DL-012797-13

United States Department of the Interior

U.S. GEOLOGICAL SURVEY
Reston, Virginia 20192

Mail Stop 926A
January 27, 1997

United States Nuclear Regulatory Commission
Region II
101 Marietta Street, N. W., Suite 2900
Atlanta, GA 30323-0199

Attn: Jay L. Henson
Materials License Reviewer
Division of Nuclear Material Safety

License No. 45-15923-01
Expiration Date: November 30, 2000

Dear Mr. Henson:

Enclosed you will find two copies of Mr. Michael Terpilak's, CHP, final survey report of rooms 3C429 and BC214C of the John Wesley Powell Building of the U.S. Geological Survey, in Reston, VA. This report was prepared at our request for the purpose of determining if those rooms may be released for unrestricted use. Based upon Mr. Terpilak's findings we believe those rooms meet the guidelines for release for unrestricted access and request that those rooms be released for unrestricted access. The release of these rooms does not terminate licensed activities in the John Wesley Powell Building of the U.S. Geological Survey.

This report does not request the release of rooms in the Solid State Physics Building, that report and request will be separate and will be submitted shortly.

If you have any questions concerning this matter please call me at (703) 648-6189 or Dr. Charles W. Naeser at (703) 648-6964.

Sincerely,

Gregory A Wandless

Gregory Wandless
Radiation Safety Officer

cc: Curt Larson, MS 954
Gary Kramer, MS 246
Wayne Martin, MS 954
RSO Files

FINAL REPORT

RADIOLOGICAL SAFETY ASSESSMENT

FOR

UNITED STATES DEPARTMENT OF THE INTERIOR

U.S. GEOLOGICAL SURVEY

RESTON, VIRGINIA 22092

TASK ORDER NO. 1434-HQ-97-SA-00304

DECEMBER 31, 1996

PREPARED BY:

RAY-SAFE ASSOCIATES

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(AMERICAN BOARD OF HEALTH PHYSICS)

PREPARED FOR:

U.S. DEPARTMENT OF INTERIOR

U.S. GEOLOGICAL SURVEY

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I. EXECUTIVE SUMMARY AND ACTION ITEMS

A radiological safety assessment of the United States Department of Interior, U.S. Geological Survey (USGS) John Wesley Building was conducted on December 3, 1996 by Michael S. Terpilak, Certified Health Physicist.

The John Wesley Powell Building on the United States Geological Survey (USGS) National Center Campus located in Reston, Virginia is a multistory administrative and laboratory complex which houses approximately a few thousand employees. The complex consists of about 75% administrative offices and about 25% laboratory space. The two laboratories surveyed will be released for unrestricted use and will be converted into office space (Figure 1).

The radiological safety assessment included a radiation survey utilizing portable radiation detection survey instruments. In addition, a total 164 swipe samples were conducted at two laboratories located in the John Wesley Building on the campus of the USGS National Center located at Reston, Virginia.

The findings and results of all samples (instrument and swipe surveys) indicated that the 2 laboratories were free of fixed and removable contamination and qualify for release for UNRESTRICTED USE in accordance with the current U.S. Nuclear Regulatory Commission (NRC) guidelines, April 1993.

II. INTRODUCTION

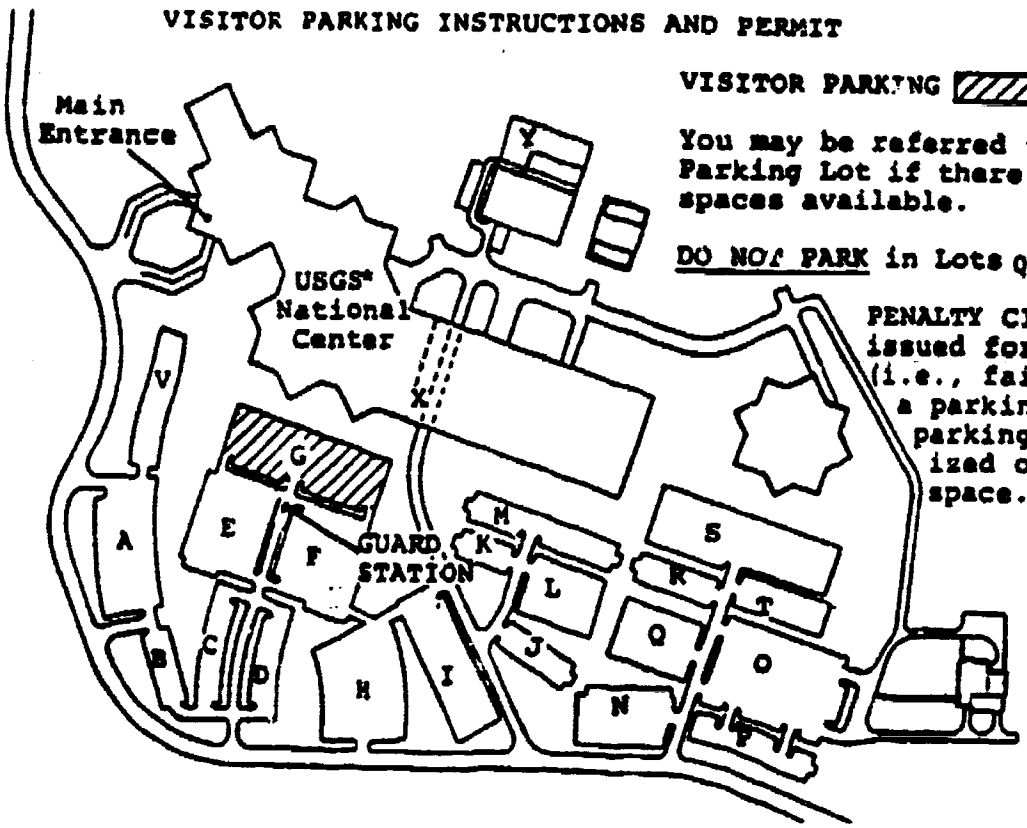
This radiological assessment was requested by Laura Schwarting, Contract Officers Technical Representative (COTR), United States Geological Survey (USGS). This radiological assessment was also conducted at the request of Mr. Greg Wandless, Radiation Safety Officer (RSO) of the USGS National Center. The date of request was November 13, 1996 and the Task Order number is USGS - 1434-HQ-97-SA-0034. The Radiological Safety Assessment was performed by Michael S. Terpilak, Certified Health Physicist, on December 3, 1996.

The objective of this report is to present the findings and results of the radiation survey and to indicate whether the laboratories surveyed qualify for release for UNRESTRICTED USE in accordance with current U.S. NRC guidelines April, 1993 and NUREG/CR-5849, "Manual for Conducting Radiological Surveys in Support of License Termination," June 1992.

FIGURE 1

JOHN WESLEY POWELL BUILDING *

VISITOR PARKING INSTRUCTIONS AND PERMIT



VISITOR PARKING

You may be referred to a General Parking Lot if there are no visitor spaces available.

DO NOT PARK in Lots Q, V, X, or Y.

PENALTY CITATIONS are issued for violations (i.e., failure to display a parking permit or parking in an unauthorized or undesignated space.

III. BACKGROUND

The U.S. Department of the Interior, U.S. Geological Survey (USGS) currently holds an NRC License (License Number - 45-15923-01 as amended through amendment 28) to operate the following facilities: (Appendix A)

- U.S. Geological Survey National Center
  12201 Sunrise Valley Drive
  Physics Building Laboratories
  Reston, Virginia 22092

- U.S. Geological Survey
  Stephenson Center
  Suite 129
  729 Gracern Road
  Columbia, South Carolina 29210

The current license expired on November 30, 1995. In accordance with 10 CFR 30.37, to ensure that the operating license did not expire, the USGS submitted the renewal application on November 27, 1995 and renewed July 29, 1996 with an expiration date of November 30, 2000.

The Nuclear Regulatory Commission (NRC) has established Technical and Financial Regulations for decommissioning Licensed Nuclear Facilities (53 CFR 24018, June 27, 1988). The regulations address decommissioning planning, needs, timing, funding methods, and environmental review requirements for Public and Private Facilities having licenses under 10 CFR parts 30, 40, 50, 70 and 72. The intent of the regulations is to ensure that the decommissioning of all Licensed facilities will be accomplished in a safe and timely manner and that licensees will provide adequate funds to cover all costs associated with decommissioning. The rule defines "Decommissioning" as the following: To safely remove nuclear facilities from service and reduce residual radioactivity to a level that permits release of the property for unrestricted use and termination of the license.

As noted in Part 30.35.f.4, government licensees may elect to submit a statement of intent containing an amount based on the possession limit of the license and indicating that funds will be obtained when decommissioning actually occurs. Part 30.35.c.2 states that if the licensee chooses to submit a certificate of financial assurance (statement of intent) at this time, a decommissioning funding plan and cost estimate shall be included in the licensee's next application for license renewal.

The regulations specify that a facility licensee either must set aside money for decommissioning activities or must provide a guarantee through a third party that funds will be

available. The funds set aside or guaranteed are determined by a decommissioning funding plan (DFP) which the licensee provides.

Specific requirements are stated in 10 CFR 30.35, "Financial Assurance and Recordkeeping for Decommissioning." 10 CFR 30.35.f.4 state that in the case of Federal, State, or local government of licensees, a statement of intent containing a cost estimate--or a value fixed by the regulation and determined by the type of facility--is acceptable. The regulations do not explicitly address the need for a DFP for all Federal, State, or Local government licensees. However, other NRC documents indicate the need for a DFP before the issuance of a new license or the renewal of an existing license.

Specific details of what the NRC considers an acceptable DFP are documented in NUREG-1336, "Interim Guidance on the Standard Format and Content of Financial Mechanisms Required for Decommissioning Under 10 CFR Parts 30, 40, and 70," and subsequent Regulatory Guide (Reg Guide) 3.66 published in June 1990. Also, NUREG-1337, "Interim Guidance on the Standard Review Plan for the Review of Financial Assurance Mechanisms for Decommissioning Under 10 CFR Parts 30, 40, and 70," serves as an outline and checklist for NRC staff reviewers. Reg Guide 3.65, "Standard Format and Content of Decommissioning Plan for Licensees under 10 CFR Parts 30, 40, and 70," outlines the overall decommissioning plan requirements showing the relationship of the DFP to the decommissioning process.

In addition, the following radiological surveys were conducted by the contractor for the Licensee prior to the release of Laboratory 3B429 and Storage Area BC214C for unrestricted use.

- Establishing Background Levels
- Scanning Survey Using Portable Radiation Instruments
- Removable Surface Contamination Measurements
- Laboratory Analysis and Measurement of Smear (Filter Paper) Samples

In accordance with section 4.2.2 Establishing Reference Grid Systems, since unaffected areas do not require gridding for the purpose of establishing measurement or sampling locations, the specific laboratory survey locations (instrument and smear) were marked, identified by white adhesive tabs and the laboratories were locked and secured to deny and prohibit areas pending the NRC review of this Radiological Safety Assessment report.

IV. EVALUATION METHODS

The radiological assessment was conducted by Michael S. Terpilak, Certified Health Physicist, in a manner similar to typical radiation surveys of licensees by the Nuclear Regulatory Commission (NRC).

These surveys were accomplished and performed in accordance with and as specified in the Nuclear Regulatory Commission (NRC), NUREG/CR-5849, "Manual for Conducting Radiological Surveys in Support of License Termination," Draft Report for Comment, June, 1992 as described and specified in section 4.2 and in addition to 10 CFRs Part 30 (Appendix B).

ESTABLISHING BACKGROUND LEVELS

Background was determined by conducting survey measurements and/or sampling at locations on the site, which are unaffected by-site operations, i.e., preferable locations for interior background determinations all within on-site buildings of similar constructions, or even John Wesley Powell Building location that has had no previous history of licensed operations, i.e., use of radioactive materials. Surveys conducted with portable radiation instruments as well as smear surveys were duplicated in laboratories and/or office space similar in dimensions and construction as the laboratories that are presently using licensed materials (i.e., laboratories using radioactive materials). Background surveys radiation instrument surveys were in the range of 0.020 mR/hr and 15-20 μR/hr and random smear samples were nondetectable, i.e., Minimum Detectable Activity (MDA).

Based on previous historical records, interviews with personnel, present licence inventory and possession license limits and conditions as well as current and present documented monthly Laboratory Contamination Surveys it was determined in accordance with Section 4.2 Designing the Survey, NUREG/CR-5849 that the one laboratory to be surveyed 3B429 was classified as an unaffected area and BC214C the storage area was also classified as an unaffected area, i.e., the laboratory and storage area are not expected to contain residual radioactivity, based on a knowledge of site history and previous survey information.

Since it can be readily shown from previous records, inventories, possession limits and many surveys that alpha radioactive material has never been used at this facility, it is recommended that only Beta-Gamma (β-γ) assessment be performed at this time.

SCANNING SURVEYS USING PORTABLE RADIATION INSTRUMENTS

These measurements typically consist of surface scanning (moving the detector at a consistent speed and distance near the surface) and measuring levels of direct radiation (surface activity and exposure rate) at representative points (Appendix D).

Before conducting any fixed measurements, surfaces are scanned to identify the presence of elevated direct radiation which might indicate residual gross activity or hot-spots. Scans are conducted for all radiations potentially present, based on the operational history. The scanning detector is kept as close as possible to the surface and moved cross the surface at a slow speed. Nominally, the distance between the detector and the surface is maintained at less than two centimeters. For particulate radiations (beta) which may have very limited ranges, the scan speed should not exceed 1 detector width per second; this speed should be reduced to as low as 1/3 detector width per second for those situations when relatively low count rates may be indicative of residual activity exceeding guideline values. For gamma radiation the scanning speed may be greater; the probe is typically moved in a serpentine pattern while advancing at a speed of about 0.5 m per second.

For optimum detection sensitivity, changes in the instrument response are monitored via the audible output (use of headphones is recommended), rather than by observing fluctuations in the analog meter reading. This use of an audible signal negates concern for the time constant related to the meter response. Locations of direct radiation, discernable above the ambient level (typically 2 to 3 times the ambient count rate), are marked on facility maps and identified fo: further measurements and/or sampling.

A complete radiological assessment included a radiation survey utilizing a Radiological Instrumentation as identified and specified in section 5 Radiological Instrumentation of NUREG/CR5489 (Appendix C).

- Beta-Gamma (β-γ) Field Survey Instrumentation

The following instruments were utilized in the survey.

- Ludlum Model #2 survey meter serial #88228 with a pancake type halogen quenched Geiger-Muller (G-M) Probe Model #44-9 Serial #010560 with a 1.7 ± 0.3 mg/cm^2 mica end window and active area of 15 cm^2.

- Ludlum Model #3 survey meter serial #22331 with an end window halogen quenched Geiger-Muller (G-M) Probe Model #44-7 Serial #PRO5338 with a 1.7 ± 0.3 mg/cm^2 mica end window and active area of 6 cm^2.

- Gamma Field Survey Instrumentation

The following instrument was utilized in the survey.

- Victoreen Model 450P Pressurized Ion Chamber Survey Meter with a high sensitivity Micro-R measurements of exposures and exposure rates. Serial #1618

The appropriate calibration certificates for the above instruments are enclosed (Appendix G).

The Beta-Gamma (β-γ) field survey instrumentation, i.e., the surface scanning (moving the detectors at a consistent speed and distance near the surface of each area approximately 1 cm) was conducted at 140 surface locations in Laboratory 3B429 and 24 locations in the storage area BC214C and reported in mR/hr (Appendix C).

The gamma (γ) field survey instruments were conducted at approximately 1 meter from the 140 surface locations in Laboratory 3B429 and 24 locations in the storage area BC214C and reported in mR/hr. In addition another reading was reported in μR/hr at 1 meter (Appendix C). The radiological surveys were conducted consistent and in compliance with 10 CFR 30 and NUREG/CR 5849. In addition a total of 164 wipe samples were conducted in the two laboratories, 140 wipes in Laboratory 3B429 and 24 wipes in storage area BC214C located in the John Wesley Powell Building, Main Campus, Reston, Virginia. **Laboratory 3B429 and storage area BC214C are classified as an UNAFFECTED AREAS, i.e., these areas were not expected to contain residual radioactivity, based on a knowledge of site history and previous survey information.**

V. RADIONUCLIDE USAGE IN LABORATORY 3B429 AND STORAGE AREA BC124C

The only radionuclides used in this laboratory space were as follows:

- Sodium-24
- Aluminum-28
- Silicon-31
- Phosphorous-32
- Potassium-42
- Calcium-49
- Manganese-52
- Manganese-54
- Iron-55
- Iron-59

Attached for your information is Table 1.0 that identifies the specific characteristics of these radionuclides.

Radiation History Room 3B429 John Wesley Powell Building, USGS, Reston, VA

Rooms 3B429, B429B and BC124C were used for handling and analyzing radioactive rock and mineral samples used in $^{40}Ar/^{39}Ar$ dating techniques. The samples were made radioactive by irradiation in the USGS TRIGA reactor in Denver. All of the samples were of sand-sized rock and mineral samples. Nuclides produced during irradiation included: ^{31}Si, ^{28}Al, $^{55,59}Fe$, $^{52,54}Mn$, ^{42}K, ^{49}Ca, ^{32}P, and ^{24}Na. Prior to irradiation all samples were packaged in aluminum capsules which were in turn placed into fused silica vials that were vacuum sealed

prior to irradiation. Upon return from the reactor the fused silica vials were opened in 3B429B. The individual aluminum capsules of samples were then stored in pyrex vials prior to analysis in 3B429. Pyrex vials of samples that were not to be immediately analyzed were stored in a concrete cave in BC214C.

Samples that were analyzed were loaded into a sidearm assembly in 3B429B. This sidearm assembly was then sealed under vacuum to a low blank furnace attached to the Mass spectrometer in 3B429 and the samples were melted and analyzed. The radioactive melts of samples that were collected in a Molybdenum (Mo) crucible liner that was removed on each reloading of the system. The radioactive (Mo) crucible liners and failed crucibles were temporarily stored in a plastic-lined radioactive waste can in room 3B429B. When that radioactive waste can filled, it was emptied into a second larger plastic-lined radioactive waste can in BC214C, within the concrete cave. When this radioactive waste can filled its contents were moved to a 55 Gal drum in the Hot Rock Shed.

Sidearm assemblies and other reusable hardware were cleaned with distilled water and 10% HCl in 3B429B and dried in an oven before being reused. All wash water and acid wash was evaporated to dryness in the right hood (as you enter the door). The evaporation containers were discarded in the radioactive waste as needed.

Radioactive material used and processed in this laboratory consisted of minerals irradiated at the USGS Triga Reactor Facility in Denver, Colorado. Since 1985 until the present, the laboratory only utilized sealed sources in Argon-Argon analysis and samples were never unsealed. Repeated instrument and swipe surveys over the last 11 years indicated no spills and no record of contamination.

Based on the above use of this laboratory it was designated as an unaffected area as defined in NUREG/CR-5849 and as such section 4.23 Selecting Measurement Sampling Locations for Structure Surveys states that scans of unaffected areas should cover a minimum of 10% of the floor and lower wall surface area. Since there was no reason to suspect residual activity exceeding 25% of the guideline value on these surfaces, a minimum of 30 measurement locations each, on vertical and horizontal surfaces where radioactive materials would likely accumulate (air exhaust rents and horizontal surfaces where dust would settle was selected). To assure a reasonable coverage of these surfaces, an average of at least 1 measurement location per $20m^2$ of surface area was selected. At least 30 randomly selected measurement locations or average measurement of 1 per 50 m^2 of building surface area, whichever is greater, for total and removable activity, should be performed for each survey unit.

RADIATION HISTORY – STORAGE AREA – BC214C

As mentioned previously when the irridated samples were not in use they were stored in a cave in the storage area located in Room BC214C.

TABLE 1.0
CHARACTERISTICS OF RADIONUCLIDES
USED IN LABORATORY 3L429
AND STORAGE AREA BC214C

Radionuclide	Radiation	Energy	Half-Life
• Sodium-24	Beta- Gamma	4.17 MeV Max. 1.390 MeV (100%) 1.369 MeV (100%) 2.754 MeV (100%)	15.0 hours
• Aluminum-28	Beta- Gamma	2.85 MeV Max. 1.780 MeV (100%)	2.30 minutes
• Silicon-31	Beta- Gamma	1.48 MeV Max. 1.26 MeV (0.07%)	2.62 hours
• Phosphorus-32	Beta-	1.710 MeV Max. .695 MeV (100%)	14.3 days
• Potassium-42	Beta- Gamma	3.52 MeV Max. 0.31 MeV (0.2%) 1.524 MeV (18%)	12.36 hours
• Calcium-49	Beta- Gamma	1.95 Max. 3.10 MeV (89%) 4.10 MeV (10%)	8.8 minutes
• Manganese-52	Beta+ (Positron) Gamma (Chromium x-rays)	0.575 Max. 0.511 MeV (60%) 0.747 MeV (90%) 0.935 MeV (95%) 1.434 MeV (100%)	5.60 days

Radionuclide	Radiation	Energy	Half-Life
• Manganese-54 (Electron Capture Decay)	Gramma (Chromium x-rays)	0.835 MeV (100%)	312 days
	Electron (-)	0.829 MeV	
• Iron-55 (Electron Capture Decay)	Manganese x-ray continuous Bremsstrahlung	0.23 MeV (0.004%)	2.70 years
• Iron-59	Beta		45 days
	B-1	130.8 keV Max. (1.27%)	
	B-2	273.4 keV Max. 80.8 keV Avg. (45.6%)	
	B-3	465.8 keV Max. 149.3 keV Avg. (52.8)	
	B-4	1.565 MeV Max. 635.8 ke V Avg. (0.18%)	
	Gamma		
	γ-1	190 keV (3%)	
	γ-2	1.10 MeV (56%)	
	γ-3	1.29 MeV (44%)	
	γ-4	143 keV (1%)	

In addition, all irradiated samples were in a solid state, no radioactive liquids or gases were involved, and all irradiated samples were sealed, no unsealed radioactive material or sources were used in these laboratories. The estimated activity of the radioactive materials never exceeded microcurie quantities.

VII. ANALYSIS AND COMPARISON OF RADIATION SURVEY RESULTS TO U.S. NUCLEAR REGULATORY COMMISSION (NRC) GUIDELINES

The USNRC guidance document entitled "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source or Special Nuclear Material," April 1993 states that depending on the specific radionuclides identified acceptable surface contamination levels shall be followed.

Therefore, based on three radionuclides identified in these two laboratories the category Beta-Gamma emitters was selected as the criteria for analysis and comparison with the radiation survey results (instrument and swipes).

The Beta-Gamma emitter nuclide category for acceptable surface contamination levels are as follows:

- Average - 5,000 dpm Bγ/100 cm^2
- Maximum - 15,000 dpm Bγ/100 cm^2
- Removable - 1,000 dpm Bγ/100 cm^2

In addition, the average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/hr at 1 cm and 1.0 mrad/hr at 1 cm respectively, measured through not more than 7 milligrams per square centimeter of total absorber.

Swipe surveys at each laboratory indicted that samples were nondetectable, i.e., Minimum Detectable Activity (MDA), the results were significantly below the average, maximum and removable acceptable surface contamination levels.

Subsequently the radiation instrument surveys of each laboratory indicated levels in the range of 0.020 mR/hr and 15-20 μR/hr which indicate natural background radiation and were significantly below the average and maximum radiation levels associated with surface contamination resulting from then beta-gamma emitters.

VIII. BRIEF SUMMARY OF STANDARDS/CRITERIA

The U.S. Nuclear Regulatory Commission (NRC), Division of Fuel Cycle, Medical, Academic and Commercial Use Safety, Washington, D.C., has published a technical document entitled "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source of Special Nuclear Material," dated April 1993. The document delineates specific instructions concerning instrument and swipe surveys and Table 1, specifically states "Acceptable Surface Contamination Levels" that shall be used for decontamination of facilities and equipment prior to release for unrestricted use (Appendix F).

IX. RESULTS AND CONCLUSIONS

Based on the results of the radiological safety assessment, i.e., the radiation instrument and swipe survey, the following laboratories located in the John Wesley Powell Building United States Geological Survey (USGS) National Center can be released for unrestricted use:

- Laboratory Room - 3B429
- Storage Room - BC2i4C

IX. REFERENCES

Title 10 Code of Federal Regulations, Part 20, Standards for Protection Against Radiation, January 1993. Federal Register Vol. 56, No. 98, Tuesday, May 2, 1991, pp. 23360-23474 (Includes revisions to Parts 2, 19, 20, 30, 31, 32, 34, 35, 39, 40, 50. 61, and 70).

Title 10 Code of Federal Regulations, Part 30, Rules of General Applicability to Domestic Licensing of Byproduct Material, January 1, 1993.

Title 10 Code of Federal Regulations, Part 33, Specific Domestic Licenses of Broad Scope for Byproduct Materials January 1, 1993.

Title 10 Code of Federal Regulations, Part 71, Packaging and Transportation of Radioactive Materials, January 1993.

Title 49 Code of Federal Regulation, Parts 100 to 177, Transportation, October 1, 1992.

Regulatory Guide 3.66, Standard Format and Content of Financial Assurance Mechanisms Required for Decommissioning under 10 CFR Parts 30, 40, 70, and 72, June 1990.

Regulatory Guide 7.10, Establishing Quality Assurance Programs for Packaging Used in the Transport of Radioactive Material, Rev. 1, June 1986.

Regulatory Guide 8.7, Instructions for Recording and Reporting Occupational Radiation Exposure Data, Rev. 1, June 1992.

Regulatory Guide 8.9, Acceptable Concepts, Models, Equations and Assumptions for a Bioassay Program, Rev. 1, July 1993.

Regulatory Guide, 8.10, Operating Philosophy for Maintaining Occupational Radiation Exposure as Low as Reasonably Achievable, Rev. 1, May 1977.

Regulatory Guide 8.18, Information Relevant to Ensuring that Occupational Radiation Exposures at Medical Institutions Will Be As Low As Reasonably Achievable, Rev. 1, October 1982.

Regulatory Guide 8.23, Radiation Safety Surveys at Medical Institutions, Rev. 1. January 1981.

Regulatory Guide 8.25, Air Sampling in the Workplace, 1992.

Regulatory Guide 8.33, Quality Management Program, October 1991.

Regulatory Guide 8.34, Monitoring Criteria and Methods to Calculate Occupational Radiation Doses, July 1992.

Regulatory Guide 8.36, Radiation Dose to the Embryo/Fetus, July 1992.

Regulatory Guide 8.37, ALARA Levels for Effluents from Materials Facilities, July 1993.

APPENDIX A

UNITED STATES GEOLOGICAL SURVEY

RADIOACTIVE MATERIALS LICENSE

NUMBER--45-15923-01

JULY 29, 1996