dose contribution is at least 10 percent of the total. Therefore, a pathway that contributes less than 10 percent of the total population dose may be considered not significant and the dose does not have to be specifically accounted for.

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QUESTION 3 - **(M)**: For the building occupancy scenario, what dose modeling pathways need to be considered for grouted pipe embedded in buildings (e.g. walls, ceilings, and floors)?

ANSWER: Only the direct dose contribution to building occupants needs to be considered for grouted, embedded pipe.

BASIS: In the building occupancy scenario the DCGLs are based on potential exposure pathways that include:

- Direct exposure to surface contaminants
- Inhalation of resuspended surface contaminants
- Ingestion of surface contaminants.

The presence of embedded piping that is contaminated with radioactive materials may contribute an additional gamma dose to building occupants that is not accounted for by the DCGL. This additional gamma dose contribution may be determined by direct measurement in occupied spaces or by analysis using characterization information on the source term in the pipe. Once the additional dose contribution is known the DCGLs may be adjusted downward such that the total dose to building occupants is less than 25 mrem/y.

The grouting of the pipe would ensure that any beta or alpha emitters would not contribute to the dose since the contaminants would be sealed in the pipe and these radiations would not be expected to escape through the pipe wall.

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QUESTION 4 - (M): What is an acceptable level of residual surface contamination on grouted, embedded pipe?

ANSWER: In addition to accounting for the direct shine dose pathway as discussed in question 3, grouted, embedded pipe containing an average contamination level of 100,000 dpm/100cm^2 and a maximum of 1,000,000 dpm/100cm^2 is acceptable.

BASIS: Using an extremely conservative scenario where an individual is in contact with an embedded pipe for 100 hours per year at the maximum contamination level, the calculated annual dose is less than 10 mrem.

For this scenario, the pipe could be removed from the wall, floor, etc., and would be handled by a worker without the benefit of concrete shielding. A single worker may handle this pipe for several hours in the course of removal from the building and final disposal. Placing a limit of 10^6 dpm/100 cm^2 for gamma emitters on the inside of the pipe will assure that the renovation worker is protected. This level of contamination in a 4 inch schedule 40 pipe will produce a dose rate of approximately 9.5×10^{-2} mrad/h at a distance of 6 inches. The contamination was conservatively assumed to be 100% Co-60 for this analysis. A pipe handling scenario in which the worker carries contaminated pipe for 100 hours in a year would result in a potential exposure of less than 10 mrem. Thus, this scenario does not have to be considered for grouted, embedded pipe that meets the stated contamination limit.

DOSE ANALYSIS INPUT ASSUMPTIONS: It was assumed that the interior surface of the pipe was uniformly coated with 10^6 dpm/100 cm^2 of Co-60. The pipe was 4" schedule 40, and was 13 feet (4 meters) long. The receptor was assumed to be at the midpoint of the pipe and at a distance of 6" from the pipe surface. Shielding credit was taken for the pipe wall but not for the grout material in the pipe. This source term produces a gamma energy release rate of 6.7×10^5 MeV/s-m of pipe length or 8.16×10^7 MeV/s-m^3 of pipe volume. The resulting dose rate is 9.5×10^{-2} mrad/h.

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QUESTION 5 - **(C)**: What methods may be used to survey embedded pipe?

ANSWER: A recent study by EPRI evaluated several techniques that proved acceptable for surveying the radiological contamination on the inside of embedded pipe. Measurement techniques included pipe crawlers, gamma-ray scanners, dose rate measurements with dose-to-curie computations, scraping samples with radiochemical analyses, and smear samples with radiochemical analyses.

BASIS: The pipe crawler uses a beta sensitive detection system that is inserted into the pipe with a cable. Spacers keep the detectors at a fixed distance from the pipe wall. Measurements can be made at various points within the pipe. Scaling factors based on a laboratory radiochemistry analysis of the deposited material can be applied to the measurements to provide radionuclide quantities in the pipe.

The gamma-ray scanner uses a calibrated, collimated high-purity Ge spectrometer to make external measurements on the pipe. This gamma-ray scanning yields an average concentration over the length of the pipe. The sensitivity of this method may be limited by the thickness of concrete between the pipe and the detector. Some radionuclide identification is possible and scaling factors can be applied as discussed above for the pipe crawler.

The dose rate measurements are also made on the external surface of the walls or floors containing the embedded pipe using a sensitive gamma detector capable of reading in the μR/hr range. The dose rate readings may be used directly in determining compliance with the dose criteria or used to make dose-to-curie conversions based on other measurements providing radionuclide identification.

Radionuclide identification for the contamination in the pipe may be accomplished by smear or scraping samples and radiochemical analysis. The EPRI report compared radionuclide ratios determined by smears and by scrapings with those found by etching the surface of the pipe. They concluded that either of these techniques yields nuclide mixes that are representative of the average total deposits.

(Reference: Cline, J. E., "Embedded Pipe Dose Calculation Method", EPRI Report No. 1000951, November, 2000)

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QUESTION 6 - (C): What are acceptable methods to employ in the determination of soil k_d values used in site-specific DCGL determination?

ANSWER: As indicated in NUREG-1727, Appendix C, Section 7.2.3, site-specific k_d values for soil may be determined by the following:

1) Identify site soil type(s). These may be found through historical records, literature sources, or direct geological investigation[3].

2) Using the soil type(s), identify the k_d range using available literature[4].

3) When using deterministic dose modeling codes, compare the k_d ranges with the default k_d value. If the range encompasses the default, then utilize the default. If, however, the default falls outside the range, then site-specific values may need to be developed. When using probabilistic dose modeling, which supports the direct input of a range of values, enter the values directly.

BASIS: The D&D default k_d values have been reviewed and approved by the NRC staff. The RESRAD probabilistic default input parameter set was developed as a joint effort with NRC staff and Argonne National Laboratory and has been approved for use in probabilistic dose assessment. Both default and site-specific k_d input are acceptable.

Provided below are two examples of regulatory guidance in which there is discussion on the determination of k_d values:

"The only geochemical parameter used in D&D is the element-specific partition coefficient. As documented in NUREG/CR-5512, Volume 3, the partition coefficients at a site are generally dependent on geochemical conditions and are generally independent of soil classification. If the licensee has used the default distributions, the staff should evaluate whether the defaults are inconsistent with known or expected conditions at the site." [5]

"For the physical parameters describing geochemical conditions (i.e., distribution coefficients), the licensee should use values that are consistent with the D&D default values, as long as the values are not inconsistent with known or expected site conditions. Justification supporting the values should be based on sensitivity analyses." [6]

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[3] ASTM D 4319-83, "Distribution Ratios by the Short-Term Batch Method"
ASTM D 4646-87, "24-h Batch-Type Measurement of Contaminant Sorption by Soils and Sediments"

[4] "Understanding Variation in Partition Coefficient, Kd, Values, Volumes I and II, EPA 402-R-99-004A, 8/99.
http://www.epa.gov/radiation/technology/partition.htm#voli

[5] "NMSS Decommissioning Standard Review Plan", Appendix C 7.4.1

[6] "NMSS Decommissioning Standard Review Plan", Appendix C 7.4.2

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QUESTION 7 - (C): What are acceptable methods to employ in the determination of concrete k_d values used in site-specific DCGL determination?

ANSWER: As indicated in NUREG-1727, Appendix C, Section 7.2.3, site specific k_d values for concrete may be determined by the following:

1) Perform a literature search (Krupka, K.M., and R.J. Serne, 1998)[7].

2) If k_d values for the radionuclide(s) of interest are not found in the literature, evaluate elements of similar chemical characteristics.

3) If no correlation can be found, a sensitivity analysis may be performed to determine a reasonably conservative k_d value.

4) If sensitivity analysis shows this k_d value to be critical, then empirical evidence may be required

BASIS: NUREG-1727 allows site-specific values of parameters to be used, with the level of justification consistent with the sensitivity of the parameter to the dose results. Site specific parameters may be based on measured data or on generic information consistent with the site conditions. The justification should demonstrate that the site-specific values selected are not inconsistent with the known or expected characteristics of the physical site being modeled.

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[7] "Effects on Radionuclide Concentration by Cement/Ground-Water Interactions in Support of Performance Assessment of Low-Level Radioactive Waste Disposal Facilities," NUREG/CR-6377, PNNL-14408

QUESTION 8 - (S): Is it acceptable to define the process and acceptance criteria for demonstrating that instruments are sufficiently sensitive rather that providing the sensitivities for all instruments in the LTP?

ANSWER: Yes, it is acceptable to define the process and acceptance criteria rather than provide a comprehensive list of all the instruments.

BASIS: MARSSIM (Section 5.5.2.6) defines investigation levels for scan and fixed point measurements for the different survey unit classifications in terms of percentages of the DCGL. Rather than provide specific individual instrument sensitivities in the LTP, it is acceptable to state that performance based criteria will be used to demonstrate that the instruments employed have the required minimum sensitivity to detect residual radioactivity at the MARSSIM investigation levels.

A licensee may wish to use new technology or different instruments than those that were investigated at the time of the submittal of the LTP. By committing to use only instruments that can be shown to perform with sensitivities that allow detection of residual radioactivity at the levels that correspond to the MARSSIM investigation levels, a licensee can show that surveys will be conducted so as to meet the Final Status Survey Data Quality Objectives (DQOs).

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QUESTION 9 - (C): Is characterization data required in the LTP for structures, components, and soils that will be removed from the facility prior to license termination?

ANSWER: No. In general, the only characterization data necessary is that which supports the financial and environmental aspects of the license termination. However, detailed characterization data need not be included in the License Termination Plan (LTP) for structures, components, and soils that will be removed from the facility.

BASIS: By definition, residual radioactivity associated with structures, components, and soils that will be removed from the facility cannot contribute to public dose controlled under 10 CFR 20,1402, "*Radiological Criteria for Unrestricted Use*." In addition, the collection of detailed characterization data from materials that will be removed from the facility exposes workers to unnecessary radiological and industrial safety risks without commensurate benefit.

These decommissioning activities are conducted under existing Radiation Protection, Safety, and Waste Management Programs. These programs are well established and are frequently inspected by the NRC. Activities conducted during decommissioning do not pose any greater radiological or safety risk than those conducted during operations, especially those during major maintenance and outage evolutions.

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[8] Enclosure 1 to CY-01-047

QUESTION 10 - (C): How much characterization data is required, in addition to the Historic Site Assessment, to support initial classification where structures, components, and soils require remediation?

ANSWER: In general, areas classified as Class 1 do not require characterization data to support that classification.

BASIS: In conjunction with the Historical Site Assessment, sufficient data must be provided in the License Termination Plan to support initial classification of survey areas. Class 1 areas, however, typically contain significant radiological and industrial safety risks. The collection of comprehensive characterization data will provide minimal benefit to future project decision making, while exposing workers to potential risk. In addition, decommissioning activities are conducted under existing Radiation Protection, Safety, and Waste Management Programs. These programs are well established and are frequently inspected by the NRC. Activities conducted during decommissioning do not pose any greater radiological or safety risk than those conducted during operations, especially those during major maintenance and outage evolutions

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