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December 19, 2000
Docket No. 50-62, License R-66

Mr. Alexander Adams, Jr., Senior Project Manager
Events Assessment, Generic Communications and Non-Power Reactors Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission M.S. 0-11-D-19
Rockville, MD 20852-2738

Subject: UNIVERSITY OF VIRGINIA – REQUEST FOR ADDITIONAL INFORMATION RE: DECOMMISSIONING AMENDMENT REQUEST (TAC NO. MA8186).

Dear Mr. Adams,

Please find enclosed the University's response package to the NRC's Request for Additional Information of November 2, 2000. In making our response we were assisted by GTS-Duratek, the contractor who performed the UVAR characterization survey. Please note that no proprietary data has been included.

The response package was reviewed and approved by the University of Virginia's Nuclear Reactor Decommissioning Committee on December 19, 2000, and in accordance with 10CFR 50.30(b) the signed original and attachments are submitted by me under oath.

We appreciate the past NRC evaluation of our amendment request under an expedited schedule. Should you have questions regarding this document, please call me at (804) 982-5440.

Sincerely,

Robert U. Mulder

Robert U. Mulder, Director
UVa. Reactor Facility & Assoc. Prof. of Nuclear Eng.

City/County of: Albemarle
Commonwealth of Virginia

I hereby certify that the attached document is a true and exact copy of a letter (type of document) **presented before me this** 19th **day of** December, ~~19~~ 2000 **by** Robert Mulder (name of person seeking acknowledgement).

Vickie S. Thomas
Notary Public

My commission expires 2/28, ~~19~~ 2002.

Enc.: UVA's Response to NRC's RAI of November 2, 2000
New UVAR TS 4.11, revised UVAR TS 6.3.1,
New UVAR TS 5.2, and revised TS 6.1.2
REFS-CALC-UVAR-001, Revision 0: Dose Assessment for UVAR Decommissioning Plan
REFS-CALC-UVAR-002, Revision 0: Radiological Accident Analysis for UVAR Decommissioning Plan

cc: Mr. Craig Basset, NRC Region II, Atlanta, GA
Document Control Desk, NRC, Washington

A020

UVA RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION UNIVERSITY OF VIRGINIA RESEARCH REACTOR DOCKET NO.50-62

1. **Section 1.2, "Background," Page 1-10. Is the pond included in the category of site environs? If not, please explain.**

   Yes, the pond is included. The areas listed in this section of the decommissioning plan included only those portions of the facility where remediation would be required based upon the characterization surveys.

   The entire facility within the fenced area is included in the site environs and all areas of the facility will be surveyed to release the facility. The total land area of the facility is about 102,000 square feet. The Reactor Facility footprint is 6,000 square feet, for a total building area of 15,650 square feet on three floor levels. There are 96,000 square feet of outdoor area, including a 16,000 square feet of pond area.

2. **Section 1.2, "Background," Page 1-10. A number of rooms are not included in the scope of the Decommissioning Plan (DP). What is the basis for not including these rooms in the scope?**

   The areas listed in this section of the UVAR decommissioning plan included only those rooms in the building where remediation could be required based upon the characterization study surveys. Some of the rooms excluded are classrooms and faculty/student offices in which radioactive materials have never been used. Also excluded from the scope are areas included under the decommissioning of the CAVALIER reactor, which are covered under a separate decommissioning plan.

   It is noted that the entire building, including all offices, classrooms and laboratories, will be surveyed as a condition for the release of the facility.

3. **Section 1.2.1.5, Reactor Decommissioning Overview," Page 1-12. It is stated that onsite interim storage of the waste is an option if there is no licensed disposal facility available. Please discuss all aspects of that storage (i.e. Where will it be? How will it be monitored and controlled?).**

   We propose to change the wording of this item as originally covered in the UVAR Decommissioning Plan. Since licensed disposal facilities are expected to be available to receive UVAR Facility waste, we request that all references to onsite storage of waste be deleted from the plan. There will be no onsite waste storage. In the event that disposal options become unavailable, then the issues of decommissioning project continuation, including potential storage of waste onsite, will be revisited with the NRC.

4. **Section 1-2.4.1, "Quality Assurance Responsibilities," Page 1-15. This section contains a statement that the contractor has overall responsibility for the QA Plan implementation and is responsible for verifying the effective Implementation of the plan. Please confirm that, as licensee, UVA has ultimate responsibility for the QA Plan implementation and is responsible for verifying the effective implementation of the plan.**

   The University of Virginia, as the reactor license holder, has the ultimate responsibility for the implementation of the QA Plan for the UVAR, both by UVA and its decommissioning subcontractor(s).

UVA shall verify, through periodic Reactor Decommissioning Committee audits, that the QA Plan is being effectively implemented.

5. **Section 1.2.4.2, "Instrument Calibration," Page 1-15. This section discusses calibration requirements for field instruments, associated detectors, and laboratory instruments. Should these surveillances be included in the Technical Specifications (TSs)? If not, please justify.**

   Yes, decommissioning instrument surveillance requirements should be included in UVAR TS. Please see in attachment our proposed new UVAR TS 4.11 Surveillance of Decommissioning Instrumentation.

   Decommissioning instrument response testing, maintenance and associated record keeping is best covered in future UVAR SOPs. To assure that this is done, we propose an additional item (8) be added to current UVAR TS 6.3.1 Items Covered by SOPs (please see attachments).

6. **Section 1.2.4.2, -Instrument Calibration." Page 1-15. Is the "LLD < 50% of the release limits" a QA requirement?**

   All final status survey measurement results will be reviewed to ensure the measurement LLD (or MDA) is less than 50% of DCGL. This performance requirement will be listed as a data quality objective (DQO) in the final status survey plan developed after the decommissioning contract is awarded for the project and will be evaluated during the assessment of the final survey data once the survey has been completed.

7. **Section 1.2.4.3, "Sample Analysis," Page 1-17. Please confirm that UVA has ultimate responsibility for ensuring that the sample analysis specifications and laboratory capabilities will meet requirements for data quality.**

   Yes, UVA has the ultimate responsibility for ensuring that decommissioning sample analysis specifications and laboratory capabilities meet data quality requirements.

8. **Section 2.2.1, "Facility Operating History," Page 2-2..**

   (a) Identify all locations, inside and outside the facility, where radiological spills, disposals, operational activities, or other radiological accidents/incidents occurred and could have resulted in contamination of structures, equipment, lay-down areas, or soils (sub floor and outside area). (b) Please give a brief description of the radiological spills, disposals, operational activities, or other radiological accidents/incidents that were considered.

   We have reviewed the "decommissioning file" which was kept current during the operational life of the UVAR, as well as all the memoranda sent to the Reactor Safety Committee, to identify events which have, or could have, resulted in contamination of areas within and without the Reactor Facility.

UVAR Radiological Spills, Disposals, Operational and Radiological Accidents/Incidents

Date of Report	*Date of Event*	*Description (a) & (b)*
Aug. 16, 1962	Aug. 13, 1962	Suspected rupture of in-core plasma thermocouple while operating. Analysis showed that some activated fiberglass in the experiment probably vaporized and became airborne in the reactor confinement. This event was considered in that the reactor confinement room floor and walls were sampled during the characterization study.
Jan. 27, 1969	Oct. 18, 1963	(Note: Belated report) A break occurred in the piping in the demineralizer room. Pool level dropped about one foot corresponding to a loss of about 2,500 gallons. Automatic isolation valves were installed. This event was considered in that demineralizer room floors and walls were sampled during the characterization study.
Several	1968	Leak rate from pool increased to 600-800 gpd. Reactor operation was suspended on Aug. 6, 1968. Diver performed an underwater inspection and found several cracks and epoxy coating bonding failure in a few locations. Diver performed some patching in the buttress area. Leak rate was not reduced. Following removal of radioactive materials, the entire pool was drained. Extensive concrete chipping was done on the buttresses and Gunite used to patch leaking areas. Entire pool was sandblasted and walls coated with three coats of epoxy. Operations resumed in January 1969, with the leak rate reduced to about 15 gpd. This event was mostly not considered in that certain types of sampling could not be done in the filled UVAR pool during the characterization study.
Nov. 8, 1971	Nov. 1, 1971	Pool water was accidentally siphoned from pool through the skimmer system and onto the reactor room floor. Water got into in-floor cable trays. This event was considered in that the reactor confinement room floor and walls were sampled during the characterization study.
Logbook	Apr. 12, 1973	A small leak was found at S.E. Access Facility in the reactor bay area, w/ a rate of one gal/week. The leak self-sealed after about one month. This event was considered in that the reactor bay area floor and walls were sampled during the characterization study.
May 30, 1973	May 17, 1973	As part of the Oxygen-19 experiment, CO_2 was released into the reactor pool causing the lowering of the pH to around 6.0. Ammonium hydroxide had to be added to the pool to raise the pH. This event was a transitory event w/o decommissioning impact possible.
Logbook	Oct. 5, 1973	The leak of April 12 reappeared at the S.E. Access Facility. Some epoxy was applied externally at the access facility and the leak stopped. This event was considered, as were other similar events involving leaked pool water in the reactor bay area.

Date of Report	*Date of Event*	*Description (a) & (b)*
Jan. 15, 1974	Late 1973	Pool makeup rate has again increased, sometimes up to >200 gpd. Reactor was shutdown Jan. 8, 1974 to start repairs. South end of pool was drained on 1/12. Repairs were started on 1/16 in the face area on the lower level (not in the pool) using an injection grouting technique. Pool was refilled on 1/31 and the leak reappeared. Again, the south end of the pool was drained. The emergency spray tanks were removed to permit inspection behind them. A crack was found behind the southwest tank and it was repaired with epoxy grouting. When pool was refilled the leak rate was reduced to about 80gpd. This event was not considered in that the reactor pool floor and walls could not be sampled during the characterization study, as the pool was still filled with water and radioactive hardware.
Nov. 8, 1976	Nov. 22, 1976	Seven geologic samples irradiated in the flux basket spilled out while in the reactor. Five were recovered, two were never found. This event was not considered, because the pool was still filled with water and hardware at the time of the characterization study.
Several	1977	June 14: South end of pool drained, patching in-pool around S.E. Access Facility. June 26: pool refilled. This event was considered. Sep 12: South end of pool drained, work on inside and outside on pool at the S.E. Access Facility. Sep. 21: pumped in liquid epoxy into the concrete around the S.E. access facility. Oct. 3: pool refilled, leak rate still about 300 gpd. Oct. 11: diver used. Oct. 13: diver found large crack in pool floor around the primary system piping. Oct: 19: south end of pool drained. Oct. 26: work completed. Nov. 3: pool refilled and the leak rate reduced to <10 gpd. This event will be addressed during future decommissioning work after the UVAR pool has been emptied and inspected.
	Apr. 6, 1979	A small fission product leak from a fuel element element was discovered. The fuel element was taken out of service and shipped off-site as spent fuel. This event was not considered explicitly in the decommissioning plan. The minute amounts of products leaked were either absorbed or adsorbed in or onto pools walls and demineralization filter resins. Note: The UVAR pool internals and coolant flow piping will be addressed in future decommissioning work.
	Nov. 24, 1980	Reactor confinement and radiochemistry lab were found to be contaminated. Highest levels found outside the "known-to-be-contaminated" controlled area on the east side of the pool., were in the control room. Identified isotopes were Ag110m and Co-60. Contamination was probably tracked in from the controlled area. This event was considered in that reactor confinement and radiochemistry area floor and walls were sampled during the characterization study.

Date of Report	*Date of Event*	*Description (a) & (b)*
	May, 1981	Contents of underground liquid waste tank #2 were pumped onto the dirt floor underneath the tanks due to a break in the recirculation system piping. The estimated volume released was 4,000 gallons. The isotopes identified were Ag-110m, Ag-108m, Cs-134, Cs137, Co-60, Co-58, Mn-54, Zn-65, Cr-51 and Eu-152. Estimated activity released to waste tank enclosure was 26 microcuries. This event was considered in that the soil underneath the tanks was sampled during the characterization study.
Jul. 23, 1984	Jul. 18, 1984	Line in demineralizer room broke and water leaked to sump and heat exchanger room. An estimate of the amount leaked was not recorded. This event was considered in that the demineralizer room floor and walls were sampled during the characterization study.
Oct. 3, 1984	Sep. 21, 1984	Pipe break in demineralizer room. About 3,000 gallons water were diverted to the sump and the heat exchanger room. Estimated that about half the water went to the waste tanks and half to the ground under the heat exchanger room though a gap in the floor/wall interface. This event was considered in that the demineralizer room floor and walls were sampled during the characterization study.
Aug. 29, 1985	Aug. 23, 1985	Water was inadvertently pumped from one underground waste tank to the other, causing the second tank to overflow. About 4,500 gallons were spilled to the tank enclosure, corresponding to an estimated 200 microcuries. There was a slight increase in pond activity and the soil under the tanks were contaminated. Isotopes identified were: Co-60, Ag-110m, Cs-137, Eu-152, Mn-54 and Zn-65. This event was considered in that the soil underneath the tanks was sampled during the characterization study.
Jul. 18, 1988	Jul. 14., 1988	Water leaked from underground waste tank #2 to ground through one-inch slit in the bottom of tank. Approximately 2,900 gallons were released with an activity of 94 microcuries. Isotopes identified were: Co-60, Ag-110m, Cs-137, Eu-152, Mn-54 and Zn-65. This event was considered in that the soil underneath the tanks was sampled during the characterization study.
Aug. 23, 1990	Aug. 23, 1990	Overflow of approximately 500 gallons of pool water to the pond while filling the pool. No measurable long lived isotopes were involved. This event was considered in that pond sediment was sampled during the characterization study.

(Cont.) UVAR Radiological Spills, Disposals, Operational and Radiological Accidents/Incidents

Date of Report	*Date of Event*	*Description (a) & (b)*
Dec. 16, 1990	Nov. 14, 1990	Water line from waste tanks to spillway is found to be broken after about 50 gallons of tank water are released onto the hillside on the side of the pond. Estimated content was about 1.2 microcuries. Earth from around spill site was removed, placed in barrels and shipped off-site as RAM waste. This event was considered in that the soil in the area of the spill were sampled during the characterization study.
Oct. 7, 1992	Sep. 22, 1992	Two spills of Ir-192 ceramic beads in pool. Most are vacuumed up. Some remain on bottom of pool and in heat exchanger. (74 day Ir-192 half-life!). This event was not considered in that the reactor pool and coolant pipes and heat exchanger were not sampled during the characterization study. Once the pool is emptied during future decommissioning work, these components will be inspected. Most of the Ir-192 has already decayed away.
Nov. 1, 1993	prev. 3 months	30-60 gpd pool leak rate increases to over 100 gpd. Treated by pumping (under high pressure) a water-activated expanding foam into the access facility walls from the lower level. Leak rate drops to about 50 gpd but returns to 100 gpd a short while later. This event was considered in that the reactor bay area floor and walls were sampled during the characterization study.
Mar. 4, 1994	previous month	Pool leak rate is at 10-40 gpd. Upper 2 feet of pool walls inside the pool are painted. This event was not considered in that the reactor pool is still filled with water and radioactive hardware, and thus was not fully sampled during the characterization study.
Aug. 26, 1995		Leaking tubes are discovered during annual cleaning of the heat exchanger. Water was found leaking from primary to secondary when reactor pump was on and the leaking in other when pump was off. Estimated rate was less than 1 gph either way. Leaking tubes were plugged. This event was not considered in that the reactor pool is still filled with water and radioactive hardware, and thus the interior of the heat exchanger was not sampled during the characterization study.
Sep. 3, 1997	Aug. 1997	Six more heat exchanger tubes are plugged, for a total of 15. This event was not considered in that the reactor pool is still filled with water and radioactive hardware, and thus the interior of the heat exchanger was not sampled during the characterization study.
Oct. 28, 1997		Two monitoring wells were drilled on south side of building. Water samples show no significantly elevated contamination of groundwater. The wells continue to be monitored, and the results were implicitly considered.

9. **Section 2.3.1.1.3, "Decontamination of the Facility," Page 2-5. It is stated that neutron activation is not expected in the surrounding soil volumes. Please list the references and experiences that were used as the bases of that determination.**

The basis for the assumption is the high attenuation of neutrons that can be expected to occur in the several feet or more of water and concrete separating the reactor core from the soil. However, we ask that this statement be deleted from the Decommissioning Plan, because it serves no practical purpose. The determination of actual residual activity will be based upon direct measurements and sampling.

10. **Section 2.3.1.1.3.2, "Reactor and Pool," Page 2-8. The reactor pool had a history of leaks. Please discuss how leakage paths will be handled during decommissioning activities.**

The pool walls and floors will be surveyed as they are remediated. Wherever activity is found in excess of limits it will be remediated. This includes the remediation of contaminated leakage paths. In addition, core bore samples will be taken through the pool walls and floor in several locations to allow sampling of materials that potentially could be contaminated by pool leakage. In the building plans for the old part of the reactor facility, a drain line around the pool perimeter is indicated. This drain line is shown to discharge into the storm drain system. The drain line and storm drains, if actually built as referenced in the drawings, will be monitored during future decommissioning work using pipe monitors.

11. **Section 2.3.1.1.3.5, "Outdoor Areas, Drains & Sewers," Page 2-10. This statement does not appear to be consistent with the information presented in Appendix A, Summary of Characterization Results, pages A-3 through A-6, which indicate areas of elevated activity. Please review, explain, and amend as necessary.**

The information on pages A-3 through A-6 is correct. There will be remediation in the "Outdoor Areas, Drains & Sewers". Section 2.3.1.1.3.5 should read as follows:

- If they exceed DCGLs, the contaminated pond sediments near the vertical reactor building discharge pipe will be removed and packaged for processing or direct disposal as radioactive waste.
- If they exceed DCGLs, the contaminated surface soil on the pond bank between the underground tanks and the pond will be removed and packaged for processing or direct disposal as radioactive waste.
- If they exceed DCGLs, the sediments with elevated activity in the storm drain will be removed and packaged for processing or direct disposal as radioactive waste. The drains will be surveyed using pipe probes to determine if there is any additional sediment with elevated activity that requires removal.
- Residual radioactivity will be reduced to levels that are ALARA.

12. **Section 2.3.1.2, "Dismantling Sequence," Page 2-11. The statement that the control rods are expected to have the highest levels of induced radioactivity does not appear to be consistent with section 2.2.2.2 on page 2-2. Please review, explain, and amend as necessary.**

The data shown on page 2-2, does not include the survey notes which provide additional details on the surveys performed. For example, the note for the EPRI experiment stand indicates that the experiment stand was sitting next to control rods that may have biased the data. Another EPRI

experiment stand had a contact dose rate of 1.9 R/hr. The most recent survey data indicates that dose rates from these control rods was in excess of 200 R/hr at contact. A revised listing of items on page 2-2 with dose rates in excess of 5 R/hr is included below.

- Control Rods (all next to each other) reading 117 to >200 R/hr at about six inches.
- Three Hot Thimbles reading about 21 to 27 R/hr.
- Two EPRI experiment stands (Al) reading about 19 R/hr.
- Mineral Irradiation Facility (MIF) shield reading about 13 R/hr.
- An old control rod (stored in the pool) reading about 10 R/hr.
- Three tangential beamport targets reading 6 to 8 R/hr.
- Hydraulic Rabbit (Al) about 25' long reading about 6 R/hr.

(Note: Near at-contact dose measurements in the reactor pool were taken underwater.)

13. **Section 2.3.1.2, "Dismantling Sequence," Pages 2-11 to 2-13. This section discusses setting up a confinement barrier to surround the reactor pool with a ventilation system to ensure a negative pressure, maintaining the Reactor Room at a negative pressure when necessary, and the continuous monitoring of radiation levels. Please propose TSs or discuss why these requirements need not be TSs.**

We are proposing a new UVAR TS 5.2. Temporary Pool Confinement (please see documents in attachment.) This TS addresses the temporary structure to be erected in the Reactor Room while decommissioning work involving the UVAR pool is in progress.

Note: The Reactor Room was constructed to provide confinement, not containment. The Reactor Room is maintained at a very small negative pressure with respect to the rest of the Reactor Facility when its ventilation system is on.

14. **Section 2.3.1.2, "Dismantling Sequence," Page 2-13. The plan states that the pool will be backfilled and capped with a concrete slab and the hole from excavating the waste tanks will be backfilled. Please confirm that these activities will occur after these areas are released for unrestricted use by NRC.**

These areas will not be backfilled until the NRC releases them for unrestricted use. (Please, also see our response to questions 15 and 16).

15. **Section 2.3.1.2, "Dismantling Sequence," Page. 2-13. Please discuss how the exposed condition of the pool pit will be maintained for the 3 to 4 months (approximate time from Figure 2-3) between remediation and the final release survey.**

The pool structural integrity will be evaluated prior to work in the pool and shoring will be installed. A professional engineer will approve the support system design. Only authorized personnel will be allowed entry into the pool area.

It is proposed that the NRC, when given advance notice of two to three weeks, could perform verification surveys for the pool area prior to the normal NRC verification surveys.

An alternative approach, also proposed, is for UVA to provide split samples to the NRC from its area surveys. This approach was used successfully during the Portsmouth, Virginia, INS facility decommissioning, which was a NRC license termination D&D project. During this project, a tank was

excavated and exhumed to a depth of 10 feet. Prior to performing one of the deeper excavations, INS project staff contacted the NRC to inquire about the need to backfill for safety reasons. The NRC said that they did not need to be present during the excavation as long as sampling was occurring. In addition to required protocol sampling, quality control split samples were taken to allow for NRC analysis.

The NRC confirmatory survey personnel took samples of the grounds and trenches after the decommissioning was complete. However, they had no problems with the deeper excavation that was no longer accessible for sampling. The NRC terminated this license without any concerns.

16. Section 2.3.1.2, "Dismantling Sequence," Page 2-14. Please discuss how the exposed condition of the buried tank area will be maintained after remediation until the final release survey. Please discuss the remediation of the Pond. When is it scheduled?

In accordance with 29 CFR 1926.652, protective systems will be provided from the commencement of soil excavation of the buried tank area through completion of confirmatory surveys. A competent person and operators knowledgeable in excavation methodology and regulations will perform the excavation activities.

Should support systems be required to facilitate the excavation activities, a professional engineer will approve the support system design. During excavation activities and through completion of confirmatory surveys, a construction fence will be installed to isolate the excavation area. Only authorized personnel will be allowed entry into the excavation areas.

It is proposed that the NRC, with advance notice of two to three weeks, could perform verification surveys for the excavation prior to the normal NRC verification surveys.

An alternative approach, also proposed, is for UVA to provide split samples to the NRC from its surveys of the area. This approach was successfully used during the Portsmouth, Virginia, INS facility decommissioning, which was a NRC license termination D&D project. During this project, a tank was excavated and exhumed to a depth of 10 feet. Prior to performing one of the deeper excavations, INS project staff contacted the NRC to inquire about the need to backfill for safety reasons. The NRC said that they did not need to be present during the excavation as long as sampling was occurring. In addition to required protocol sampling, quality control split samples were taken to allow for NRC analysis.

The NRC confirmatory survey personnel took samples of the grounds and trenches after the decommissioning was complete. However, they had no problems with the deeper excavation that was no longer accessible for sampling. The NRC terminated this license without any concerns.

17. Section 2.4, "Decommissioning Organization and Responsibilities," Page 2-15. Please discuss the responsibilities and authority of the Radiation Safety Officer during decommissioning.

The Radiation Safety Officer for the University of Virginia will continue to function as the RSO for the Reactor Facility during D&D. The scope of his oversight will include all D&D operations that involve work with systems or materials that have a radiological component. The RSO is responsible for ensuring that:

a. Radiological controls are in place prior to and during any work involving radiation

b. Applicable license conditions are satisfied

c. Applicable state and federal regulations are met.

The Radiation Safety Officer has the authority to:

a. Implement any actions necessary to ensure that radiological controls are implemented and followed

b. Stop or modify radiological work immediately and then make changes to RWPs within 24 hours.

18. **Section 2.4.1, "Contractor Assistance," Page 2-15. If a contractor has been selected, please show the organization of the contractor(s). Please list the key positions in the organization and specify the responsibilities and authority of the person in that position. Please discuss the qualifications and experience of the contractor(s). If a contractor(s) has (have) not been selected, please specify the minimum qualifications and experience that a contractor must meet to be acceptable.**

A contractor has not been selected yet. However, the selection process is underway (UVA Request for Proposal VC0042700). UVA will select a contractor through state- and university-established procurement procedures and standards requiring a rigorous source evaluation and review process. The review and evaluation specifications define scope and method of selection and criteria for contractor qualifications, experience, and reputation. The contractor qualifications and experience required, as indicated in the above RFP, include the following:

Demonstration of experience in the performance of the following tasks:

- Integration of decommissioning, dismantlement and demolition plans.
- Waste management and other methods used to minimize final waste disposal costs.
- Decontamination and remediation of facilities and equipment.
- Use of survey equipment and techniques suitable for compliance with current NRC or MARSSIM survey criteria.
- Use of inventory and tracking mechanisms to assure accurate waste tracking.
- Provision of data collection packages that can capture data used in job estimates, work tasks and other data that will assist in the planning and execution of the decommissioning.
- Development and execution of radiological and industrial safety programs that will be used during the D & D.
- Selection, design and/or procurement of appropriate containers and packaging for radioactive and hazardous waste, and transportation to approved treatment and disposal facilities.
- Performing license termination surveys on a project of similar size and scope.
- Package, manifest, transport, process and dispose of radioactive waste.
- Instrumentation and procedures to perform embedded pipe surveys.

The decommissioning contractor selected must have a QA program that meets the requirements under 10 CFR 71 "Packaging and Transportation of Radioactive Material", Subpart H "Quality Assurance". In addition the contractors QA program must meet the applicable criteria from 10 CFR 50, Appendix B; the American Society of Mechanical Engineers (ASME) NQA-1. One of the applicable criteria that must be included is a QA Approved Suppliers List.

The contractor should be prepared to provide qualified personnel, including but not limited to the following:

- Project Manager
- Certified Health Physicist that meet ANSI 3.1 qualifications with MARSSIM survey experience
- Waste management
- Industrial hygienist
- Civil and mechanical engineer
- Quality assurance engineer
- Construction supervisor who has completed OSHA 40 hour compliance training
- Cost estimating and control specialist
- Planning and scheduling specialist
- Database administrator
- Decontamination and waste technicians
- Radiological safety engineer, foreman and technicians

19. Section 2.4.1, "Contractor Assistance," Page 2-15. If the decommissioning contractor is not GTS Duratek and Merrimac, confirm that the Xtreme PM^{SM} System will be available for use by UVA and the selected decommissioning contractor. If not, what process will be used to perform the functions of the System?

Official decommissioning project records will be hardcopy documents maintained in secure cabinets. The contractor will maintain working documents, and UVA will maintain completed or final project records and documents. These decommissioning files will be available at the Reactor Facility for inspection by the NRC. However, the use of a project management tool such as Xtreme PM^{SM} should be considered optional.

In Section 2.4.1. of the decommissioning plan we mention that Xtreme PM^{SM} ***can be*** used as a tool to allow centralized computer access to information from multiple locations. Thus, Xtreme PM^{SM} , or some other project management system provided by the chosen contractor, could be used to *support and supplement* the official project records which will serve as reference for the final license termination report for submission to the NRC.

20. Section 2.4.1, "Contractor Assistance," Page 2-15. Please discuss how it is assured that the Xtreme PM^{SM} System is used effectively and accurately during the decommissioning. What position has the responsibility for its effective and accurate use?

The Reactor Director will be responsible for the effective and accurate use of the Xtreme PM^{SM} (or any other) management system chosen by the licensee and contractor. He will assure, through regular personal inspection of the generated databases, that decommissioning data is accurately logged, and correspoding hard-copies are filed in a timely manner. (Also, please see our response to Questions 19 and 24.)

21. **Section 2.4.3, "Reactor Supervisor," Page 2-19. The minimum qualifications for the reactor supervisor position differ from those in the TSs. Please address.**

Please amend the qualifications for the Reactor Supervisor given in the Decommissioning Plan (at the end of section 2.4.3, page 2.18) to conform to the qualifications given in UVAR TS 6.1.2.:

"The minimum qualifications for the Reactor Supervisor are:

* A bachelor's degree in science or engineering and have at least 2 years of experience in Reactor Operation at the UVAR Facility, or at least 6 years of experience in Reactor Operations. Equivalent education or experience may be substituted for a degree."

22. **Section 2.4.4, "Radiation Safety Officer," Page 2-19. Should the minimum qualifications and duties of the RSO be described in the TS? If not, please justify.**

The minimum qualifications and duties of the RSO are described in a proposed additional paragraph to current UVAR TS 6.1.2. Responsibility (please see attachments.)

23. **Section 2.5, "Training Program," Page 2-19. Please specify the qualifications of the training instructors.**

The training instructors will comply with the requirements of ANSI/ANS-3.1, American National Standard for Selection, Qualification, and Training of Personnel for Nuclear Power Plants.

24. **Section 2.5, "Training Program," Page 2-19. Please discuss details of the training provided for the users and reviewers of the Xtreme PM^{SM} System.**

In our response to Question 19 we acknowledge that Xtreme PM^{SM}, or some other computer software provided by the contractor, may be used as a decommissioning record management system.

Formal and hands-on training will be provided by the contractor for users and reviewers of such a system. Training will include distribution of operating manuals, demonstration of record filing and access, and maintenance of system security. All authorized users shall be passworded. Most users will only be allowed "read only" access and will not be able to enter, modify nor delete records. The contractor-provided system will be maintained by the contractor. This means information will be updated primarily by the contractor. However, the Reactor Director will verify on a regular basis that the system and hardcopy records are being kept as required.

25. **Section 2.5.2, "Radiation Worker Training," Page 2-20. Please discuss the Scope of ALARA program training.**

The management of the University of Virginia Office of Environmental Health and Safety, and of the Nuclear Research Reactor Facility, is committed to a program for keeping individual and collective doses as low as reasonably achievable. This commitment is stated in the University of Virginia ALARA Program, and it is also described in Section 3.0 of the Decommissioning Plan.

An administrative organization for the promotion of radiation safety, which includes the Radiation Safety Committee and the Radiation Safety Officer, is in place at the University. Instruction in the ALARA concept is provided to all radiation workers at the University. Although ALARA training is

routinely included in all radiation worker training at the University, it has not been specifically listed in Section 2.5.2. of the Decommissioning Plan. Therefore, we propose amending this section to read:

"

2.5.2. Radiation Worker Training

Radiation Worker Training (RWT) will be required for all individuals directly associated with the UVAR Decommissioning, and the training will include the following topics:

Fundamentals of Radiation
Biological Effects of Radiation
External Radiation Exposure Limits and Controls
Internal Radiation Limits and Controls
ALARA Program (Program, Objectives, Investigational Limits, Keeping Doses ALARA)
Contamination Limits and Controls
Management and Control of Radioactive Waste.
"

26. Section 2.7, "Facility Release Criteria," **Page 2-22. You have identified isotopes as being present on site beyond those listed in Table 2-6. Please address.**

The following amended Table 2-6 lists all the radionuclides that were found during the site characterization. The release of surfaces is based on the "sum of fraction" rule. In the application of this rule, a mixture of radionuclides in measured abundances is used to generate a dpm/100 cm^2 limit for the mixture. The abundances of the radionuclides added to this table will not result in a change to the gamma and beta emitter limit for the mixture.

The revised Table 2-6 is shown below. About half the values are expected to change based on ALARA analysis.

Table 2-6 License Termination Screening Values for Building Surface Contamination

Radionuclide	Symbol	Acceptable screening levels[1] for unrestricted release (dpm/100 cm^2)[2]
Hydrogen-3 (Tritium)	^{3}H	1.2E+08
Carbon-14	^{14}C	3.7E+6
Manganese-54	^{54}Mn	3.2E+04
Iron-55	^{55}Fe	4.5E+06
Cobalt-57[3]	^{57}Co	2.12E+05
Cobalt-60	^{60}Co	7.1E+03
Nickel-63	^{63}Ni	1.8E+06
Zinc-65[3]	^{65}Zn	4.81E+04
Strontium-90	^{90}Sr	8.7E+03

Radionuclide	Symbol	Acceptable screening levels[1] for unrestricted release (dpm/100 cm^2)[2]
Niobium-94[3]	^{94}Nb	8.28E+03
Technetium-99	^{99}Tc	1.3E+06
Antimony -125[3]	^{125}Sb	4.43E+04
Cesium-137	^{137}Cs	2.8E+04
Europium-152[3]	^{152}Eu	1.27E+04
Europium-154[3]	^{154}Eu	1.15E+04
Iridium-192	^{192}Ir	7.4E+04
Radium-226[3] +Decay Chain	^{226}Ra	3.14E+02
Natural Uranium[3]	Unat	9.51E+01
Uranium-233[3] +Decay Chain	^{233}U	4.96E+00
Uranium-234[3]	^{234}U	8.99E+01
Uranium-238[3] +Decay Chain	^{238}U	1.94E+01
Plutonium-241[3]	^{241}Pu	1.41E+03
Americium-241[3]	^{241}Am	2.68E+01

[1] Screening levels are based on the assumption that the fraction of removable surface contamination is equal to 0.1.

[2] Units are disintegrations per minute per 100 square centimeters (dpm/100 cm^2). 1 dpm is equivalent to 0.0167 becquerel (Bq). The screening values represent surface concentrations of individual radionuclides that would be deemed in compliance with the 0.25 mSv/yr (25 mrem/yr) unrestricted release dose limit in 10 CFR 20.1402. For radionuclides in a mixture, the "sum of fractions" rule applies; see 10 CFR Part 20, Appendix B, Note 4. NRC Draft Guidance DG-4006 for provides further information on application of the values in this table.

[3] The screening values represent surface concentrations of individual radionuclides that would be deemed in compliance with the 0.25 mSv/yr (25 mrem/yr) and were calculated using the NRC DandD code version 1.0, Build 1.00.02.

27. Section 3.1.1, "Ensuring As low As Reasonably Achievable (ALARA) Radiation Exposures," Page 3-1. 10 CFR 20.1402 contains a requirement that residual radioactivity be reduced to levels that are ALARA. Please discuss how you plan to show compliance with this part of the regulation.

The ALARA requirement will be met by performing an ALARA analysis. See in attachment REFS-CALC-UVAR-001, DOSE ASSESSMENT FOR UVAR DECOMMISSIONING PLAN.

28. **Section 3.1.3, "Radioactive Materials Controls," Page 3-13. The NRC standard for the release of materials as clean waste is no detectable activity. Please review the information in IE Circular 81-07, "Control of Radioactively Contaminated Material," May 14, 1981, and IE Information Notice 85-92, "Survey of Wastes Before Disposal From Nuclear Reactor Facilities," December 2, 1985 (attached). Your proposed values based on license termination screening values are too high.**

We propose to change our discussion on releasing decommissioning waste materials from the site. We will not release any decommissioning waste from the UVAR site for disposal at a landfill. All decommissioning waste will be shipped to an off-site licensed radioactive waste processing facility for surveying and disposal.

With this position, IN 85-92 and 81-07 will not apply. Section 3.1.3 of the Plan has been revised, and the revised section is shown below:

3.1.3 Radioactive Materials Controls

UVA's radiation protection program establishes radioactive material controls that ensure the following:

- Prevention of inadvertent decommissioning radioactive waste (licensed) material release to uncontrolled areas.
- Assurance that personnel are not inadvertently exposed to radiation from licensed radioactive decommissioning waste materials.
- Minimization of the amount of radioactive waste material generated during decommissioning.

Decommissioning waste materials will not be released as clean waste. Such waste materials to be removed from the Reactor Facility will be shipped to an off-site licensed radioactive waste processing facility for survey, processing and disposal.

Poolwater releases will be analyzed and filtered to ensure that discharges to sanitary sewerage will meet the requirements of 10 CFR 20.2003 disposal by release into sanitary sewerage and University of Virginia liquid discharge procedures.

29. **Section 3.1.4, "Dose Estimate," Page 3-14. Please discuss how the task-specific dose estimates shown on Table 3-2 were calculated.**

The task-specific dose estimates shown on Table 3-2 were calculated in the estimating module of Xtreme PM^{SM}. First tasks are entered, then resources including various categories of workers are entered, and the task duration is entered. This information is used to generate the man-hours required to complete a task. Each task is also assigned an estimated average dose rate. This dose rate is multiplied by the man-hours for the task to generate the anticipated dose for the task. The task estimating structure was generated such that only production personnel were included in tasks that had dose rates above the estimated facility background dose rate. All other personnel were assigned the estimated facility background dose rate for their activities.

30. **Section 3.2.2. "Radioactive Waste Disposal," Page 3-17. This section of the DP discusses a UVA approved Project-Specific Quality Assurance Plan. Is this plan written to meet the requirements of 10 CFR Part 71, Subpart H? If so has it been submitted to NRC for approval?**

The UVA Quality Assurance Program for Radioactive Materials Packages, Appendix I, Rev. 2 was written to assure compliance with 10 CFR Part 71, Subpart H was submitted to NRC on August 11, 1997 and approved by the NRC. The plan has the expiration date of September 30, 2002.

31. **Section 3.4, "Radiological Accident Analysis," Page 3-19. Please discuss the specific contractor experience referred to as justification for classifying handling accidents as low risk. Could activated and/or contaminated components be dropped during handling resulting in an airborne release? If so, please analyze this event. Please justify the statement that the radiological hazard resulting from a fire would be minimal.**

A "Radiological Accident Analysis" was performed. This analysis, REFS-CALC-UVAR-002, Rev. 0, *Radiological Accident Analysis for UVAR Decommissioning Plan*, is attached.

32. **Section 4.2, "Background Survey Results," Page 4-2. Please discuss the "activities of other licensees" considered as possible sources contributing to man-made background radiation.**

To the best of our knowledge, other radiological activities that may contribute to man-made background radiation include the ongoing operations conducted under the University of Virginia's Broadscope License, the nuclear medicine and radiation oncology operations at Martha Jefferson Hospital, small private companies conducting research and development, industrial and density gages used the manufacturing and construction industries, and the now-decommissioned, DOE Ordinance Research Laboratory was located nearby the UVAR.

University of Virginia (UVa) Broadscope License activities

The University of Virginia has active research and clinical programs that use radioactive material. These programs include nuclear medicine, nuclear cardiology, radiation oncology, extensive research in medicine, the life sciences and the physical sciences. These programs use radioactive material in unsealed forms. An active environmental monitoring program collects and analyzes samples to ensure that effluents are ALARA.

Martha Jefferson Hospital

Martha Jefferson Hospital (MJH) is located about three miles from the UVAR in downtown Charlottesville. MJH operates a nuclear medicine department and a radiation oncology department. These two operations use unsealed radioactive materials in expected quantities. The radiopharmaceuticals that are used include I-131, Tc-99m, Tl-201, Xe-133 and others, in diagnostic and therapeutic applications. To our knowledge, MJH does not conduct off-site environmental monitoring.

The Ordinance Research Laboratory (ORL)

The ORL, which was decommissioned by Oak Ridge National Laboratory in the late 80'sperformed uranium separation using a centrifuge. The UVAR site characterization study did not endeavor to qualify or quantify that immediate site. However, there is no indication, as determined from the UVAR site characterization that effluents from that operation impacted the UVAR site.

Small entities using radioactive material

There are a number of small companies located in Charlottesville and the surrounding area that use research quantities of radioactive material. The typical radionuclides being used by these companies include H-3, P-32,C-14, I-125, I-131 and S-35. To our knowledge, none of these companies perform off-site environmental monitoring.

Manufacturing and industrial gages

Construction companies and other industrial operations use Troxler gages and other general licensed material in their routine business. The Troxler gages incorporate sealed Cs-137 sources and there is no expectation that they contribute to background radiation under normal circumstances.

33. **Section 4.0, "Proposed Final Radiation Survey Plan," Pages 4-1 through 4-6. Please provide more detail of the proposed Final Status Survey Plan.**

    a. **Provide plots, diagrams, and facility layout drawings to illustrate the classification of the survey units. For each type of survey unit, discuss unit sizes, grid spacing, scan area selection (when less than 100%), and static measurement number, location, and spacing.**

    During the planning stage for the final status survey, the facility will be sectioned into survey units according to guidance provided in MARSSIM. Plots, diagrams, and facility layout drawings will be developed to illustrate the classification of the survey units. In addition to a final status survey plan, a survey package portfolio will be developed for each survey unit. Each survey package will include survey unit specific instructions, describe the survey unit size, grid spacing, scan area prescribed and prescribed number of static measurements including the location and spacing. Once the planning stage is complete, the survey packages with instructions, plots, diagrams, and facility layout drawings illustrating the classification of the survey units will be provided for review.

    b. **Please discuss facility history, characterization survey results, and evaluations to support the classification of the survey units.**

    The final status survey plan and survey package portfolios for each survey unit will include a discussion regarding the facility history, characterization survey results, and evaluations used to support survey unit classification. The survey plan and survey packages will be provided for review after the decommissioning contract is awarded.

    c. **Please list and discuss the proposed DCGLs and how they were selected. If DandD was used to estimate the DCGLs please provide a copy of the DandD report which shows the version used. Please discuss the input parameters used and justification for using those parameters. If other modeling software was used to develop site-specific DCGLs, please discuss the use of that software. What pathway scenario was used? Please show how the DCGLs meet the 10 CFR 20.1402 dose limits. Were surrogate ratio DCGLs used to develop DCGLs for "hard-to-detect" radionuclides? If so, please discuss. If you developed any gross activity DCGLs or elevated measurement comparison DCGLs, please discuss.**

    An ALARA analysis was performed for the University of Virginia Reactor site. The proposed DCGL values based on this analysis are with the response to question 27. For ALARA analysis details refer to REFS-CALC-UVAR-001, DOSE ASSESSMENT FOR UVAR DECOMMISSIONING PLAN, attached.

Surrogate ratio DCGLs will be developed for the final status survey, after the decommissioning contract is awarded, that will consider "hard-to-detect" radionuclides for soil. Gross activity DCGLs and $DCGL_{EMC}$ (elevated measurement comparison DCGLs) will be developed during the planning stage for the final status survey that will consider "hard-to-detect" radionuclides for surfaces. The site specific DCGLs will be calculated based on the relative fraction of each radionuclide in the expected radionuclide mix. Once developed, the values will be submitted for review and concurrence.

d. **Please discuss the method that will be used to reclassify a survey unit.**

The criteria used for designating areas, as Class 1, 2, or 3 will be described in the final status survey plan. Compliance with the classification criteria will be demonstrated in the final status survey report. A thorough analysis of HSA findings and the results of the characterization survey will provide the basis for each area's classification. As the survey progresses, reevaluation of this classification may be necessary based on newly acquired survey data. For example, if contamination is identified in a Class 3 area > 50% of the DCGL, an investigation and reevaluation of that area will be performed to determine if the Class 3 area classification was appropriate. The investigation may result in part or all of the area being reclassified as a Class 1 or Class 2 survey unit for re-survey after remediation, if necessary. If survey results identify residual contamination in a Class 2 area exceeding the DCGL or suggest that there may be a reasonable potential that contamination is present in excess of the DCGL, an investigation will be initiated to determine if all or part of the area should be reclassified to a Class 1 survey unit for re-survey after remediation, if necessary.

e. **Please discuss how changes in the proposed final survey will be made.**

After the decommissioning contract is awarded, a final status survey plan will be developed according to guidance provided in MARSSIM. The survey plan will serve as the guidance document for development of the survey package instructions used during implementation of the final status survey. The survey plan will be provided to the UVAR Decommissioning Committee for review and concurrence prior to the performance of a final status survey.

As the survey progresses, reevaluation of the survey plan may be necessary based on newly acquired survey data. If a condition not encompassed by the survey plan is discovered, the survey plan may undergo revision to address the condition. The condition with the revised survey plan will be fully disclosed and provided to the Reactor Decommissioning Committee for review and concurrence prior to further performance of the final status survey as it applies to the revised information.

f. **Please provide a description of the access control procedures to prevent recontamination of clean areas.**

Administrative and physical access controls of surveyed radiologically clean areas will be instituted to prevent recontamination of the clean areas. Control of surveyed areas will be accomplished administratively by written instruction contained in the final survey plan and by training of project personnel. Control of surveyed areas may be accomplished physically by placing rope barriers, locking doors where able, etc. and placing signs to notify personnel regarding the condition of an area.

g. **Please discuss the assessment of your survey data including statistical tests and data conclusions.**

After the decommissioning contract is awarded, a final status survey plan will be developed according to guidance provided in MARSSIM. The survey plan will contain the criteria used to assess all final survey data including the statistical tests performed and state the conclusion based upon statistical test results. The survey plan will be provided for review and concurrence prior to performance of the final status survey.

The final status survey plan will be developed according to the guidance provided in MARSSIM and based upon the following assumptions:

- The results of the HSA and characterization have been reviewed and the site was determined to be impacted based on the operational history, characterization data and professional judgment.
- The null hypothesis recommended for use in MARSSIM is: "The residual radioactivity in the survey unit exceeds the release criterion."
- The decision error rates will be set to 0.05 for Type I (α) error and 0.05 for Type II (β) error.
- The Sign test, nonparametric statistical test, will be used to compares the distribution of a set of measurements in a survey unit to the DCGL. Note that values from a background study for materials of construction performed during the characterization survey will be used to adjust final survey direct measurements for background radiation. The material background adjustment to the final survey direct measurements will eliminate the need for a background reference area required if using the Wilcoxon Rank Sum (WRS) test.
- Once the final survey has been performed, survey data will be converted to DCGL units and compared to the DCGLs. Individual measurements and sample concentrations will be compared to DCGL levels for evidence of small areas of elevated activity. Data will then be evaluated using the Sign test statistical method to determine if they exceed the release criterion. If the release criterion has been exceeded (null hypothesis proven true) or if results indicate the need for additional data points, appropriate further actions will be determined by the UVA, Project Manager and the NRC.
- If the release criterion has not been exceeded (null hypothesis proven false), the results of the survey will be compared with the data quality objectives established during the planning phase of the project. If the data quality objectives have been satisfied, the survey unit will be suitable to release for unrestricted use.

34. **Section 4.5.1, "Laboratory/Radiological Measurements Quality Assurance," Page 4-7. Please discuss the development of the QA Approved Suppliers List. What position is responsible for its maintenance?**

The decommissioning contractor selected must have a QA program that meets the requirements under 10 CFR 71 "Packaging and Transportation of Radioactive Material", Subpart H "Quality Assurance". In addition the contractor's QA program must meet the applicable criteria from 10 CFR 50, Appendix B; the American Society of Mechanical Engineers (ASME) NQA-1. One of the applicable criteria that must be included is a QA Approved Suppliers List. The contractor will maintain the QA Approved Suppliers List. The University will assess the effectiveness of the contractors QA program either through direct audits performed by either the Decommissioning Committee, or by

audits performed by contracted audit personnel, or by the acceptance of audits performed by other organizations.

35. **TS 6.2.A.2 (1), Page 41. Please discuss the role of the ex-officio members of the Radiation Safety Committee.**

The Radiation Safety Committee (RSC) has two ex-officio members, the Radiation Safety Officer (RSO) and a University management representative. These are voting positions and they receive standing appointments to the RSC. In this capacity, the RSO is responsible for providing expertise and advice in matters related to the use and control of radioactive material and radiation producing equipment, and for implementing directives from the RSC. The University's management representative is a member of the RSC to assure that the appropriate levels of management (those that are outside the Office of Environmental Health and Safety) are informed of issues important to radiation safety.

36. **Environmental Report, Section 1.0, "Purpose and Need for Action," Page B4. Please describe the need for the action of approving a decommissioning amendment for the University of Virginia Research Reactor. Why is the University asking for the amendment?**

The University of Virginia shut off entry to students into its Nuclear Engineering Programs in January of 1998 and permanently shut down the UVAR on July 1, 1998. Currently, there are no plans to ever again resume reactor operations. Temporarily mothballing the reactor is not a reasonable option. Hence, the University wishes to proceed with its decommissioning and the termination of the associated reactor license. UVA therefore has filed the appropriate decommissioning amendment requests together with a decommissioning plan with the NRC.

37. **Environmental Report, Section 2.2, "Proposed Action and Alternatives," Page B-12. Please discuss the environmental impact of a proposed action of taking no action.**

A no-action alternative would leave the facility in its current status with the current support staff having to maintain the facility under the existing license conditions. This action will involve maintaining:

- The facility license with the associated Technical Specifications
- Personnel to support facility maintenance and surveillance
- Surveillance and maintenance of Reactor Pool Water Level, Purity and pH
- The Reactor Facility physical security plan

The reactor pool still contains activated hardware with some items reading over 200R/hr on contact. The reactor pool has a history of minor leakage, despite the many repairs attempted over the operating history of the facility. Keeping the facility in this status over a long period of time can be expected to lead to a continuing degradation of the pool. That degradation will require either the repair, or the decommissioning of that portion of the facility, which constitutes a major portion of the decommissioning.

There is contaminated soil in the area below the underground liquid waste storage tanks. This activity has the potential to migrate and contaminate a larger area of the facility over time.

The facility will not become available for conversion into useable space for other university educational needs. The university will incur expenses for maintenance of the facility without making beneficial use of the facility.

The NRC requirement in 10CFR 50.82(b)(1)(ii) providing for non-power reactor decommissioning without significant delay following permanent shutdown would not be satisfied.

38. **Environmental Report, Section 3.2.4, "Biology," Page B-26. Are there any endangered or threatened Federal or State species on site? What is the basis for your conclusion?**

There are no known endangered and threatened plant or animal species at the UVA Reactor site. The Virginia Department of Conservation and Recreation's (DCR) database on endangered and threatened species was accessed to make this determination. We also consulted the Virginia Natural Heritage Program home-page on the world-wide web.

Endangered and Threatened Species

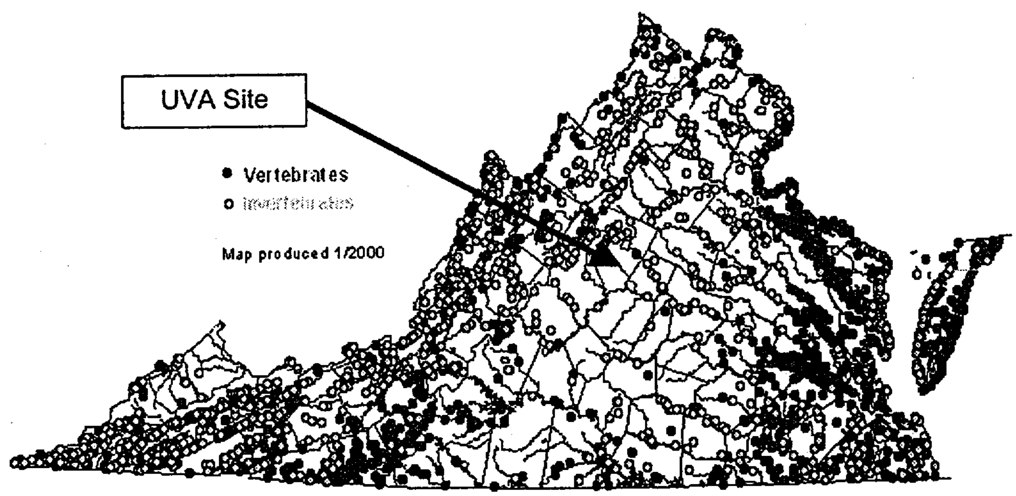

Figure 1. Known presence of Threatened and Endangered Animal Species.

Notes About This Map: The data depicted here are not based on a comprehensive inventory of VA. The lack of data for any geographic area cannot be construed to mean that no significant features are present. VANHP is not responsible for any inaccuracies in the data and does not necessarily endorse any interpretations or products derived from the data. The data were compiled from a variety of sources including field surveys, universities, systematic collections, government agencies, and individual biologists. Additions and changes to these data are constant. This map only depicts the state of knowledge at the listed date.

39. **Environmental Report, Section 4.1.2, "Potential Exposures," Page B-28. What are the potential exposures to the decommissioning staff and the public from potential accidents?**

A "Radiological Accident Analysis" was performed. This analysis, REFS-CALC-UVAR-002, Rev. 0, Radiological Accident Analysis for UVAR Decommissioning Plan, is in attachment.

40. **Environmental Report, Section 4.2.3, "Non-Hazardous Solid Waste," Page B-30. Please estimate the volume of construction debris that will be sent to the local sanitary landfill.**

We propose to change our discussion on releasing materials from the site. Section 4.2.3 has been revised to indicate that no decommissioning waste materials will be released from the site for disposal at a landfill. The proposed revision to section 4.2.3 is given below.

4.2.3 Non-Hazardous Solid Waste

We will not release any decommissioning waste materials from the site for disposal at a landfill. All radioactive decommissioning waste for disposal will be shipped to an off-site licensed processing facility for surveying and disposal.

ATTACHMENTS TO ANSWERS TO NRC'S R.A.I.

- **New UVAR TS 4.11**
- **Revised UVAR TS 6.3.1,**
- **New UVAR TS 5.2**
- **Revised TS 6.1.2**
- **REFS-CALC-UVAR-001, Revision 0:
  Dose Assessment
  for UVAR Decommissioning Plan**
- **REFS-CALC-UVAR-002, Revision 0:
  Radiological Accident Analysis
  for UVAR Decommissioning Plan**

PROPOSED NEW UVAR TS 4.11

4.11 Surveillance of Decommissioning Instrumentation

Applicability: This specification applies to the traceability and frequency of the calibration of those field and laboratory radiation detection instrumentation, and associated detectors, used in decommissioning activities at the UVAR Facility.

Objective: The objective is to have only legally well-calibrated radiation survey and detection instrumentation used in decommissioning work.

Specification:

Laboratory instruments and associated detectors used in decommissioning activities shall be calibrated on an annual basis.

Field radiation detection instruments and associated detectors used in decommissioning activities shall be calibrated on an annual basis.

National Institute of Standards and Technology (NIST) traceable sources and appropriate calibration equipment shall be used in the calibration of this equipment.

Basis: Accurate measurements to meet license conditions and federal regulations require that properly calibrated instrumentation be used.

PROPOSED NEW ITEM 8 TO CURRENT UVAR TS 6.3.1

6.3.1. Items Covered by SOPs

....

(8) Maintenance, response testing and record keeping involving radiation detecting field instrumentation and associated detectors utilized in the decommissioning of the Reactor Facility.

PROPOSED NEW UVAR TS 5.2

5.2.1 Temporary Pool Confinement

Applicability: This specification applies to the utilization of a confinement barrier surrounding the reactor pool, with an associated local ventilation system, operating whenever airborne hazards could arise within the reactor pool during decommissioning work.

Objective: The barrier surrounding the reactor pool and its associated ventilation and filtration system are intended to minimize potential risks associated with worker inhalation of radioactive material made airborne by D&D work.

Specification: While decommissioning activities involving the reactor pool are in progress, such that airborne hazards may be produced, a confinement barrier surrounding the reactor pool shall have been erected and placed into use. A local ventilation system shall be operating during these periods, to ensure negative pressure within the confinement with respect to the Reactor Room and to provide high-efficiency filtration of the air exhausted from the enclosure.

Basis: The barrier and ventilation system together will ensure that reactor pool confinement air is scrubbed clean by high-efficiency filters prior to release to the Reactor Room.

PROPOSED ADDITION TO THE END OF CURRENT UVAR TS 6.1.2.

6.1.2. Responsibility

......

The Radiation Safety Officer shall be responsible for providing radiological support in the decommissioning of the UVAR. This function ensures that the activities involving potential radiological exposure are conducted in compliance with the applicable licenses, Federal and State regulations, and UVAR standard operating procedures. The position includes responsibility for maintaining the UVAR surveillance and monitoring program and for HP radiological protection procedures.

The minimum qualifications for the Radiation Safety Officer positions are a four-year degree in Health Physics or a related field, three years supervisory experience in Health Physics and five years operational experience related to radiation safety.

Proposed Version, with vertical lines in right margin to indicate changes

APPENDIX A

TECHNICAL SPECIFICATIONS

FOR THE

UNIVERSITY OF VIRGINIA REACTOR

FACILITY LICENSE No. R-66
DOCKET No. 50-62

As Revised to Facilitate
Permanent Reactor Shutdown, Decontamination and Decommissioning

Effective Dates of Amendments
February 9, 2000 - - #25
January xx, 2001 - - #26

4.11 Surveillance of Decommissioning Instrumentation

Applicability: This specification applies to the traceability and frequency of the calibration of those field and laboratory radiation detection instrumentation, and associated detectors, used in decommissioning activities at the UVAR Facility.

Objective: The objective is to have only legally well-calibrated radiation survey and detection instrumentation used in decommissioning work.

Specification:

Laboratory instruments and associated detectors used in decommissioning activities shall be calibrated on an annual basis.

Field radiation detection instruments and associated detectors used in decommissioning activities shall be calibrated on an annual basis.

National Institute of Standards and Technology (NIST) traceable sources and appropriate calibration equipment shall be used in the calibration of this equipment.

Basis: Accurate measurements to meet license conditions and federal regulations require that properly calibrated instrumentation be used.

5.2. Reactor Building.

TS 5.2 has been deleted, for the specifications on confinement, ventilation and reactor room free volume have been required to restrict leakage of radionuclides produced during reactor operation at power. The UVAR is no longer operated.

5.2.1 Temporary Pool Confinement

Applicability: This specification applies to the utilization of a confinement barrier surrounding the reactor pool, with an associated local ventilation system, operating whenever airborne hazards could arise within the reactor pool during decommissioning work.

Objective: The barrier surrounding the reactor pool and its associated ventilation and filtration system are intended to minimize potential risks associated with worker inhalation of radioactive material made airborne by D&D work.

Specification: While decommissioning activities involving the reactor pool are in progress, such that airborne hazards may be produced, a confinement barrier surrounding the reactor pool shall have been erected and placed into use. A local ventilation system shall be operating during these periods, to ensure negative pressure within the confinement with respect to the Reactor Room and to provide high-efficiency filtration of the air exhausted from the enclosure.

Basis: The barrier and ventilation system together will ensure that reactor pool confinement air is scrubbed clean by high-efficiency filters prior to release to the Reactor Room.

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The Reactor Supervisor shall have the equivalent of a bachelor's degree in science or engineering and at least 2 years of experience in Reactor Operations at this facility, or an equivalent facility, or at least 6 years of experience in Reactor Operations. Equivalent education or experience may be substituted for a degree. Within nine months after being assigned to the position, the Reactor Supervisor shall obtain and maintain a NRC Senior Reactor Operator license if reactor fuel elements are still at the Facility. A NRC Senior Reactor Operator license, or a Reactor Operator license, is not required for level 3 and 4 personnel once all reactor fuel elements have been shipped offsite.

The Radiation Safety Officer shall be responsible for providing radiological support in the decommissioning of the UVAR. This function ensures that the activities involving potential radiological exposure are conducted in compliance with the applicable licenses, Federal and State regulations, and UVAR standard operating procedures. The position includes responsibility for maintaining the UVAR surveillance and monitoring program and for HP radiological protection procedures.

The minimum qualifications for the Radiation Safety Officer positions are a four-year degree in Health Physics or a related field, three years supervisory experience in Health Physics and five years operational experience related to radiation safety.

6.1.3. Staffing

A licensed Senior Reactor Operator shall supervise any movement of reactor fuel. One or more health physicists, organizationally independent of the Reactor Staff as shown in Figure 6.1, shall be responsible for radiological safety at the Reactor Facility.

6.1.4. Selection and Training of Personnel

The selection, training and requalification of Reactor Facility personnel shall follow the American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4-1988, Sections 4-6, to the extent applicable to the decommissioning status of the facility. The selected criteria for the personnel will be contained in the NRC-approved Operator Requalification Program, as amended.

Bases: Sections 6.1, 6.1.1, 6.1.2, 6.1.3 and 6.1.4 of the American National Standard ANSI/ANS 15.1-1990 "The Development of Technical Specifications for Research Reactors," describe a generic and generally acceptable organizational structure for U.S. research reactors.
They provide the bases for TS 6.1 above. Some of the ANSI standard recommendations apply to operable or operating reactor facilities, and are not necessarily valid for staff hired to perform decommissioning activities.

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6.3. Standard Operating Procedures

Applicability: The specification below concerns the procedural controls used to operate the University of Virginia Reactor (UVAR) and conduct experiments.

Objective: The objective is the safe operation of the reactor in compliance with license conditions, federal regulations.

Specifications:

6.3.1. Items Covered by SOPs

Written procedures, reviewed and approved by the Reactor Safety Committee shall be in effect and followed for the items listed below. These procedures shall be adequate to ensure the safe decommissioning of the reactor, but should not preclude the use of independent judgment and action should the situation require such.

(1) Startup, operation and shutdown of the reactor.

(2) Installation or removal of fuel elements, control rods, experiments, and experimental facilities.

(3) Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms, suspected primary coolant system leaks, abnormal reactivity changes.

(4) Emergency conditions involving potential or actual release of radioactivity, including provisions for evacuation, re-entry, recovery, and medical support.

(5) Preventative and corrective maintenance operations that could have an effect on reactor safety.

(6) Periodic surveillance.

(7) Radiation control.

(8) Maintenance, response testing and record keeping involving radiation detecting field instrumentation and associated detectors utilized in the decommissioning of the Reactor Facility.

6.3.2. Changes to SOPs

Substantive changes to approved procedures shall be made only with the approval of the Reactor Safety Committee (or by the Reactor Decommissioning Committee after the ReSC ceases to exist). Changes that do not change the original intent of the procedures may be made with the approval of the Facility Director. All such minor changes shall be documented and subsequently reviewed by the Reactor Safety Committee (or by the Reactor Decommissioning Committee after the ReSC ceases to exist).

Basis: Section 6.4 of American National Standard ANSI/ANS 15.1-1990, "The Development of Technical Specifications for Research Reactors," suggests acceptable procedural controls to applied to operating U.S. research reactors.

Proposed Version, with no vertical lines in right margin to indicate changes

APPENDIX A

TECHNICAL SPECIFICATIONS

FOR THE

UNIVERSITY OF VIRGINIA REACTOR

FACILITY LICENSE No. R-66
DOCKET No. 50-62

As Revised to Facilitate
Permanent Reactor Shutdown, Decontamination and Decommissioning

<u>Effective Dates of Amendments</u>
February 9, 2000 - - #25
January xx, 2001 - - #26

TABLE OF CONTENTS

4.11 Surveillance of Decommissioning Instrumentation

Applicability: This specification applies to the traceability and frequency of the calibration of those field and laboratory radiation detection instrumentation, and associated detectors, used in decommissioning activities at the UVAR Facility.

Objective: The objective is to have only legally well-calibrated radiation survey and detection instrumentation used in decommissioning work.

Specification:

Laboratory instruments and associated detectors used in decommissioning activities shall be calibrated on an annual basis.

Field radiation detection instruments and associated detectors used in decommissioning activities shall be calibrated on an annual basis.

National Institute of Standards and Technology (NIST) traceable sources and appropriate calibration equipment shall be used in the calibration of this equipment.

Basis: Accurate measurements to meet license conditions and federal regulations require that properly calibrated instrumentation be used.

5.2. Reactor Building

TS 5.2 has been deleted, for the specifications on confinement, ventilation and reactor room free volume have been required to restrict leakage of radionuclides produced during reactor operation at power. The UVAR is no longer operated.

5.2.1 Temporary Pool Confinement

Applicability: This specification applies to the utilization of a confinement barrier surrounding the reactor pool, with an associated local ventilation system, operating whenever airborne hazards could arise within the reactor pool during decommissioning work.

Objective: The barrier surrounding the reactor pool and its associated ventilation and filtration system are intended to minimize potential risks associated with worker inhalation of radioactive material made airborne by D&D work.

Specification: While decommissioning activities involving the reactor pool are in progress, such that airborne hazards may be produced, a confinement barrier surrounding the reactor pool shall have been erected and placed into use. A local ventilation system shall be operating during these periods, to ensure negative pressure within the confinement with respect to the Reactor Room and to provide high-efficiency filtration of the air exhausted from the enclosure.

Basis: The barrier and ventilation system together will ensure that reactor pool confinement air is scrubbed clean by high-efficiency filters prior to release to the Reactor Room.

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The Reactor Supervisor shall have the equivalent of a bachelor's degree in science or engineering and at least 2 years of experience in Reactor Operations at this facility, or an equivalent facility, or at least 6 years of experience in Reactor Operations. Equivalent education or experience may be substituted for a degree. Within nine months after being assigned to the position, the Reactor Supervisor shall obtain and maintain a NRC Senior Reactor Operator license if reactor fuel elements are still at the Facility. A NRC Senior Reactor Operator license, or a Reactor Operator license, is not required for level 3 and 4 personnel once all reactor fuel elements have been shipped offsite.

The Radiation Safety Officer shall be responsible for providing radiological support in the decommissioning of the UVAR. This function ensures that the activities involving potential radiological exposure are conducted in compliance with the applicable licenses, Federal and State regulations, and UVAR standard operating procedures. The position includes responsibility for maintaining the UVAR surveillance and monitoring program and for HP radiological protection procedures.

The minimum qualifications for the Radiation Safety Officer positions are a four-year degree in Health Physics or a related field, three years supervisory experience in Health Physics and five years operational experience related to radiation safety.

6.1.3. Staffing

A licensed Senior Reactor Operator shall supervise any movement of reactor fuel. One or more health physicists, organizationally independent of the Reactor Staff as shown in Figure 6.1, shall be responsible for radiological safety at the Reactor Facility.

6.1.4. Selection and Training of Personnel

The selection, training and requalification of Reactor Facility personnel shall follow the American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4-1988, Sections 4-6, to the extent applicable to the decommissioning status of the facility. The selected criteria for the personnel will be contained in the NRC-approved Operator Requalification Program, as amended.

Bases: Sections 6.1, 6.1.1, 6.1.2, 6.1.3 and 6.1.4 of the American National Standard ANSI/ANS 15.1-1990 "The Development of Technical Specifications for Research Reactors," describe a generic and generally acceptable organizational structure for U.S. research reactors.
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Written procedures, reviewed and approved by the Reactor Safety Committee shall be in effect and followed for the items listed below. These procedures shall be adequate to ensure the safe decommissioning of the reactor, but should not preclude the use of independent judgment and action should the situation require such.

(1) Startup, operation and shutdown of the reactor.

(2) Installation or removal of fuel elements, control rods, experiments, and experimental facilities.

(3) Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms, suspected primary coolant system leaks, abnormal reactivity changes.

(4) Emergency conditions involving potential or actual release of radioactivity, including provisions for evacuation, re-entry, recovery, and medical support.

(5) Preventative and corrective maintenance operations that could have an effect on reactor safety.

(6) Periodic surveillance.

(7) Radiation control.

(8) Maintenance, response testing and record keeping involving radiation detecting field instrumentation and associated detectors utilized in the decommissioning of the Reactor Facility.

6.3.2. Changes to SOPs

Substantive changes to approved procedures shall be made only with the approval of the Reactor Safety Committee (or by the Reactor Decommissioning Committee after the ReSC ceases to exist). Changes that do not change the original intent of the procedures may be made with the approval of the Facility Director. All such minor changes shall be documented and subsequently reviewed by the Reactor Safety Committee (or by the Reactor Decommissioning Committee after the ReSC ceases to exist).

Basis: Section 6.4 of American National Standard ANSI/ANS 15.1-1990, "The Development of Technical Specifications for Research Reactors," suggests acceptable procedural controls to applied to operating U.S. research reactors.

DURATEK

Radiological Engineering
And
Field Services

REFS-CALC-UVAR-001

REVISION 0

DOSE ASSESSMENT FOR UVAR DECOMMISSIONING PLAN

PREPARED BY: Paul Ely — Document Preparer — 12/11/00 Date

CHECKED BY: [signature] — Verifier — 12/11/00 Date

APPROVED BY: Robert Hamburd — Manager — 12/12/00 Date

AFFECTED PAGES: All

1. PURPOSE AND OBJECTIVES

This calculation provides the cleanup goals for decommissioning the University of Virginia Reactor (UVAR). Derived concentration guideline levels, DCGLs, are calculated for inclusion in the UVAR Decommissioning Plan. The UVAR DCGLs meet both the 25 mrem per year requirement and the ALARA requirement from the NRC regulation (10 CFR 20.1402). The proposed guideline levels for soil and building surface areas are presented in Tables 1 and 2. The tables included those radionuclides detected by samples analysis during the site characterization. The DCGLs established for alpha emitting radionuclides developed, using DandD code version1, are overly conservative (NMSS Decommissioning Program Standard Review Plan - Draft Report). These radionuclides are not anticipated to be of concern for the UVAR because they have only been found in areas where activity has been concentrated by ion exchangers. If alpha emitters prove to be of concern at a later time, these values may be reevaluated using DandD code version 2 (under development).

Table 1: UVAR Soil DCGLs

Radionuclide	Limit (pCi/g)	Radionuclide	Limit (pCi/g)	Radionuclide	Limit (pCi/g)
H-3	110	Sr-90	1.7	Ra-226	0.70
C-14	12	Nb-94	5.8	U-nat*	0.48
Mn-54	15	Tc-99	19	U-233*	0.46
Fe-55	10,000	Sb-125*	26	U-234	13
Co-57	150	Cs-137	11	U-238	14
Co-60	3.8	Eu-152	8.7	Pu-241	72
Ni-63	2,100	Eu-154	8.0	Am-241	2.1
Zn-65*	6.3	Pb-210	0.90		

*These DCGLs represent DandD code results modified based on conservative ALARA analyses. (Refer to section 5.) All other values are from Reference 3.5 modified based on conservative ALARA analyses.

Table 2: UVAR Building DCGLs

Radionuclide	Limit (dpm/100cm2)	Radionuclide	Limit (dpm/100cm2)	Radionuclide	Limit (dpm/100cm2)
H-3	120,000,000	Sr-90	8,700	Ra-226*	244
C-14	2,860,000	Nb-94*	6,400	U-nat*	95
Mn-54	32,000	Tc-99	1,000,000	U-233*	3.8
Fe-55	4,500,000	Sb-125*	44,300	U-234*	69
Co-57*	212,000	Cs-137	28,000	U-238*	15
Co-60	7,100	Eu-152*	12,700	Pu-24*	1,410
Ni-63	1,540,000	Eu-154*	11,500	Am-241*	21
Zn-65*	48,000	Pb-210*	543		

*These DCGLs represent DandD code results modified based on conservative ALARA analyses. (Refer to section 5.) All other values are from Reference 3.4 modified based on conservative ALARA analyses. The most conservative ALARA scenario from table 9 in section 5 was used for the proposed guideline levels.

2. ASSUMPTIONS/INPUTS

2.1 Assumptions

2.1.1 The following assumptions are assumed to be valid for surface contamination in UVAR buildings at the time of the final status survey:

1. Residual radioactivity will be reduced to levels that are "as low as is reasonably achievable" (ALARA),

2. The residual radioactivity is contained in the top layer of the building surface (i.e., there is no volumetric contamination), and

3. The fraction of removable surface contamination does not exceed 0.1.

2.1.2 The following assumptions are assumed to be valid for surface soil contamination at the UVAR site at the time of the final status survey:

1. Residual radioactivity has been reduced to levels that are ALARA,

2. The vertical saturated hydraulic conductivity at the specific site is greater than the infiltration rate.

3. After soil remediation, the excavated areas will be backfilled with uncontaminated soil from offsite.

2.1.3 The numerical release criterion proposed for demonstrating that the dose criterion has been met will be that the sum-of-fractions quotients of concentrations and DCGLs of contributing radionuclides shall be less than unity. If a survey unit fails to meet this numerical release criterion, the need for additional sampling or remediation will be evaluated.

2.2 The default DandD parameters for the resident farmer and building occupancy scenarios were used.

3. REFERENCES

3.1 Beyeler, W. E., et al. 1998. *Review of Parameter Data for the NUREG/CR-5512 Building Occupancy Scenario and Probability Distributions for the DandD Parameter Analysis, Letter Report for NRC Project JCN W6227.*

3.2 GTS Duratek Calculation REFS-CALC-UVAR-002, Rev. 0. *Accident Analysis for UVAR Decommissioning Plan.*

3.3. GTS Duratek, *2000. Characterization Survey Report for the University of Virginia Reactor Facility.*

3.4 NRC, 1998. *Supplemental Information on the Implementation of the Final Rule on Radiological Criteria for License Termination.* Federal Register: November 18, 1998 (Volume 63, Number 222).

3.5 NRC, 1999. *Supplemental Information on the Implementation of the Final Rule on Radiological Criteria for License Termination.* Federal Register: December 7, 1999 (Volume 64, Number 234).

3.6 NRC, 1999. Draft Regulatory Guide DG-4006, *Demonstrating Compliance with the Radiological Criteria for License Termination*; including the DandD code Version 1.0 Build 1.00.02.

3.7 NRC, 2000. Use of Screening Values to Demonstrate Compliance with the Final Rule on Radiological Criteria for License Termination. Federal Register: June 13, 2000 (Volume 65, Number 114).

4. ANALYSIS METHODS

4.1 The DandD code was run for surface soils to provide limits for radionuclides not contained in *Supplemental Information on the Implementation of the Final Rule on Radiological Criteria for License Termination*. Federal Register: December 7, 1999 (Volume 64, Number 234). These radionuclides include zinc-65, antimony-125, uranium-233 and natural uranium. Dose rates were calculated from 1 pCi/g of each of these radionuclides. The results are provided in Appendix 6.1 and are summarized in Table 3. These values were then used to calculate the activity (in pCi/g) required to generate a TEDE of 25 mrem per year. The results from theses calculations are presented below in Table 4.

Table 3: Surface Soil Calculated Dose from 1 pCi/g

Radionuclide	mrem/yr per (pCi/g)	Radionuclide	mrem/yr per (pCi/g)	Radionuclide	mrem/yr per (pCi/g)
Zn-65	3.99	Sb-125	0.98	U-233	54
				U-nat	51.7

Table 4: Surface Soil Activity (pCi/g) equivalent to a TEDE of 25 mrem/year

Radionuclide	Limit (pCi/g)	Radionuclide	Limit (pCi/g)	Radionuclide	Limit (pCi/g)
Zn-65	6.3	Sb-125	26	U-233	0.46
				U-nat	0.48

4.2 DandD was also run for surface contamination to of those radionuclides not contained in *Supplemental Information on the Implementation of the Final Rule on Radiological Criteria for License Termination*. Federal Register: November 18, 1998 (Volume 63, Number 222). These radionuclides include cobalt-57, zinc-65, niobium-94, antimony-125, europium-152, europium-154, lead-210, radium-226, uranium-233, uranium-234, uranium-238, plutonium-241, and americium-241. Dose rates were calculated from 1 dpm/100cm^2 of each of these radionuclides. The results are provided in Appendix 6.2 and are summarized in Table 5. These values were then used to calculate the activity (in dpm/100cm^2) required to generate a dose rate of 25 mrem per year. The results from theses calculations are presented below in Table 6.

Table 5: Building Surface Calculated Dose from 1 dpm/100cm^2

Radionuclide	mrem/yr per (dpm/100cm2)
Co-57	1.18E-04
Zn-65	5.20E-04
Nb-94	3.02E-03
Sb-125	5.64E-04
Eu-152	1.97E-03

Radionuclide	mrem/yr per (dpm/100cm2)
Eu-154	2.18E-03
Pb-210	4.60E-02
Ra-226	7.97E-02
U-nat	2.63E-01
U-233	5.04E+00

Radionuclide	mrem/yr per (dpm/100cm2))
U-234	2.78E-01
U-238	1.29E+00
Pu-241	1.77E-02
Am-241	9.34E-01

Table 6: Building Surface Activity (dpm/100cm^2) equivalent to 25 mrem/year

Radionuclide	Limit (dpm/100cm2)
Co-57	2.12E+05
Zn-65	4.81E+04
Nb-94	8.28E+03
Sb-125	4.43E+04
Eu-152	1.27E+04

Radionuclide	Limit (dpm/100cm2)
Eu-154	1.15E+04
Pb-210	5.43E+02
Ra-226	3.14E+02
U-nat	9.51E+01
U-233	4.96E+00

Radionuclide	Limit (dpm/100cm2))
U-234	8.99E+01
U-238	1.94E+01
Pu-241	1.41E+03
Am-241	2.68E+01

5. ALARA CALCULATIONS

The NRC regulation (20.1402) contains two separate and independent requirements for license termination: 1) a dose limit must be met; 2) levels of residual radioactivity must be ALARA. The UVAR Decommissioning Plan meets both requirements.

There are two different ALARA requirements that apply to decommissioning: 1) 20.1101(b) requires means to keep doses ALARA during operations (includes decommissioning activities); 2) 20.1402 requires that levels of residual radioactivity be ALARA in order to terminate a license. This ALARA requirement is the topic of these calculations.

The method used to demonstrate compliance with the license termination ALARA requirement is the quantitative method in NRC DG-4006 (Reference 3.6).

The DG-4006 method evaluates the practicality of a cleanup action by quantitatively comparing its costs with the monetary worth of its benefits. A simple equation is used to evaluate the costs versus the benefits of each potential cleanup action.

The performance of the DG-4006 method includes the following steps: 1) identify potential cleanup actions that will not be necessary to meet the dose limit; 2) for each potential cleanup action, compare the costs vs. the benefits using the equation in DG-4006; 3) if a potential cleanup action is cost beneficial, the action must be performed to meet the ALARA requirement.

This method is performed during the remediation planning process before remediation starts. The method uses characterization data to determine if potential cleanup actions not necessary to meet the dose limit are necessary to meet the ALARA requirement

5.1 ALARA Calculations for Building Surfaces Release Limits

5.1.1 The UVAR site in Charlottesville, Virginia has the potential for tritium, carbon-14, Manganese-54, iron-55, cobalt-57, cobalt-60, nickel-63, zinc-65, strontium-90, niobium-94, cesium-137, europium-152, europium-154, lead-210, radium-226, natural uranium, uranium-233, uranium-234, uranium-238, plutonium-241, and americium-241 on building surfaces from licensed operations. UVA proposes to release the site for unrestricted use. Surfaces with activity levels in excess of the DCGLs would be decontaminated to meet these limits.

Table 7: Dose Based DCGLs (in dpm/100cm^2) for UVAR:

Radionuclide	Limit (dpm/100cm2)	Radionuclide	Limit (dpm/100cm2)	Radionuclide	Limit (dpm/100cm2)
H-3	1.20E+08	**Sr-90**	8.70E+03	**Ra-226***	3.14E+02
C-14	3.70E+06	**Nb-94***	8.28E+03	**U-nat***	9.51E+01
Mn-54	3.20E+04	**Tc-99**	1.30E+06	**U-233***	4.96E+00
Fe-55	4.50E+06	**Sb-125***	4.43E+04	**U-234***	8.99E+01
Co-57*	2.12E+05	**Cs-137**	2.80E+04	**U-238***	1.94E+01
Co-60	7.10E+03	**Eu-152***	1.27E+04	**Pu-24***	1.41E+03
Ni-63	1.80E+06	**Eu-154***	1.15E+04	**Am-241***	2.68E+01
Zn-65*	4.81E+04	**Pb-210**	5.43E+02		

*These DCGLs represent DandD code results. All other values are from Reference 3.4

5.1.2 The ALARA goals proposed by the University of Virginia are lower than the dose based guideline values in Table-7 for C-14, Ni-63, Nb-94, Tc-99, Ra-226, U-nat, U-233, U-234, U-238 and Am-241.

5.1.3 The DG-4006 method was applied to UVAR. For the simplest case, areas that contain only radium for unrestricted release, Equation 18 from DG-4006 was used to calculate the ALARA concentration.

$$\frac{C_{ALARA}}{DCGL_W} = \frac{Cost_T}{\$2{,}000 \times P_D \times 0.025 \times F \times A} \times \frac{r+\lambda}{1-e^{-(r+\lambda)N}}$$

5.1.4 The first step was to identify the potential cleanup actions to be evaluated. The actions proposed include: 1) Decontamination of the surfaces exceeding the DCGL values using several different methods with Envirocare of Utah disposal of waste, and 2) leave the activity in place.

5.1.5 The second step was to estimate the cost of each alternative. The leave in place option has a zero cost. Uranium-238 is the most conservative radionuclide for the ALARA calculations on building surfaces at UVAR. ALARA calculations utilizing all other radionuclides in Table 7 were also performed. The direct cost of surface decontamination with offsite disposal of all waste at Envirocare of Utah was used in the evaluation. These costs are shown as $Cost_T$ in Table 9, with additional detail on the development of these numbers provided in Appendix 6.3.

5.1.6 The third step was to evaluate the other parameters in Equation 18:

Table 8: Equation 18 Parameters

Parameter	Value	Basis
DCGLw	19 dpm/100cm^2	DandD Dose modeling for U-238
P_D	0.09 person/m^2	NUREG-1496, Vol 2, App. B, Table A.1
F	1	Total Removal
$/person-rem	$2,000	NUREG/BR-0058
A	100 m^2	Site Information
r	0.07/yr	NUREG/BR-0058
N	70 yr	NUREG-1496, Vol 2, App. B, Table A.1
λ	1.54E-10 yr-1	rad decay data for U-238

5.1.7 The fourth step was to calculate the ALARA concentrations using the above values in equation 18. The ALARA calculations for several decontamination methods and each radionuclide of concern are given in Table-9 below.

Table 9: Surface Activity ALARA Results

Method	1/4" Scabbling of Floor Surfaces	1/8" Scabbling of Wall Surfaces	Manual Decontamination All Surfaces
$Cost_T$*	$4,931	$11,033	$5,738
H-3 $C_{ALARA}/DCGL_W$	1.39	3.10	1.50
C-14 $C_{ALARA}/DCGL_W$	0.77	1.73	0.84
Mn-54 $C_{ALARA}/DCGL_W$	9.91	22.18	10.72
Fe-55 $C_{ALARA}/DCGL_W$	3.69	8.25	3.99
Co-57 $C_{ALARA}/DCGL_W$	11.03	24.69	11.93
Co-60 $C_{ALARA}/DCGL_W$	2.21	4.94	2.39
Ni-63 $C_{ALARA}/DCGL_W$	0.85	1.91	0.92
Zn-65 $C_{ALARA}/DCGL_W$	12.08	27.03	13.07
Sr-90 $C_{ALARA}/DCGL_W$	1.04	2.33	1.13
Nb-94 $C_{ALARA}/DCGL_W$	0.77	1.73	0.84
Tc-99 $C_{ALARA}/DCGL_W$	0.77	1.73	0.84
Sb-125 $C_{ALARA}/DCGL_W$	3.57	7.99	3.86
Cs-137 $C_{ALARA}/DCGL_W$	1.02	2.29	1.11
Eu-152 $C_{ALARA}/DCGL_W$	1.37	3.05	1.48
Eu-154 $C_{ALARA}/DCGL_W$	1.24	2.78	1.34
Pb-210 $C_{ALARA}/DCGL_W$	3,261	7,298	3,528
Ra-226 $C_{ALARA}/DCGL_W$	0.78	1.74	0.84
U-nat $C_{ALARA}/DCGL_W$	0.77	1.73	0.84
U-233 $C_{ALARA}/DCGL_W$	0.77	1.73	0.84
U-234 $C_{ALARA}/DCGL_W$	0.77	1.73	0.84
U-238 $C_{ALARA}/DCGL_W$	0.77	1.73	0.84
Pu-241 $C_{ALARA}/DCGL_W$	1.34	3.00	1.45
Am-241 $C_{ALARA}/DCGL_W$	0.79	1.77	0.85
H-3 Surface C_{ALARA}	166,000,000	372,000,000	180,000,000
C-14 Surface C_{ALARA}	2,860,000	6,410,000	3,100,000
Mn-54 Surface C_{ALARA}	317,000	710,000	343,000
Fe-55 Surface C_{ALARA}	16,600,000	37,100,000	17,900,000
Co-57 Surface C_{ALARA}	2,340,000	5,230,000	2,530,000
Co-60 Surface C_{ALARA}	15,700	35,100	17,000
Ni-63 Surface C_{ALARA}	1,540,000	3,440,000	1,660,000
Zn-65 Surface C_{ALARA}	581,000	1,300,000	628,000

Method	1/4" Scabbling of Floor Surfaces	1/8" Scabbling of Wall Surfaces	Manual Decontamination All Surfaces
Sr-90 Surface C_{ALARA}	9,070	20,300	9,810
Nb-94 Surface C_{ALARA}	6,400	14,300	6,920
Tc-99 Surface C_{ALARA}	1,000,000	2,250,000	1,090,000
Sb-125 Surface C_{ALARA}	158,000	354,000	171,000
Cs-137 Surface C_{ALARA}	28,600	64,000	30,900
Eu-152 Surface C_{ALARA}	17,300	38,800	18,700
Eu-154 Surface C_{ALARA}	14,200	31,900	15,400
Pb-210 Surface C_{ALARA}	1,770,000	3,970,000	1,920,000
Ra-226 Surface C_{ALARA}	244	546	264
U-nat Surface C_{ALARA}	73	164	79
U-233 Surface C_{ALARA}	4	9	4
U-234 Surface C_{ALARA}	69	156	75
U-238 Surface C_{ALARA}	15	34	16
Pu-241 Surface C_{ALARA}	1,900	4,240	2,050
Am-241 Surface C_{ALARA}	21	47	23

* $Cost_T$ units are given in Appendix 6.3.

5.1.8 This calculation indicates the U-238 $C_{ALARA}/DCGL_W$ ratio is 0.77 for concrete floor surfaces, or the C_{ALARA} value is smaller than the $DCGL_W$ value. For building concrete floor surfaces a U-238 $DCGL_W$ value of 19.4 dpm/100cm^2 results in a C_{ALARA} value of 14.9 dpm/100cm^2. Therefore, building concrete floor surfaces with concentration in excess of 14.9 dpm/100cm^2 U-238 will need to be decontaminated further to meet the ALARA requirement because it is more restrictive than the dose limit.

5.1.9 Similarly these calculations indicate the U-238 $C_{ALARA}/DCGL_W$ ratio is 1.77 for concrete wall surfaces, or the C_{ALARA} value is larger than the $DCGL_W$ value. For building concrete wall surfaces a U-238 $DCGL_W$ value of 19.4 dpm/100cm^2 results in a C_{ALARA} value of 34.3 dpm/100cm^2. Therefore, building concrete wall surfaces must meet the 19.4 dpm/100cm^2 U-238 dose limit only.

5.1.10. In addition these calculations indicate the U-238 $C_{ALARA}/DCGL_W$ ratio is 0.84 for manual decontamination (wipe down) of non-porous surfaces, or the C_{ALARA} value is smaller than the $DCGL_W$ value. For manual decontamination of non-porous surfaces a U-238 $DCGL_W$ value of 19.4 dpm/100cm^2 results in a C_{ALARA} value of 16.5 dpm/100cm^2. Therefore, manual decontamination of non-porous surfaces with concentration in excess of 16.5 dpm/100cm^2 U-238 will need to be decontaminated further to meet the ALARA requirement because it is more restrictive than the dose limit.

5.1.11 All building areas where elevated activities levels are indicated from characterization surveys, will be decontaminated to remove surface material by a method such as HEPA vacuuming prior to evaluating the need for additional remediation by comparison with the ALARA concentration levels.

5.2 ALARA Calculations for Surface Soils

5.2.1 The UVAR site in Charlottesville, Virginia has the potential for tritium, carbon-14, Manganese-54, iron-55, cobalt-57, cobalt-60, nickel-63, zinc-65, strontium-90, niobium-94, cesium-137, europium-152, europium-154, lead-210, radium-226, natural uranium, uranium-233, uranium-234, uranium-238, plutonium-241, and americium-241 in surface soil from licensed operations.

5.2.2 UVA proposes to release the site for unrestricted use. Soil with high concentrations would be removed from the site to a licensed off-site disposal facility.

Table 10: DCGLs (in pCi/g) PROPOSED BY UVA:

Radionuclide	Limit (pCi/g)	Radionuclide	Limit (pCi/g)	Radionuclide	Limit (pCi/g)
H-3	110	Sr-90	1.7	Ra-226	0.70
C-14	12	Nb-94	5.8	U-nat	0.60
Mn-54	15	Tc-99	19	U-233	0.46
Fe-55	10,000	Sb-125	26	U-234	13
Co-57	150	Cs-137	11	U-238	14
Co-60	3.8	Eu-152	8.7	Pu-241	72
Ni-63	2,100	Eu-154	8.0	Am-241	2.1
Zn-65	6.3	Pb-210	0.90		

5.2.3 The ALARA goals proposed by the University are the same as the dose based guideline values in Table-10 above.

5.2.4 The DG-4006 method was applied to UVAR. For the simplest case, areas that contain only uranium-238 for unrestricted release, Equation 18 from DG-4006 was used to calculate the ALARA concentration.

$$\frac{C_{ALARA}}{DCGL_W} = \frac{Cost_T}{\$2{,}000 \times P_D \times 0.025 \times F \times A} \times \frac{r+\lambda}{1-e^{-(r+\lambda)N}}$$

5.2.5 The first step was to identify the potential cleanup actions to be evaluated. The actions proposed include 1) Disposal of soil to Envirocare of Utah; 2) leave the activity in place.

5.2.6 The second step was to estimate the cost of each alternative. The leave in place option has a zero cost. The direct costs of offsite disposal of all surface soils (15 cm depth) over a 100 square meter area with more than 14 pCi U-238 per gram of soil was estimated to be \$42,900 (530 cubic feet or about 30 tons of soil). The direct costs of offsite disposal that contribute to $Cost_T$ are given in Appendix 6.3.

5.2.7 Other costs can include the costs of worker accidents and dose during remediation, traffic fatalities, etc. For UVAR there was no need to estimate other costs because they would not affect the outcome of the analysis.

5.2.8 The third step was to evaluate the other parameters in Equation 18:

Table 12: Equation 18 Parameters

Parameter	Value	Basis
DCGLw	14 pCi/g	DandD Dose modeling for U-238
P_D	0.0004	NUREG-1496, Vol 2, App. B, Table A.1
F	1	Total Removal
$/person-rem	$2,000	NUREG/BR-0058
A	100 m^2	Site Information
R	0.03/yr	NUREG/BR-0058
N	1,000 yr	NUREG-1496, Vol 2, App. B, Table A.1
λ	1.54E-10 yr-1	rad decay data

5.2.9 The fourth step was to calculate the ALARA concentrations using the above values in equation 18. This calculation indicates the U-238 $C_{ALARA}/DCGL_W$ ratio is 636 or the C_{ALARA} value is 636 times larger than the $DCGL_W$ value. For a surface soil U-238 $DCGL_W$ value of 14 pCi/g the C_{ALARA} value is 8,900 pCi/g. Therefore, the 14 pCi/g U-238 surface soil limit applies because it is more restrictive than the 8,900 pCi/g ALARA limit.

5.2.10 Similar calculations were performed for the other radionuclides of concern at the UVAR site. The only Equation 18 values that change for each radionuclide are the $DCGL_W$ value and the λ value. Results from these calculations are presented in the table below.

Table 13: Soil ALARA Calculation Results

Radionuclide	λ	C_{ALARA}/ $DCGL_W$	Soil $DCGL_W$	Soil C_{ALARA}
H-3	0.0565	1,830	110	202,000
C-14	0.000121	639	12	7,660
Mn-54	0.835	18,300	15	275,000
Fe-55	0.267	6,290	10,000	62,900,000
Co-57	0.9373	20,500	150	3,070,000
Co-60	0.132	3,430	3.8	13,000
Ni-63	0.007533	796	2,100	1,670,000
Zn-65	1.03	22,500	6.3	142,000
Sr-90	0.025	1,170	1.7	1,980
Nb-94	0.000035	637	5.8	3,690
Tc-99	3.0E-6	636	19	12,100
Sb-125	0.256	6,060	26	157,000
Cs-137	0.023	1,130	11	12,400
Eu-152	0.0546	1,790	8.7	15,600
Eu-154	0.0433	1,550	8.0	12,400
Pb-210	297	6,310,000	0.90	5,680,000
Ra-226	0.00043	645	0.70	452
U-nat	1.5E-10	636	0.48	305
U-233	4.3E-06	636	0.46	293
U-234	2.8E-06	636	13	8,270
U-238	1.5E-10	636	14	8,900
Pu-241	0.052500	1,750	72	126,000
Am-241	0.0015	668	2.1	1,400

6. APPENDICES

6.1 DandD SESSION: UVAR SOIL DCGLs
6.2 DandD SESSION: UVAR SURFACE DCGLs
6.3 Cost Parameters: $Cost_T$

APPENDIX 6.1
DandD SESSION: UVAR SOIL DCGLs, PRINTOUT FOR RADIONUCLIDES BELOW

Zinc-65, 2 pages
Antimony-125, 2pages
Uranium-233, 2 pages
Uranium-natural, 3 pages

(9 pages of printout attached)

Program   : DandD Version 1.0 Build 1.00.02
Session   : UVA Reactor Soils
Description :
    DCGLs for Soils at the University of Virginia

Executed   : 12/07/00 at 13:10:32

NRC Report

Residential Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	pCi/gram
65Zn	1.00

Code-Generated Radionuclide Activities

Chain	pCi/gram
65Zn	1.0000E+000

Variable Parameters

No parameters have been changed.

Residential Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 1000.00 year(s).

The peak dose of   3.99E+000 TEDE (mrem) occurred 1.00 year(s) after license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	9.59E-001	24.07
Inhalation	1.03E-006	0.00
Agricult.	3.03E+000	75.93
Soil	1.27E-004	0.00
Drinking	1.02E-006	0.00
Irrigated	9.58E-006	0.00
Aquatic	5.35E-005	0.00
Total	3.99E+000	100.00

Radionuclide Component of
Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
65Zn	3.99E+000	100.00
Total	3.99E+000	100.00

```
Program      : DandD Version 1.0 Build 1.00.02
Session      : UVA Reactor Soils
Description :
     DCGLs for Soils at the University of Virginia

Executed     : 12/07/00 at 13:14:10
```

NRC Report

Residential Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	pCi/gram
125Sb	1.00

Code-Generated Radionuclide Activities

Chain	pCi/gram
125Sb	1.0000E+000
125mTe	0.0000E+000

Variable Parameters

No parameters have been changed.

Residential Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 1000.00 year(s).

The peak dose of 9.79E-001 TEDE (mrem) occurred 1.00 year(s) after
license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	9.55E-001	97.60
Inhalation	7.60E-007	0.00
Agricult.	2.35E-002	2.40
Soil	4.34E-005	0.00
Drinking	5.72E-008	0.00
Irrigated	3.71E-008	0.00
Aquatic	4.82E-007	0.00
Total	9.79E-001	100.00

Radionuclide Component of
Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
125Sb	9.72E-001	99.36
125mTe	6.29E-003	0.64
Total	9.79E-001	100.00

```
Program      : DandD Version 1.0 Build 1.00.02
Session      : UVA Reactor Soils
Description :
     DCGLs for Soils at the University of Virginia

Executed     : 12/07/00 at 13:04:55
```

NRC Report

Residential Input Section

Execution Options

```
History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.
```

Initial Radionuclide Activities

Chain	pCi/gram
233U	1.0000

Code-Generated Radionuclide Activities

Chain	pCi/gram
233U	1.0000E+000
229Th	0.0000E+000
225Ra	0.0000E+000
225Ac	0.0000E+000
221Fr	0.0000E+000
217At	0.0000E+000
213Bi	0.0000E+000
213Po	0.0000E+000
209Tl	0.0000E+000
209Pb	0.0000E+000

Variable Parameters

No parameters have been changed.

Residential Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 1000.00 year(s).

The peak dose of 5.41E+001 TEDE (mrem) occurred 4.00 year(s) after
license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	9.38E-005	0.00

Inhalation	5.97E-004	0.00
Agricult.	9.64E-002	0.18
Soil	2.92E-004	0.00
Drinking	1.31E+001	24.14
Irrigated	2.71E+001	50.19
Aquatic	1.38E+001	25.49
Total	5.41E+001	100.00

Radionuclide Component of
Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
233U	5.40E+001	99.82
229Th	8.57E-002	0.16
225Ra	8.01E-003	0.01
225Ac	1.83E-003	0.00
221Fr	4.72E-006	0.00
217At	5.14E-008	0.00
213Bi	3.43E-005	0.00
213Po	0.00E+000	0.00
209Tl	7.46E-006	0.00
209Pb	3.52E-006	0.00
Total	5.41E+001	100.00

```
Program      : DandD Version 1.0 Build 1.00.02
Session      : UVA Reactor Soils
Description  :
     DCGLs for Soils at the University of Virginia

Executed     : 12/08/00 at 14:39:54
```

NRC Report

Residential Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	pCi/gram
U_Nat	1.00

Code-Generated Radionuclide Activities

Chain	pCi/gram
234U	4.8900E-001
230Th	0.0000E+000
226Ra	0.0000E+000
222Rn	0.0000E+000
218Po	0.0000E+000
214Pb	0.0000E+000
218At	0.0000E+000
214Bi	0.0000E+000
214Po	0.0000E+000
210Pb	0.0000E+000
210Bi	0.0000E+000
210Po	0.0000E+000
235U	2.2500E-002
231Th	0.0000E+000
231Pa	0.0000E+000
227Ac	0.0000E+000
223Fr	0.0000E+000
227Th	0.0000E+000
223Ra	0.0000E+000
219Rn	0.0000E+000
215Po	0.0000E+000
211Pb	0.0000E+000
211Bi	0.0000E+000
211Po	0.0000E+000
207Tl	0.0000E+000
238U	4.8900E-001
234Th	0.0000E+000
234mPa	0.0000E+000
234Pa	0.0000E+000
234U	0.0000E+000
230Th	0.0000E+000
226Ra	0.0000E+000
222Rn	0.0000E+000
218Po	0.0000E+000
214Pb	0.0000E+000
218At	0.0000E+000
214Bi	0.0000E+000
214Po	0.0000E+000
210Pb	0.0000E+000

210Bi	0.0000E+000
210Po	0.0000E+000

Variable Parameters

No parameters have been changed.

Residential Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now and ran for 1000.00 year(s).

The peak dose of 5.17E+001 TEDE (mrem) occurred 4.00 year(s) after license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	2.69E-003	0.01
Inhalation	5.45E-004	0.00
Agricult.	9.08E-002	0.18
Soil	2.76E-004	0.00
Drinking	1.24E+001	24.01
Irrigated	2.58E+001	49.91
Aquatic	1.34E+001	25.89
Total	5.17E+001	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
234U	2.59E+001	50.07
230Th	6.20E-004	0.00
226Ra	5.37E-007	0.00
222Rn	3.27E-012	0.00
218Po	7.58E-014	0.00
214Pb	2.16E-009	0.00
218At	0.00E+000	0.00
214Bi	1.27E-008	0.00
214Po	6.91E-013	0.00
210Pb	8.48E-008	0.00
210Bi	7.21E-011	0.00
210Po	4.49E-008	0.00
235U	1.12E+000	2.16
231Th	7.04E-003	0.01
231Pa	1.74E-003	0.00
227Ac	7.26E-005	0.00
223Fr	6.31E-010	0.00
227Th	2.42E-007	0.00
223Ra	3.63E-006	0.00
219Rn	1.81E-009	0.00
215Po	5.86E-012	0.00
211Pb	4.61E-009	0.00
211Bi	1.50E-009	0.00
211Po	7.40E-013	0.00
207Tl	1.11E-010	0.00
238U	2.33E+001	44.97
234Th	1.44E+000	2.78

234mPa	1.35E-003	0.00
234Pa	8.02E-004	0.00
Total	5.17E+001	100.00

APPENDIX 6.2

DandD SESSION: UVAR SURFACE DCGLs, PRINTOUT FOR RADIONUCLIDES BELOW

Cobalt-57, 2 pages
Zinc-65, 2 pages
Niobium-94, 2 pages
Antimony-125, 2pages
Europium-152, 2 pages
Europium-154, 2 pages
Lead-210, 2 pages
Radium-226, 2 pages
Natural Uranium, 3 pages
Uranium-233, 2 pages
Uranium-234, 2 pages
Uranium-238, 2 pages
Plutonium-241, 2 pages
Americium-241, 2 pages

(29 pages of printout attached)

```
Program      : DandD Version 1.0 Build 1.00.02
Session      : UVA Reactor Surfaces
Description :
     DCGLs for Building Surfaces at the University of Virginia
     Reactor

Executed     : 12/07/00 at 13:20:24
```

NRC Report

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	dpm/100cm^2
57Co	1.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
57Co	1.0000E+000

Variable Parameters

No parameters have been changed.

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 1.00 year(s).

The peak dose of 1.18E-004 TEDE (mrem) occurred 1.00 year(s) after
license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	1.05E-004	88.82
Inhalation	1.23E-005	10.42
Ingestion	9.00E-007	0.76
Total	1.18E-004	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
57Co	1.18E-004	100.00
Total	1.18E-004	100.00

```
Program      : DandD Version 1.0 Build 1.00.02
Session      : UVA Reactor Surfaces
Description :
     DCGLs for Building Surfaces at the University of Virginia
     Reactor

Executed     : 12/07/00 at 13:25:46
```

NRC Report

Occupancy Input Section

```
Execution Options
====================
History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities
========================================
Chain      dpm/100cm^2
========================================

65Zn              1.00

Code-Generated Radionuclide Activities
========================================
Chain      dpm/100cm^2
========================================

65Zn       1.0000E+000

Variable Parameters
======================================================================
No parameters have been changed.
```

Occupancy Output Section

Maximum Annual TEDE

```
This scenario started 0.00 year(s) from now
and ran for 1.00 year(s).

The peak dose of    5.20E-004 TEDE (mrem) occurred 1.00 year(s) after
license termination.
```

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	4.83E-004	92.87
Inhalation	2.66E-005	5.11
Ingestion	1.05E-005	2.02
Total	5.20E-004	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
65Zn	5.20E-004	100.00
Total	5.20E-004	100.00

```
Program      : DandD Version 1.0 Build 1.00.02
Session      : UVA Reactor Surfaces
Description :
     DCGLs for Building Surfaces at the University of Virginia
     Reactor

Executed     : 12/07/00 at 13:27:39
```

NRC Report

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	dpm/100cm^2
94Nb	1.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
94Nb	1.0000E+000

Variable Parameters

No parameters have been changed.

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 1.00 year(s).

The peak dose of   3.02E-003 TEDE (mrem) occurred 1.00 year(s) after
license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	2.14E-003	70.98
Inhalation	8.68E-004	28.74
Ingestion	8.35E-006	0.28
Total	3.02E-003	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
94Nb	3.02E-003	100.00
Total	3.02E-003	100.00

```
Program      : DandD Version 1.0 Build 1.00.02
Session      : UVA Reactor Surfaces
Description :
    DCGLs for Building Surfaces at the University of Virginia
    Reactor

Executed     : 12/07/00 at 13:29:23
```

NRC Report

Occupancy Input Section

```
Execution Options
====================
History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities
=========================================
Chain     dpm/100cm^2
=========================================

125Sb            1.00

Code-Generated Radionuclide Activities
=========================================
Chain     dpm/100cm^2
=========================================

125Sb     1.0000E+000
  125mTe    0.0000E+000

Variable Parameters
======================================================================
No parameters have been changed.
```

Occupancy Output Section

Maximum Annual TEDE

```
This scenario started 0.00 year(s) from now
and ran for 1.00 year(s).

The peak dose of    5.64E-004 TEDE (mrem) occurred 1.00 year(s) after
license termination.
```

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	5.35E-004	94.92
Inhalation	2.51E-005	4.44
Ingestion	3.59E-006	0.64
Total	5.64E-004	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
125Sb	5.53E-004	98.02
125mTe	1.12E-005	1.98
Total	5.64E-004	100.00

Program : DandD Version 1.0 Build 1.00.02
Session : UVA Reactor Surfaces
Description :
DCGLs for Building Surfaces at the University of Virginia Reactor

Executed : 12/07/00 at 13:31:20

NRC Report

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	dpm/100cm^2
152Eu	1.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
152Eu	1.0000E+000
152Gd	0.0000E+000

Variable Parameters

No parameters have been changed.

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 1.00 year(s).

The peak dose of 1.97E-003 TEDE (mrem) occurred 1.00 year(s) after license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	1.51E-003	76.70
Inhalation	4.51E-004	22.93
Ingestion	7.38E-006	0.38
Total	1.97E-003	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
152Eu	1.97E-003	100.00
152Gd	7.07E-018	0.00
Total	1.97E-003	100.00

```
Program      : DandD Version 1.0 Build 1.00.02
Session      : UVA Reactor Surfaces
Description  :
     DCGLs for Building Surfaces at the University of Virginia
     Reactor

Executed     : 12/07/00 at 13:33:19
```

NRC Report

Occupancy Input Section

```
Execution Options
====================
History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities
========================================
Chain     dpm/100cm^2
========================================

154Eu            1.00

Code-Generated Radionuclide Activities
========================================
Chain     dpm/100cm^2
========================================

154Eu     1.0000E+000

Variable Parameters
========================================================================
No parameters have been changed.
```

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 1.00 year(s).

The peak dose of 2.18E-003 TEDE (mrem) occurred 1.00 year(s) after
license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	1.59E-003	73.08
Inhalation	5.76E-004	26.43
Ingestion	1.07E-005	0.49
Total	2.18E-003	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
154Eu	2.18E-003	100.00
Total	2.18E-003	100.00

Program : DandD Version 1.0 Build 1.00.02
Session : UVA Reactor Surfaces
Description :
DCGLs for Building Surfaces at the University of Virginia Reactor

Executed : 12/07/00 at 13:57:17

NRC Report

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	dpm/100cm^2
210Pb	1.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
210Pb	1.0000E+000
210Bi	0.0000E+000
210Po	0.0000E+000

Variable Parameters

No parameters have been changed.

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 1.00 year(s).

The peak dose of 4.60E-002 TEDE (mrem) occurred 1.00 year(s) after license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	4.85E-006	0.01
Inhalation	3.86E-002	84.02
Ingestion	7.34E-003	15.97
Total	4.60E-002	100.00

Radionuclide Component of

Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
210Pb	3.42E-002	74.36
210Bi	4.04E-004	0.88
210Po	1.14E-002	24.76
Total	4.60E-002	100.00

```
Program      : DandD Version 1.0 Build 1.00.02
Session      : UVA Reactor Surfaces
Description :
     DCGLs for Building Surfaces at the University of Virginia
     Reactor

Executed     : 12/07/00 at 13:35:15
```

NRC Report

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.
Concentrations of radionuclides in equilibrium chains will be
   equilibrated based on initial concentration of parent.

Initial Radionuclide Activities

Chain	dpm/100cm^2
226Ra+C	1.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
226Ra	1.0000E+000
222Rn	1.0000E+000
218Po	1.0000E+000
214Pb	9.9981E-001
218At	2.0000E-004
214Bi	1.0000E+000
214Po	9.9981E-001
210Pb	1.0142E+000
210Bi	1.0142E+000
210Po	1.0144E+000

Variable Parameters

No parameters have been changed.

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 1.00 year(s).

The peak dose of   7.97E-002 TEDE (mrem) occurred 1.00 year(s) after
license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage

External	2.34E-003	2.93
Inhalation	6.72E-002	84.31
Ingestion	1.02E-002	12.76
Total	7.97E-002	100.00

Radionuclide Component of
Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
226Ra	1.95E-002	24.50
222Rn	5.54E-007	0.00
218Po	1.25E-008	0.00
214Pb	3.58E-004	0.45
218At	0.00E+000	0.00
214Bi	2.00E-003	2.50
214Po	1.14E-007	0.00
210Pb	3.52E-002	44.15
210Bi	4.25E-004	0.53
210Po	2.22E-002	27.87
Total	7.97E-002	100.00

```
Program     : DandD Version 1.0 Build 1.00.02
Session     : UVA Reactor Surfaces
Description :
     DCGLs for Building Surfaces at the University of Virginia
     Reactor

Executed    : 12/07/00 at 13:36:47
```

NRC Report

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	dpm/100cm^2
U_Nat	1.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
234U	4.8900E-001
230Th	0.0000E+000
226Ra	0.0000E+000
222Rn	0.0000E+000
218Po	0.0000E+000
214Pb	0.0000E+000
218At	0.0000E+000
214Bi	0.0000E+000
214Po	0.0000E+000
210Pb	0.0000E+000
210Bi	0.0000E+000
210Po	0.0000E+000
235U	2.2500E-002
231Th	0.0000E+000
231Pa	0.0000E+000
227Ac	0.0000E+000
223Fr	0.0000E+000
227Th	0.0000E+000
223Ra	0.0000E+000
219Rn	0.0000E+000
215Po	0.0000E+000
211Pb	0.0000E+000
211Bi	0.0000E+000
211Po	0.0000E+000
207Tl	0.0000E+000
238U	4.8900E-001
234Th	0.0000E+000
234mPa	0.0000E+000
234Pa	0.0000E+000
234U	0.0000E+000
230Th	0.0000E+000
226Ra	0.0000E+000
222Rn	0.0000E+000
218Po	0.0000E+000
214Pb	0.0000E+000
218At	0.0000E+000
214Bi	0.0000E+000
214Po	0.0000E+000

210Pb	0.0000E+000
210Bi	0.0000E+000
210Po	0.0000E+000

Variable Parameters

No parameters have been changed.

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now and ran for 1.00 year(s).

The peak dose of 2.63E-001 TEDE (mrem) occurred 1.00 year(s) after license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	2.31E-005	0.01
Inhalation	2.63E-001	99.87
Ingestion	3.22E-004	0.12
Total	2.63E-001	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
234U	1.36E-001	51.63
230Th	1.50E-006	0.00
226Ra	6.22E-012	0.00
222Rn	1.69E-016	0.00
218Po	3.79E-018	0.00
214Pb	1.09E-013	0.00
218At	0.00E+000	0.00
214Bi	6.07E-013	0.00
214Po	3.47E-017	0.00
210Pb	8.04E-014	0.00
210Bi	8.97E-016	0.00
210Po	1.28E-014	0.00
235U	5.80E-003	2.20
231Th	6.59E-007	0.00
231Pa	6.36E-007	0.00
227Ac	3.47E-008	0.00
223Fr	3.49E-015	0.00
227Th	6.70E-011	0.00
223Ra	3.02E-011	0.00
219Rn	1.34E-013	0.00
215Po	4.27E-016	0.00
211Pb	1.57E-013	0.00
211Bi	1.12E-013	0.00
211Po	5.20E-017	0.00
207Tl	9.16E-015	0.00
238U	1.21E-001	46.15
234Th	4.47E-005	0.02
234mPa	9.47E-006	0.00
234Pa	2.29E-006	0.00

Total	2.63E-001	100.00

Program   : DandD Version 1.0 Build 1.00.02
Session   : UVA Reactor Surfaces
Description :
    DCGLs for Building Surfaces at the University of Virginia
    Reactor

Executed   : 12/07/00 at 13:38:15

NRC Report

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.
Concentrations of radionuclides in equilibrium chains will be
equilibrated based on initial concentration of parent.

Initial Radionuclide Activities

Chain	dpm/100cm^2
233U+C	1.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
233U	1.0000E+000
229Th	1.0485E+000
225Ra	1.0485E+000
225Ac	1.0485E+000
221Fr	1.0485E+000
217At	1.0485E+000
213Bi	1.0485E+000
213Po	1.0259E+000
209Tl	2.2648E-002
209Pb	1.0485E+000

Variable Parameters

No parameters have been changed.

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 1.00 year(s).

The peak dose of 5.04E+000 TEDE (mrem) occurred 1.00 year(s) after
license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage

External	4.69E-004	0.01
Inhalation	5.04E+000	99.89
Ingestion	5.28E-003	0.10
Total	5.04E+000	100.00

Radionuclide Component of
Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
233U	2.84E-001	5.63
229Th	4.72E+000	93.54
225Ra	1.76E-002	0.35
225Ac	2.39E-002	0.47
221Fr	4.38E-005	0.00
217At	4.44E-007	0.00
213Bi	2.33E-004	0.00
213Po	0.00E+000	0.00
209Tl	6.03E-005	0.00
209Pb	9.12E-007	0.00
Total	5.04E+000	100.00

Program : DandD Version 1.0 Build 1.00.02
Session : UVA Reactor Surfaces
Description :
DCGLs for Building Surfaces at the University of Virginia Reactor

Executed : 12/07/00 at 13:39:38

NRC Report

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	dpm/100cm^2
234U	1.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
234U	1.0000E+000
230Th	0.0000E+000
226Ra	0.0000E+000
222Rn	0.0000E+000
218Po	0.0000E+000
214Pb	0.0000E+000
218At	0.0000E+000
214Bi	0.0000E+000
214Po	0.0000E+000
210Pb	0.0000E+000
210Bi	0.0000E+000
210Po	0.0000E+000

Variable Parameters

No parameters have been changed.

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 1.00 year(s).

The peak dose of 2.78E-001 TEDE (mrem) occurred 1.00 year(s) after license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage

External	1.05E-006	0.00
Inhalation	2.77E-001	99.88
Ingestion	3.32E-004	0.12
Total	2.78E-001	100.00

Radionuclide Component of
Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
234U	2.78E-001	100.00
230Th	3.08E-006	0.00
226Ra	1.27E-011	0.00
222Rn	3.45E-016	0.00
218Po	7.75E-018	0.00
214Pb	2.23E-013	0.00
218At	0.00E+000	0.00
214Bi	1.24E-012	0.00
214Po	7.09E-017	0.00
210Pb	1.64E-013	0.00
210Bi	1.83E-015	0.00
210Po	2.61E-014	0.00
Total	2.78E-001	100.00

Program : DandD Version 1.0 Build 1.00.02
Session : UVA Reactor Surfaces
Description :
DCGLs for Building Surfaces at the University of Virginia Reactor

Executed : 12/07/00 at 13:41:16

NRC Report

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.
Concentrations of radionuclides in equilibrium chains will be equilibrated based on initial concentration of parent.

Initial Radionuclide Activities

Chain	dpm/100cm^2
238U+C	1.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
238U	1.0000E+000
234Th	1.0000E+000
234mPa	9.9800E-001
234Pa	2.0000E-003
234U	1.0001E+000
230Th	1.0001E+000
226Ra	1.0001E+000
222Rn	1.0001E+000
218Po	1.0001E+000
214Pb	9.9987E-001
218At	2.0001E-004
214Bi	1.0001E+000
214Po	9.9987E-001
210Pb	1.0001E+000
210Bi	1.0001E+000
210Po	1.0001E+000

Variable Parameters

No parameters have been changed.

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 1.00 year(s).

The peak dose of 1.29E+000 TEDE (mrem) occurred 1.00 year(s) after license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	2.38E-003	0.18
Inhalation	1.27E+000	98.93
Ingestion	1.13E-002	0.88
Total	1.29E+000	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
238U	2.48E-001	19.28
234Th	1.01E-004	0.01
234mPa	2.14E-005	0.00
234Pa	5.17E-006	0.00
234U	2.78E-001	21.57
230Th	6.83E-001	53.01
226Ra	1.95E-002	1.52
222Rn	5.54E-007	0.00
218Po	1.25E-008	0.00
214Pb	3.58E-004	0.03
218At	0.00E+000	0.00
214Bi	2.00E-003	0.15
214Po	1.14E-007	0.00
210Pb	3.47E-002	2.70
210Bi	4.19E-004	0.03
210Po	2.19E-002	1.70
Total	1.29E+000	100.00

```
Program      : DandD Version 1.0 Build 1.00.02
Session      : UVA Reactor Surfaces
Description :
     DCGLs for Building Surfaces at the University of Virginia
     Reactor

Executed     : 12/07/00 at 13:42:54
```

NRC Report

Occupancy Input Section

Execution Options

```
===================
History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.
```

Initial Radionuclide Activities

```
=======================================
Chain     dpm/100cm^2
=======================================

241Pu            1.00
```

Code-Generated Radionuclide Activities

```
=======================================
Chain     dpm/100cm^2
=======================================

241Pu     1.0000E+000
  241Am     0.0000E+000
  237U      0.0000E+000
  237Np     0.0000E+000
  233Pa     0.0000E+000
  233U      0.0000E+000
  229Th     0.0000E+000
  225Ra     0.0000E+000
  225Ac     0.0000E+000
  221Fr     0.0000E+000
  217At     0.0000E+000
  213Bi     0.0000E+000
  213Po     0.0000E+000
  209Tl     0.0000E+000
  209Pb     0.0000E+000
```

Variable Parameters

```
======================================================================
No parameters have been changed.
```

Occupancy Output Section

Maximum Annual TEDE

```
=============================
This scenario started 0.00 year(s) from now
and ran for 1.00 year(s).

The peak dose of    1.77E-002 TEDE (mrem) occurred 1.00 year(s) after
license termination.
```

Pathway Component of
Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	3.74E-008	0.00
Inhalation	1.76E-002	99.54
Ingestion	8.15E-005	0.46
Total	1.77E-002	100.00

Radionuclide Component of
Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
241Pu	1.70E-002	95.84
241Am	7.36E-004	4.16
237U	4.70E-009	0.00
237Np	1.01E-010	0.00
233Pa	1.98E-014	0.00
233U	1.93E-017	0.00
229Th	1.14E-020	0.00
225Ra	3.85E-023	0.00
225Ac	3.61E-023	0.00
221Fr	6.61E-026	0.00
217At	6.71E-028	0.00
213Bi	3.51E-025	0.00
213Po	0.00E+000	0.00
209Tl	9.11E-026	0.00
209Pb	1.38E-027	0.00
Total	1.77E-002	100.00

```
Program      : DandD Version 1.0 Build 1.00.02
Session      : UVA Reactor Surfaces
Description :
     DCGLs for Building Surfaces at the University of Virginia
     Reactor

Executed     : 12/07/00 at 13:44:14
```

NRC Report

Occupancy Input Section

Execution Options
====================
History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	dpm/100cm^2
241Am	1.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
241Am	1.0000E+000
237Np	0.0000E+000
233Pa	0.0000E+000
233U	0.0000E+000
229Th	0.0000E+000
225Ra	0.0000E+000
225Ac	0.0000E+000
221Fr	0.0000E+000
217At	0.0000E+000
213Bi	0.0000E+000
213Po	0.0000E+000
209Tl	0.0000E+000
209Pb	0.0000E+000

Variable Parameters
==
No parameters have been changed.

Occupancy Output Section

Maximum Annual TEDE
==============================

This scenario started 0.00 year(s) from now
and ran for 1.00 year(s).

The peak dose of   9.34E-001 TEDE (mrem) occurred 1.00 year(s) after
license termination.

Pathway Component of
Maximum Annual Dose
==============================

Pathway	TEDE (mrem)	Percentage
External	3.85E-005	0.00
Inhalation	9.29E-001	99.54
Ingestion	4.26E-003	0.46
Total	9.34E-001	100.00

Radionuclide Component of
Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
241Am	9.34E-001	100.00
237Np	1.84E-007	0.00
233Pa	3.91E-011	0.00
233U	4.96E-014	0.00
229Th	1.70E-017	0.00
225Ra	5.03E-020	0.00
225Ac	5.75E-020	0.00
221Fr	1.05E-022	0.00
217At	1.07E-024	0.00
213Bi	5.60E-022	0.00
213Po	0.00E+000	0.00
209Tl	1.45E-022	0.00
209Pb	2.20E-024	0.00
Total	9.34E-001	100.00

APPENDIX 6.3

Cost Parameters: $Cost_T$

Decontamination Cost Parameters: $Cost_T$

- Unit waste disposal cost: $2,860 per cubic meter
- Burdened Cost Multiplier 2.97
- Area to be decontaminated: 100 square meters

METHODOLOGY	TOTAL PROCESS COST (\$/m^2)	BURDENED COST (\$/m^2)	PRIMARY WASTE VOLUME (m^3/100m^2)	SECONDARY WASTE VOLUME (m^3/m^2)	WASTE DISPOSAL COST (\$/100m^2)	$Cost_T$ (\$/100m^2)
1/4" Scabbling of Floor Surfaces	$7.44	$22.09	0.952	0	$2,723	$4,931
1/8" Scabbling of Wall Surfaces	$32.57	$96.74	0.476	0	$1,362	$11,033
Manual Decontamination All Surfaces	$16.88	$50.12	0	0.254	$726	$5,738

Excavation Cost Parameters: $Cost_T$

- Area to be remediated: 100 square meters
- Depth of remediation: 15 centimeters (0.15 meters)
- Volume of soil removed: 15 cubic meters (530 cubic feet)
- Unit waste disposal cost: $2,860 per cubic meter
- Total Disposal cost: $42,900

There are many additional costs that could be added to the disposal charge to arrive at the $Cost_T$ value. These include the cost to excavate the material, the burdened cost multiplier, etc. These costs were not added because they are not necessary to demonstrate that the soil C_{ALARA} values are larger than the dose limit $DCGL_W$ values.

DURATEK

Radiological Engineering
And
Field Services

REFS-CALC-UVAR-002

REVISION 0

RADIOLOGICAL ACCIDENT ANALYSIS FOR UVAR DECOMMISSIONING PLAN

PREPARED BY:	Paul C Ely Document Preparer	12.12.00 Date
CHECKED BY:	John McBride Verifier	12-13-00 Date
APPROVED BY:	Robert Wangland Manager	12/13/00 Date

AFFECTED PAGES: All

1. PURPOSE AND OBJECTIVES

This calculation provides an analysis of the potential radiological accidents that could occur during decommissioning of the University of Virginia Reactor (UVAR) Facility and affect the public or occupational health and safety. The accident analyses show that the doses to the public from potential accidents are below the U.S. EPA Protective Action Guides (PAGs) that have been developed to protect members of the public from the consequences of accidents (EPA 400-R-92-001, 1992). Also, doses to workers from potential accidents are below the permitted annual exposure limits. Therefore, no new protective measures are required to protect public or occupational health and safety. A screening analysis of potential accidents at a level of detail consistent with existing information about the radiological hazards at the UVAR Facility was performed.

2. CONCLUSIONS

The accident analysis shows that the postulated accident scenarios would result in TEDEs to a member of the public at the site boundary that are much less than the U.S. EPA's lower PAG of 1 rem (1000 mrem) (USEPA 1992). Also, doses that workers could receive from an accident are much less than the allowable annual exposure for workers, 5 rem (5,000 mrem) (NRC 1991).

The results of the accident analysis show that offsite consequences from accidents are well below the U.S. EPA's PAGs; therefore, offsite emergency plans are not needed.

3. ASSUMPTIONS/INPUTS

3.1 Assumptions

The following assumptions were used in all of the accident analyses:

- The radionuclide inventories were based on the data provided in the 2000 characterization of the site by GTS Duratek.

- To be conservative, unfavorable weather conditions for atmospheric dispersion were assumed. For the purposes of this analysis, atmospheric stability class F with a wind speed of 2 m/s (6.6 ft/s) was assumed, which represents a situation with minimal dispersion of a potential radioactive plume. In addition, the radioactive material was assumed to be released at ground level and to remain airborne as it travels downwind.

- A screening analysis approach was used for the UVAR Facility accident analysis because the radioactive inventories are very small compared to those in operating reactors and in fuel cycle facilities subject to NRC regulation.
- The screening analysis for the UVAR Facility consists of identifying and analyzing plausible accident scenarios that could occur during decommissioning activities.

3.2 Potential Radiological Accidents

Identifying potential accident scenarios included evaluating UVAR Facility areas that contain the highest inventories of radioactive material, describing energy sources and external events, reviewing proposed activities, and considering combinations of these elements that could lead to a release of radioactive material. Because of the limited inventory, the evaluation of accident scenarios conservatively assumed that no design or procedural controls would be available to prevent or mitigate accidental releases, even though such controls will be implemented during decommissioning activities. This assumption allows for a worst-case accident analysis to be performed.

3.3 Highest Radionuclide Inventories at the UVAR Facility

GTS Duratek estimated radiological inventories for the UVAR Facility based on the 2000 characterization of the site. One area with a high radiological material inventory at the UVAR Facility was the reactor pool in the reactor room. Most of the activity in the pool is contained in activated components that include control rods, instrument thimbles, etc. During decommissioning, these components will be cut mechanically underwater and lifted and placed into transport liners underwater. A waste shipping liner containing activated hardware was estimated to contain 1,460 curies (NUREG/CR-1756-v1, Tables E1-2, E.1-3, E.1-5 and E.1-6, for Reference Research Reactor). Most of the activity is Co-60 (84%), with Fe-55 (8.6%), Mn-54 (4.7%), and smaller inventories of other radionuclides. These are simple operations, and the worst-case accident scenario would be dropping one of these filled transport liners as it is being lifted.

The reactor room is another area considered for a potential accident. The Reactor Room is relatively large area and assuming the entire area is contaminated with a worst-case hot spot of 12,593 dpm/100cm2, an inventory of 0.362 millicuries was estimated. Most of the inventory is Co-60 (41%), with smaller inventories of other radionuclides Ni-63 (21%), Mn-54 (20%), Zn-65 (10%), and Fe-55 (8%).

All of the other rooms at the UVAR Facility have smaller radioactive inventories. Therefore, the results of accident analyses conducted for decommissioning the reactor pool and the Reactor Room bound the potential impacts of inside accidents during decommissioning of the UVAR Facility.

Accidents could also occur outside buildings in soil at areas of past environmental contamination. These areas include the buried waste tank area, the buried hot cell tank area and the pond. This buried waste tanks have the highest concentration of radionuclides, it is estimated to contain 353 Ci of Co-60, 21.6 Ci of Cs-137, 21.2 Ci of Sb-125, 13.9 Ci of Zn-65, 8.5 Ci of Mn-54, 5.2 Ci of Eu-154 as well as smaller inventories of other radionuclides. The soils around the buried waste tanks have the highest levels of soil contamination; a 55-gallon drum of this soil is estimated to contain 2.73 Ci of Co-60 and 2.08 Ci of Cs-137. Therefore, accident analyses were performed for these items because they bound the potential impacts of exterior accidents at the UVAR Facility.

4. REFERENCES

4.1 U.S. EPA, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*, 400-R-92-001, 1992.

4.2 10CFRPart 20, *Standards for Protection Against Radiation*, Federal Register, Vol. 56, No. 88, NRC May 21, 1991

4.3 GTS Duratek, *2000. Characterization Survey Report for the University of Virginia Reactor Facility.*

4.4 *Technology, Safety and Costs of Decommissioning Reference Nuclear Research and Test Reactors*, NUREG/CR-1756, March 1982

4.5 International Commission on Radiological Protection (ICRP), *Limits for Intakes of Radionuclides by Workers*, 1979, ICRP Publication No. 30, Annals of the ICRP Vol. 2, No. 3/4

4.6 *Federal Guidance Report No. 11*0, EPA-520/1-88-020, September 1988

4.7 *Federal Guidance Report No. 12*, EPA-402-R-93-081, September 1993.

4.8 GTS Duratek Calculation REFS-CALC-UVAR-001, Rev. 0. *Dose Assessment For UVAR Decommissioning Plan.*

5. POTENTAIAL ACCIDENT SCENARIO'S

Considering the planned decommissioning activities, accident scenarios that could result in releasing radioactive material as airborne particles small enough to be respirable were evaluated. Such releases could occur during structural demolition operations, a fire during demolition operations, or dropping of a container of radioactively contaminated dust or soil. Because all UVAR Facility buildings are outside of the 500-year floodplain and releases from soil areas near the river will be minimal based on historical erosion information, extreme precipitation events are not expected to cause offsite radiological impacts. The potential onsite and offsite impacts of accidents will be mitigated by emergency procedures required by UVAR Facility technical specifications.

Based on the decommissioning activities outlined in the UVAR Decommissioning Plan and the radiological inventories identified in Section 3.3, the following accident scenarios were evaluated:

- A waste shipping liner containing activated hardware could be dropped while moving it from the pool to a transportation cask.
- The potential for fires was also considered. The reactor room area has combustible material that includes wood framing and asphalt roofing.
- Areas of environmental contamination outside of buildings, such as the buried waste tanks, will be decontaminated by removing the tanks and digging up contaminated soil and placing it into containers. The dropping of a waste tank that then bursts could produce airborne particles.
- The dropping of a container of excavated soil that then bursts could produce airborne particles.

6. Methodology for Calculating Total Effective Dose Equivalent

The consequences of accidents were quantified by calculating the TEDE to a member of the public at the site boundary. Then the calculated TEDE was compared to the U.S. EPA's lower PAG of 1 rem (1,000 mrem), to determine whether or not the calculated exposure is acceptable. Equation 6-1 was used to calculate the TEDE[2]:

$$TEDE_i = CEDE_i + Ext_i \tag{6-1}$$

[2] This estimate of the TEDE neglects any contribution from gamma rays emitted by radionuclides deposited on the ground. Such doses build up relatively slowly and, if necessary, can be controlled by various countermeasures.

Where

TEDE = total effective dose equivalent
CEDE = committed effective dose equivalent
Ext = contribution from external irradiation
i = radionuclide.

The committed effective dose equivalent (CEDE) is the dose contribution from inhalation as the cloud passes by the receptor. Consistent with the lung model developed by the International Commission on Radiological Protection (ICRP 1979), the CEDE is found by

$$CEDE_i = Q_i\,(\chi/Q) \times B \times D_i \tag{6-2}$$

Where

Q_i = the total released activity of nuclide i, in Ci

χ/Q = the airborne dosage (concentration integrated over the duration of cloud passage) per unit activity released, in s/m³. The derivation of χ/Q presented in Appendix B shows that for a distance of 100 meters (0.06 mi) in atmospheric stability class F with a wind speed of 2 m/s, $\chi/Q = 4.52 \times 10^{-3}$ s/m^3.

B = the breathing rate, typically 3.3×10^{-4} m^3/s. (This is the breathing rate for adults during light activity [ICRP 1979]).

D_i = the factor that converts the amount of activity inhaled into the CEDE. Values of D_i are given in Federal Guidance Report No. 11 (EPA-520/1-88-020, September 1988).

The dose contribution from external irradiation is found by

$$Ext_i = Q_i\ \chi/Q)\ F_i \qquad (6\text{-}3)$$

Where

F_i = the dose coefficient for air submersion. Values of F_i are given in Federal Guidance Report No. 12 (EPA-402-R-93-081, September 1993).

7. ANALYSIS OF POTENTAIL ACCIDENT SCENARIO'S

7.1 Scenario 1: Release While Moving Irradiated Hardware Liner

Most of the activity in the pool is contained in activated components that include control rods, instrument thimbles, etc. Because cutting operations for components will be performed underwater, no cutting accident releases were postulated. However a liner filled with irradiated hardware could be dropped while it is lifted for placement into a shipping container. The activity in irradiated hardware is contained within the metal structure of the hardware item except for surface contamination. It would be highly unlikely for a component in the liner to break. If it did break, the diameters of any particles produced would be large enough that it is unlikely that the particles would remain airborne and be respirable. However, even though it is not plausible that an accident could result in measurable exposures at the site boundary this scenario was evaluated because it includes the largest curie inventory and it demonstrates that potential exposures to the public are acceptable even when worst case assumptions are utilized.

A waste shipping liner containing 120 cubic feet of activated hardware was estimated to contain 1,460 curies (NUREG/CR-1756-v1, Tables E1-2, E.1-3, E.1-5 and E.1-6, for Reference Research Reactor). Most of the activity is Co-60 (84%), with Fe-55 (8.6%), Mn-54 (4.7%), and smaller inventories of other radionuclides. The worst-case accident scenario would be dropping the filled liner as it is being lifted. If 1 percent of the activity of this liner was in a **respirable form material** and 1 percent of the respirable material escaped the liner and became airborne during the accident (i.e., approximately 6.8×10^2 g, assuming a waste density of 2.0 g/cm^3), the airborne quantities of radionuclides would be 1.22×10^{-1} Ci of Co-60, 1.25×10^{-2} Ci of Fe-55, 6.83×10^{-3} Ci of Mn-54, and lesser quantities of other radionuclides. Using the values of χ/Q and B given in Appendix 9 in Equations 6-1 through 6-3, the CEDE, the contribution from external irradiation (Ext), and the TEDE were calculated as shown in Table 7-1.

Table 7-1. TEDE Calculation Table for Scenario 1: Hardware Liner Drop

Nuclide i	Q_I (Ci)	D_I (mrem/Ci)	F_i ([mrem/s]/ [Ci/m^3])	CEDE$_i$ (mrem)	Ext$_i$ (mrem)	TEDE$_i$ (mrem)
C-14	3.90E-06	2.09E+06	0.0008	1.21E-05	1.46E-11	1.21E-05
Mn-54	6.83E-03	2.19E+08	466.6662	2.23E+00	1.44E-02	2.24E+00
Fe-55	1.25E-02	1.34E+06	0.0000	2.48E-02	0.00E+00	2.48E-02
Co-60	1.22E-01	2.19E+08	466.6662	3.97E+01	2.57E-01	4.00E+01
Ni-59	2.36E-05	9.18E+05	0.0000	3.24E-05	0.00E+00	3.24E-05
Ni-63	2.71E-03	2.30E+06	0.0000	9.29E-03	0.00E+00	9.29E-03
Zn-65	1.82E-03	2.04E+07	103.3332	5.55E-02	8.52E-04	5.63E-02
Nb-93m	4.31E-09	3.21E+06	0.0164	2.07E-08	3.21E-13	2.07E-08
Nb-94	5.58E-08	3.61E+07	285.1849	3.01E-06	7.20E-08	3.08E-06
Total	**1.46E-01**	–	--	**4.20E+01**	**2.72E-01**	**4.23E+01**

As shown in Table 7-1, the TEDE is less than 43 mrem, to which the external dose is a negligible contributor. The TEDE of less than 43 mrem derived using conservative conditions is less than the U.S. EPA's lower PAG of 1 rem (1,000 mrem) (USEPA 1992).

7.2 Scenario 2: Release from a Reactor Room Structural Fire

The Reactor Room has one area where there was a significant hot spot. Assuming the entire area is contaminated with the worst-case hot spot activity of 12,593 dpm/100cm2, an inventory of 0.362 curies was estimated. Most of the inventory is Co-60 (41%), with smaller inventories of other radionuclides Ni-63 (21%), Mn-54 (20%), Zn-65 (10%), and Fe-55 (8%). It is estimated that combustion of this structural material would release approximately 10% of the contamination. Thus, combustion of the entire reactor room area would release 3.62×10^{-5} curies of respirable material into the air. Using the values of χ/Q and B given in Appendix 9 in Equations 6-1 through 6-3, the CEDE, the contribution from external irradiation (Ext), and the TEDE were calculated as shown in Table 7-2.

Table 7-2. TEDE Calculation Table for Scenario 2: Reactor Room Structural Fire

Nuclide i	Q_I (Ci)	D_I (mrem/Ci)	F_i ([mrem/s]/ [Ci/m^3])	CEDE$_i$ (mrem)	Ext$_i$ (mrem)	TEDE$_i$ (mrem)
Mn-54	7.35E-06	6.70E+06	151.4813	7.34E-05	5.03E-06	7.84E-05
Fe-55	2.83E-06	1.34E+06	0.000	5.64E-06	0.00E+00	5.64E-06
Co-60	1.49E-05	2.19E+08	466.6662	4.85E-03	3.14E-05	4.88E-03
Ni-63	7.68E-06	2.30E+06	0.0000	2.63E-05	0.00E+00	2.63E-05
Zn-65	3.49E-06	2.04E+07	103.3332	1.06E-04	1.63E-06	1.08E-04
Total	**3.62E-05**	--	--	**5.06E-03**	**3.80E-05**	**5.10E-03**

As shown in Table 7-2, the TEDE is less than 1 mrem, to which the external dose is a negligible contributor. The TEDE of less than 1 mrem derived using conservative conditions is less than the U.S. EPA's lower PAG of 1 rem (1,000 mrem) (USEPA 1992).

7.3 Scenario 3: Release While Removing Buried Waste Tank

The east buried waste tanks has the highest concentration of radionuclides in an outdoor area. The east tank is conservatively estimated to contain 100 gallons or less of contaminated sediment. The radionuclide inventory of the east tank would then be 353 Ci of Co-60, 21.6 Ci of Cs-137, 21.2 Ci of Sb-125, 13.9 Ci of Zn-65, 8.5 Ci of Mn-54, 5.2 Ci of Eu-154 as well as smaller inventories of other radionuclides. An accident analyses was performed for the east waste tank because it bounds the potential impacts of exterior accidents at the UVAR Facility.

The worst-case accident scenario would be dropping the tank as it is being lifted. If 50 percent of the activity of this liner was in a **respirable form material** and 20 percent of the respirable material became airborne during the accident (i.e., approximately 4.54×10^4 g, assuming a waste density of 1.2 g/cm^3), the airborne quantities of radionuclides would be 3.53×10^{-5} Ci of Co-60, 2.16×10^{-6} Ci of Cs-137, 2.12×10^{-6} Ci of Sb-125, 1.39×10^{-6} Ci of Zn-65, 8.45×10^{-7} Ci of Mn-54, 5.22×10^{-7} Ci of Eu-154, and lesser quantities of other radionuclides. Using the values of χ/Q and B given in Appendix 9 in Equations 6-1 through 6-3, the CEDE, the contribution from external irradiation (Ext), and the TEDE were calculated as shown in Table 7-3.

Table 7-3. TEDE Calculation Table for Scenario 3: Waste Tank Drop

Nuclide i	Q_I (Ci)	D_I (mrem/Ci)	F_i ([mrem/s]/ [Ci/m^3])	CEDE$_i$ (mrem)	Ext$_i$ (mrem)	TEDE$_i$ (mrem)
Mn-54	8.45E-07	2.19E+08	466.6662	2.76E-04	1.78E-06	2.77E-04
Co-57	3.86E-07	3.19E+07	0.0287	1.84E-05	5.00E-11	1.84E-05
Co-60	3.53E-05	2.19E+08	466.6662	1.15E-02	7.45E-05	1.16E-02
Zn-65	1.39E-06	2.04E+07	103.3332	4.24E-05	6.51E-07	4.31E-05
Sb-125	2.12E-06	1.22E+07	74.8147	3.86E-05	7.17E-07	3.93E-05
Cs-137	2.16E-06	3.19E+07	0.0287	1.03E-04	2.80E-10	1.03E-04
Eu-154	5.22E-07	3.19E+07	0.0287	2.49E-05	6.77E-11	2.49E-05
Total	**4.27E-05**	--	--	**1.20E-02**	**7.76E-05**	**1.21E-02**

As shown in Table 7-3, the TEDE is less than 1 mrem, to which the external dose is a negligible contributor. The TEDE of less than 1 mrem derived using conservative conditions is less than the U.S. EPA's lower PAG of 1 rem (1,000 mrem) (USEPA 1992).

7.4 Scenario 4: Release While Removing Contaminated Soil

The contaminated soil area of the UVAR Facility having the highest concentration and inventory of radionuclides is the area around buried tanks. The estimated concentrations of radionuclides are 6.4 pCi/g Cs-137, and 8.4 pCi/g Co-60. If 20 percent of a 55-gal drum of this contaminated soil became airborne **in respirable form** (i.e., approximately 6.5×10^4 g, assuming a soil density of 1.56 g/cm^3), the airborne quantities of respirable radionuclides would be 5.46×10^{-7} Ci of Co-60, and 4.16×10^{-7} Ci of Cs-137. Using the values of χ/Q and B given in Appendix 9 in Equations 6-1 through 6-3, the CEDE, the contribution from external irradiation (Ext), and the TEDE were calculated as shown in Table 7-4.

Table 7-4. TEDE Calculation Table for Scenario 1: Contaminated Soil Released

Nuclide i	Q_I (Ci)	D_I (mrem/Ci)	F_i ([mrem/s]/ [Ci/m^3])	CEDE$_i$ (mrem)	Ext$_i$ (mrem)	TEDE$_i$ (mrem)
Co-60	5.46E-07	2.19E+08	466.6662	1.78E-04	1.15E-06	1.79E-04
Cs-137	4.16E-07	3.19E+07	0.0287	1.98E-05	5.39E-11	1.98E-05
Total	**9.61E-07**	--	--	**1.98E-04**	**1.15E-06**	**1.99E-04**

As shown in Table 7-4, the TEDE is less than 1 mrem, to which the external dose is a negligible contributor. The TEDE of less than 1 mrem derived using conservative conditions is less than the U.S. EPA's lower PAG of 1 rem (1,000 mrem) (USEPA 1992).

8. WORKER EXPOSURE ANALYSIS

8.1 Evaluation of Worker Exposure from Accident Scenarios

The accident scenario that would have the greatest potential release during decommissioning is Scenario 1 (Section 7.1), Release While Moving Irradiated hardware Liner. This accident could result in 1.22×10^{-1} Ci of Co-60, 1.25×10^{-2} Ci of Fe-55, and 6.83×10^{-3} Ci of Mn-54, and lesser quantities of other radionuclides becoming airborne. The movement of a loaded liner would likely be performed using the overhead polar crane with restricted access to the area during the liner movement. The engineering details of the hardware removal operations have not been finalized, but it is assumed that engineering controls such as spraying surfaces with demineralized water and using transfer shields would greatly reduce the potential for generating airborne material during these operations. Conservatively, in case the above operations, if precautions do not protect the worker from exposure, it was assumed that a worker inhales a fraction (i.e., 1×10^{-6} [one millionth]), of the radioactive material released following the demolition accident. If 1.22×10^{-1} Ci of Co-60, 1.25×10^{-2} Ci of Fe-55, and 6.83×10^{-3} Ci of Mn-54 become airborne, the worker would inhale 1.22×10^{-7} Ci of Co-60, 1.25×10^{-8} Ci of Fe-55, and 6.83×10^{-9} Ci of Mn-54. Using the dose conversion factors, D_i, for Co-60 of 2.19×10^{8} mrem/Ci, Fe-55 of 1.34×10^{6}, and Mn-54 of 2.19×10^{8} mrem/Ci, the dose to the worker would be $(1.22 \times 10^{-7}$ Ci$)$ $(2.19 \times 10^{8}$ mrem/Ci$)$ + $(1.25 \times 10^{-8}$ Ci$)$ $(1.34 \times 10^{6}$ mrem/Ci$)$ + $(6.83 \times 10^{-9}$ Ci$)$ $(2.19 \times 10^{8}$ mrem/Ci$)$ = 28.2 mrem. This dose is well below the allowable annual exposure for workers, 5 rem (5,000 mrem) (NRC 1991).

The accidents discussed in Sections 7.2 through 7.4 would result in even less severe consequences than the 28.2 mrem calculated above. Therefore, it is unlikely that an accident could occur where a worker would accumulate a significant fraction of the 5-rem annual exposure limit. As described in the UVAR Decommissioning Plan the radiation protection program will include worker protection and approved work control permits and procedures.

8.2 Conclusions

The accident analysis shows that the postulated accident scenarios would result in TEDE's to a member of the public at the site boundary that are much less than the U.S. EPA's lower PAG of 1 rem (1,000 mrem) (USEPA 1992). Also, doses that workers could receive from an accident are much less than the allowable annual exposure for workers, 5 rem (5,000 mrem) (NRC 1991).

The results of the accident analysis show that offsite consequences from accidents are well below the U.S. EPA's PAGs; therefore, offsite emergency plans are not needed.

9. APPENDICES

DERIVATION OF /Q

APPENDIX 9

DERIVATION OF /Q

In the accident analysis presented in Section 6.0, the quantity χ/Q is used to express the dilution of the released effluent as it travels 100 meters (0.06 mi) to the site boundary. χ/Q is calculated using the well-established formula for Gaussian Dispersion, which is applicable when the effluent is released at such a rate that it does not perturb the existing pattern of turbulent eddies in the atmosphere. This is the expected case for small releases such as are evaluated in Section 7.0. χ/Q was calculated using the formula:

$$\frac{\chi}{Q} = \frac{1}{U_{10}\pi\Sigma_y\sigma_z} \qquad (9\text{-}1)$$

where $\Sigma_y = M\sigma_y$, for distances of 800 meters or less and the value of M is determined from Figure 3 of NRC Regulatory Guide 1.145 (M=1 for all cases where the wind speed is 6 meters per second or more). For a wind speed of 2 m/s and atmospheric stability class F, M=4.

Then for a wind speed of 2 m/s:

$$\frac{\chi}{Q} = \frac{1}{U_{10}\pi M\sigma_y\sigma_z} \qquad (9\text{-}2)$$

where:

σ_y = the lateral plume spread in meters (m)
σ_z = the vertical plume spread in meters (m)
U_{10} = the wind speed (meters/second) measured at a height of 10 meters.

The value of σ_y is determined from Figure 1 of NRC Regulatory Guide 1.145 and σ_z from Figure 2 of NRC Regulatory Guide 1.145. The values are a function of the distance, d, from the source and Pasquill's turbulence types. For the UVAR Facility a distance of 100 meters (328 feet) and category F (Moderately Stable) meteorological conditions were utilized. Using d = 100 meters for category F meteorological conditions yielded σ_y = 4.0 meters and σ_z = 2.2 meters. Using these values of σ_z and σ_y and a wind speed, u, of 2 m/s, Equation 9-2 yields

$$\frac{\chi}{Q} = \frac{1}{(2)\pi(4)(4.0m)(2.2m)} = 4.52\times10^{-3}\ \text{sec}/m^3$$