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United States Government

Department of Energy

memorandum

DATE: November 17, 1995

REPLY TO: Air, Water and Radiation Division: EH-412: Wallo:2025864996
ATTN OF:

SUBJECT: Application of DOE 5400.5 requirements for release and control of property containing residual radioactive material.

TO: Distribution

Field and program offices have requested additional clarification on several issues that relate to Order DOE 5400.5 requirements for control of residual radioactive material (Section II.5 and Chapter IV). The issues in question have to varying degrees been clarified in the proposed 10 CFR Part 834 ("Radiation Protection of the Public and Environment") and in an implementation guide on residual radioactive material being prepared for that rule. However, due to the delay in the final promulgation of that Part 834, the Office of Environment is providing the attached guidance which clarifies the issues identified. The attached interim guidance:

- 1) Describes the relationship of DOE standards for release of property containing residual radioactive material to Nuclear Regulatory Commission and State requirements for control of such material.
- 2) Provides an update on EH expectations about the use of surface contamination guidelines from DOE 5400.5 and associated guidance reports for release of property. Particular concern was expressed in regard to the limits for transuranics.
- 3) Provides guidance and clarification about requirements for tritium in property being released or reused.

If you have questions regarding the attached material please contact Mr. Andrew Wallo, EH-412 (202-586-4996, email "andrew.wallo@hq.doe.gov").

for Raymond F. Pelletier
Director
Office of Environmental Policy and Assistance

Attachment

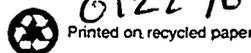


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Jeff Baker, NREL, Golden Office
Jim Doskocil, SSCPO
Ward Best, Ashtabula Area Office (AB)

cc: Other Organizations

National Low-Level Waste Management Program, EG&G Idaho
Hazardous Waste Remedial Action Program, (HAZWRAP)
Remedial Action Program Information Center
Center for Environmental Management Information

**Response to Questions and Clarification of Requirements and Processes:
DOE 5400.5, Section II.5 and Chapter IV Implementation
(Requirements Relating to Residual Radioactive Material)**



November 17, 1995

**Department of Energy
Office of the Assistant Secretary for Environment, Safety and Health
Office of Environment**

**Response to questions and guidance regarding implementation of
DOE 5400.5 Section II.5 and Chapter IV.**

DOE Radiological Release Criteria:

Order DOE 5400.5 chapters II and IV contain the Department's requirements for controlling and releasing property containing residual radioactive material. This guidance addresses release of non-real property and supplements information on release of structures.

The requirements for releasing real property, in lands and structures, are specifically documented in Chapter IV of DOE 5400.5 and additional guidance for applying the process is included in the "Manual for Implementing Residual Radioactive Material Guidelines Using RESRAD, Version 5.0," ANL/EAD/LD-2, September 1993 and related materials (see reference list). It is the responsibility of DOE field and, as appropriate, program offices to review and, where appropriate, approve measurement procedures and methodology and authorized limits for soil (lands) which meet DOE requirements for restricted or unrestricted use as specified in the Order and associated guidance. The information that follows relates primarily to the release non-real property including non-hazardous wastes, small items and equipment. The discussion relating to the use of DOE-approved surface contamination guidelines is also applicable to the release of structures.

DOE Requirements and Related Commercial Requirements:

Statement of Issue: Under the Atomic Energy Act of 1954 as amended, DOE has a responsibility to regulate the activities of its contractors and operations in a manner that protects the public and environment from radiation hazards associated with its operations. The Nuclear Regulatory Commission (NRC) and its Agreement States have similar authorities and responsibilities with regard to the commercial sector. In general, DOE requirements with regard to public and environmental protection are consistent with, and similarly protective as, those of the Nuclear Regulatory Commission and hence, are compatible with commercial standards. These include discharge limits as well as release limits. However, the residual radioactive material release limits for property are somewhat more complicated than effluent releases in that the property is likely being released to members of the general public.

Section 61.3 of 10 CFR Part 61 states that:

"(a) No person may receive, possess, and dispose of radioactive waste containing source, special nuclear or by-product material at a land disposal facility unless authorized by a license issued by the Commission pursuant to this part, or unless exemption has been granted by the Commission under §61.6 of this Part."

Many of the states' have enacted legislation that specifically preclude the disposal of any radioactive material, or formerly regulated radioactive material, except in disposal facilities designed and licensed for radioactive waste. The words "formally regulated" have apparently

been included to preclude the disposal of "Below Regulatory Concern" materials according to a draft policy that at one time had been published by the Nuclear Regulatory Commission.

The DOE field elements have asked EH to explain the relationship between DOE release criteria and policy and the requirements established for the commercial sector and non-DOE materials.

Analysis and response: The discussion and analysis to follow is limited to radiological protection. All DOE facilities and operations must conform to applicable external regulatory requirements. There are three general situations for which the DOE radiological criteria may be used. They are:

- 1) Application of DOE-derived and -approved radiological release criteria for disposal of material and property in a DOE-operated onsite landfill.
- 2) Application of DOE-derived and -approved radiological release criteria for disposal of material and property in a public or offsite landfill.
- 3) Application of DOE-derived and -approved radiological release criteria for sale or transfer of property to members of the public.

DOE On-site Landfill:

In situation 1) the Department has the responsibility and authority to establish limits for protection of the public and environment either in the form of radionuclide release criteria or waste acceptance criteria for disposal of materials in a DOE onsite landfill. Disposal of such material must conform to the requirements of Order DOE 5400.5 (and, when promulgated, as final rule 10 CFR Part 834) and applicable portions of Order DOE 5820.2A. DOE must establish limits such that doses to the public will be as far below the dose limits in DOE 5400.5 (or 10 CFR Part 834, as appropriate) as is practicable. This is determined on the basis of the ALARA Process (As Low As Reasonable Achievable process, see DOE March 1991 environmental ALARA guidance). The criteria should be such that it is not likely that disposal of materials into the landfill will result in a future requirement for remediation of the landfill subject to Chapter IV of DOE 5400.5. In making this determination, consideration should also be given to radionuclide limits established in CERCLA and RCRA corrective action Records of Decisions in neighboring areas of the site. To assure that these requirements and goals are achieved, authorized radiological limits for material sent to a DOE landfill (which is not an authorized low-level waste disposal facility) must be approved by DOE and should be:

- o Selected (and approved by DOE) on the basis of an assessment under the ALARA process to optimize the balance between risks and benefits including costs and collective doses and selected to ensure that individual doses to the public are less than 25 mrem in a year with a goal of a few millirem in a year or less.

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- o Evaluated to ensure ground water will be protected in a manner consistent with the objectives of the site's Ground-Water Protection Program objectives (DOE 5400.1) and/or applicable Federal or State requirements.
- o Evaluated to verify that release of the landfill property would not be expected to require remediation under DOE 5400.5 requirements for release of property containing residual radioactive material giving due consideration to experience gained from past or on going CERCLA or RCRA cleanup actions.

The ALARA process should consider factors such as estimated concentrations in waste, total activity (source term) being or likely to be disposed in the landfill, fraction of total waste containing residual radioactivity, estimated individual doses from expected or likely use scenarios, an estimate or assessment of collective doses in relation to other alternatives and potential impacts on natural resources such as ground water. In considering and assessing dose factors such as land use plans and site maintenance, benchmark cleanup standards, special waste form characteristics, and so forth may be considered in the development of authorized limits and acceptance criteria. The detail and complexity of the analysis should be commensurate with potential risks and costs, i.e., if potential individual and collective doses are very low a semi-quantitative or screening analysis may be acceptable (see DOE environmental ALARA guidance). However, other factors may also be important in determining the level of detail needed to approve such limits. For example, although screening analyses (conservative bounding estimates) of activity and potential doses that demonstrate low risk potential may be adequate to show that ALARA has been implemented, they are likely to significantly overestimate residual activity. The use of bounding estimates without adequate documentation of uncertainties or likely doses or quantities of material may result in misleading documentation that in turn could lead to costly and unnecessary investigations in the future. Therefore, it is recommended that procedures be established to document source term estimates as realistically as practicable or that bounding estimates be qualified with a discussion of uncertainty or estimates of expected quantities of residual radioactive material. Documentation supporting the authorized limits or acceptance criteria and disposal records should be sufficient to ensure that the site will not have to be remediated in the future or even unnecessarily surveyed to document its radiological condition.

Off-site Landfills:

In situation 2) DOE establishes and approves authorized limits and associated survey and release protocol for material that will be disposed in a non-DOE landfill. The recommended criteria for such a situation are similar to those established for release of property except that there is an additional consideration. Many local landfills have waste acceptance criteria or are subject to State requirements for radioactive material. In addition to meeting DOE requirements to establish authorized limits and survey, review and documentation protocols that ensure doses are as far below the primary dose limit as is practicable, authorized limits and release protocol must meet acceptance criteria and State requirements for the subject landfills. To ensure that these requirements and goals are achieved, authorized limits for

material sent to a non-DOE landfill (which is not an authorized low-level waste disposal facility) should be:

- o Selected (and approved by DOE) on the basis of an assessment under the ALARA process to optimize the balance between risks and benefits including costs and collective doses and to ensure that individual doses to the public are less than 25 mrem in a year with a goal of a few millirem in a year or less.
- o Evaluated to ensure that ground water will be protected in a manner consistent with the objectives of the applicable State regulations and guidelines.
- o Assessed to ensure that release of the landfill property would not be expected to require remediation under DOE 5400.5 or other applicable requirements for release of property containing residual radioactive material as a result of DOE disposals.
- o Coordinated with and acceptable to the landfill operator implementing the acceptance criteria and State representatives responsible for implementing solid waste regulations to ensure that DOE releases do not violate landfill-specific radiological protection requirements.

Equipment and Personal Property:

Under situation 3) the Department transfers ownership (either by sale or other means) to members of the public or releases personal property from DOE radiological control. Although DOE and DOE contractors are exempt from 10 CFR Part 61 and 10 CFR Part 20, individuals receiving the subject material are not. The Department will not transfer licensable materials to members of the public who are not licensed to receive them. Therefore, as part of the process for developing authorized limits for residual radioactive material and the associated survey and review protocol to ensure that release material and property are acceptable for public use, the Department must ensure that such property and material do not contain licensable amounts or concentrations of radionuclides. Therefore, the following criteria should be implemented to comply with DOE 5400.5 residual radioactive material requirements:

- o Authorized limits for property must ensure that doses to the public from all sources are less than the primary dose limit for all sources (100 mrem in a year).
- o Authorized limits for the property must be developed and approved by DOE consistent with the ALARA process. Appropriate protocols for survey and review of the release of such property must accompany the approval of the authorized limits. These limits shall be based on a documented finding that they are as low as practicable as determined through the ALARA process with a goal being to maintain individual doses low in comparison to background (e.g., a few millirem in a year or less). In any

case, the limits must be a fraction (e.g., 1/4 or less of the primary dose limit for the public). ALARA analysis should be consistent with the March 1991 DOE environmental ALARA guidance.

- o To ensure that DOE releases do not violate NRC licensing requirements, authorized limits for the release of property from DOE control should be coordinated with, and found acceptable to, appropriate Agreement State representatives or, where appropriate, NRC.

The all source criterion may be assumed satisfied if the ALARA criterion and its associated dose constraint and goals are adequately addressed. Generally, the use of the surface contamination guidelines discussed below will not require a quantitative dose assessment or detailed ALARA analysis; however, a qualitative review should be done and documented to determine if it is practicable to set authorized limits for surfaces lower than the guideline values.

DOE Approval of Authorized Limits and Measurement Protocols for Release:

While application, implementation and approval of authorized limits for property subject to surface contamination (consistent with guidelines described below) are the responsibility of DOE field and program elements, DOE 5400.5 requires EH-1 approval of authorized limits for residual radioactive material in mass or volume. However, authorized limits and survey protocol for residual radioactive material in mass or volume or surface contamination limits in lieu of Table 1 may be derived and approved by DOE field office managers without EH-1 written approval if:

- 1) The applicable criteria above are appropriately addressed;
- 2) Based on a realistic but reasonably conservative assessment of potential doses, it is demonstrated to the satisfaction of the responsible field office manager, that:
 - o the release or releases of the subject material will not cause a maximum individual dose to a member of the public in excess of 1 mrem in a year or a collective dose of more than 10 person-rem in a year;
 - o a procedure is in place to maintain records of the releases consistent with DOE 5400.5 requirements and that survey or measurement results are reported consistent with the data reporting guidelines in the DOE November 1992 radiological survey guidance and DOE/EH-173T; and
- 3) A copy of the authorized limits, measurement/survey protocols and procedures, supporting documentation including a statement that the ALARA process requirements have been achieved, and appropriate material documenting any necessary coordination with the state(s)

or NRC are provided to the Office of Environment, EH-4, at least 40 working days prior to the authorized limits becoming effective.

- o EH-4 will provide written notification to the field office of the receipt of the material and
- o notify the field, if the authorized limits or supporting material are not acceptable, within 20 days of receipt, otherwise the authorized limits (including any conditions or limitations set forth by the approving DOE field elements) may be considered approved without written EH-1 approval.

Field office elements may request technical assistance in the review or development of such authorized limits; however, such assistance should be requested as early as possible in the process but at least 90 working days before the desired implementation date for the authorized limits. Nothing in this guidance should be construed to override or replace the need for field elements to coordinate or consult with DOE program offices having jurisdiction over actions or portions of the actions covered by the authorized limits. Authorized limits for residual radioactive material in mass or volume that do not meet the field approval criteria stated above must be approved by EH-1. It is recommended that the DOE elements responsible for requesting EH approval, coordinate the analyses with EH-412, the Air, Water and Radiation Division prior to submitting the request to EH-1.

Guidelines for Property:

Surface Contamination Guidelines:

Statement of issue: DOE guidelines for release of residual radioactive material on surfaces are incomplete; the values for transuranics and alpha emitters are not included. EH-41 was requested to clarify existing guidance for the use of these guidelines.

Response: DOE 5400.5 Figure IV-1 includes surface guidelines for radionuclides other than transuranics and alpha emitters (Row 1 of Figure IV-1¹) and tritium. The only DOE-approved guidelines for release of property and material having residual surface concentrations of transuranics and the row 1 alpha emitters are contained in DOE/CH/8901, June 1989, see DOE 5400.5 Section IV.2, and were first approved for DOE-wide application in 1984 (memorandum from J. R. Maher to distribution, March 15, 1984). These values are consistent with NRC guidance ("Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source and Special Nuclear Material," July 1992 and "Termination of Operating Licenses for Nuclear Reactors," Regulatory Guide 1.86, June 1964). The Surface Concentration Guidelines are restated in Table 1.

¹Transuranics, I-125, I-129, Ra-226, Ac-227, Ra-228, Th-228, Th-230, Pa-231.

Field offices may approve authorized limits and survey protocol that meet these requirements. ALARA process requirements apply in addition to the guidelines restated in Table 1; however, in most cases, the ALARA requirements can be satisfied with a semi-quantitative or qualitative assessment.² Although full optimization studies are likely to be unnecessary because use of the surface guidelines generally ensure individual and collective doses will be low, it is desirable, where practicable, to estimate or bound potential individual doses and collective doses to the public associated with the release or annual releases (if the authorized limit will be applied to operational releases) and include the estimates in the documentation supporting the authorized limits. This may be important when the authorized limits are developed as part of a process for releasing non-real property on a regular basis over a long operational period. The level of detail should be commensurate to the potential doses. Qualitative screening estimates are adequate if they project collective doses to be less than 10 person-rem in total or annually. The attached reference list includes several reports and dose assessment tools which may be useful in computing or bounding doses.

Volume and Mass Contamination and Alternate Surface Limits: DOE has no DOE-wide approved guidelines for release of non-real property or structures containing residual radioactive material in mass or volume. Authorized limits for property subject to contamination in mass or volume must be derived consistent with the ALARA process and approved by DOE headquarters (EH-1) consistent with DOE 5400.5, Section II.5.c and this guidance (see "DOE Approval of Authorized Limits" above). Similarly, authorized limits for surface contamination different than those previously discussed may be approved by DOE on a case-by-case basis using the ALARA process. Authorized limits for the release of non-real property such as equipment or a number of similar items may be developed and approved by the Department. Guidance for the development of necessary protocols is also contained in the "Environmental Implementation Guide for Radiological Survey Procedures," Section 4.5, which was released for comment and use on November 30, 1992.

Tritium:

Statement of Issue: DOE surface guidelines in DOE 5400.5 do not specifically address tritium (³H). EH was requested to indicate whether the guidelines for beta emitters apply to tritium or if other values are applicable.

Response: Because tritium typically penetrates material it contacts, the surface guidelines in Figure IV-1 are not directly applicable to tritium contamination. Furthermore, the measurement of "fixed" tritium on surfaces at these levels is problematic. As a result, the

² While DOE has reviewed the surface contamination guidelines in the table and determined that they are protective, the level of protection is not necessarily uniform and hence, although the ALARA assessment may be qualitative or at most semi-quantitative, the level of detail should be commensurate with the potential maximum dose associated with the release. Radionuclides such as Th-232, Ra-226, and natural uranium have potential maximum doses up to a few mrem/year while I-129, Th-230, and Sr-90 have potential maximum doses of much less than 0.1 mrem/year. Release of property that meet the guidelines for the latter radionuclides justify very minimal ALARA consideration.

beta emitter values were not specifically recommended for tritium. The Department has reviewed the analysis conducted by the DOE Tritium Surface Contamination Limits Committee in the report, "Recommended Tritium Surface Contamination Release Guides", February 1991, and have assessed potential doses associated with the release of property containing residual tritium. The Department recommends the use of 10,000 dpm/100 cm² as an interim guideline for removable tritium. This guideline for removable surface contamination ensures that non-removable fractions and contamination in mass will not cause unacceptable exposures. The measurements should be conducted by a standard smear measurement³ but using a wet swipe or piece of styrofoam. If the property has been recently contaminated or recently decontaminated, followup measurements (smears) should be conducted at regular time intervals to ensure that there is not a build-up of contamination over time.

General Issues and Coordination:

The Department is presently conducting analyses, developing methodologies and working with EPA and NRC to develop more risk-based values that will ultimately replace (or confirm) the values in Figure IV-1, DOE/CH-8901, and the interim tritium limit discussed above. However, in the interim, the Department has determined that although the current levels are not internally consistent they are protective of the public and environment and can, therefore, continue to be used in the establishment of authorized limits for release at DOE facilities.

The Department also permits case-by-case determination of other limits where they are based on an ALARA process assessment and ensure that doses to the public will be as far below the DOE dose limits and constraints as is practicable. The derivation and DOE approval of such authorized limits should be consistent with the criteria discussed above. EH and EM have provide various tools (models, codes and handbooks) to support these analyses. They are listed in the references.

While risk-based standards are being developed NRC and its Agreement States are continuing to conduct site specific reviews and approvals consistent with the Commission's existing guidance (see attached October 25, 1995, letter Weber, NRC, to Wallo, DOE, commenting on this guidance). However, in coordinating with Agreement States or NRC, it may be useful for DOE elements to be aware of NRC's proposed 15 mrem/year dose constraint. Although this is only a proposed standard, the Commission has issued several draft guidance documents which may provide useful information in developing DOE survey protocols (see references). NUREG-1500 and NUREG-5512 may also be useful in benchmarking DOE dose assessments.

³See Section 4.2, Page 4.12, "Environmental Implementation Guide for Radiological Survey Procedures," November 1992, distributed for use and comment to Distribution for Raymond F. Pelletier, Office of Environmental Guidance, November 30, 1992.

Table 1. Surface Activity Guidelines
Allowable Total Residual Surface Activity (dpm/100 cm²)⁴

Radionuclides ⁵	Average ^{6,7}	Maximum ^{8,9}	Removable ^{9,9}
Group 1 - Transuranics, I-125, I-129, Ac-227, Ra-226, Ra-228, Th-228, Th-230, Pa-231	100	300	20
Group 2 - Th-natural, Sr-90, I-126, I-131, I-133, Ra-223, Ra-224, U-232, Th-232	1000	3000	200
Group 3 - U-natural, U-235, U-238, and associated decay products, alpha emitters	5000	15000	1000
Group 4 - Beta-gamma emitters (radionuclides with decay modes other than alpha emission or spontaneous ¹⁰ fission) except Sr-90 and others noted above	5000	15000	1000
Tritium (applicable to surface and subsurface) ¹¹	N/A	N/A	10000

⁴ As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by counts per minute measured by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

⁵ Where surface contamination by both alpha- and beta-gamma-emitting radionuclides exists, the limits established for alpha- and beta-gamma-emitting radionuclides should apply independently.

⁶ Measurements of average contamination should not be averaged over an area of more than 1 m². For objects of smaller surface area, the average should be derived for each such object.

⁷ The average and maximum dose rates associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/h and 1.0 mrad/h, respectively, at 1 cm.

⁸ The maximum contamination level applies to an area of not more than 100 cm².

⁹ The amount of removable material per 100 cm² of surface area should be determined by wiping an area of that size with dry filter or soft absorbent paper, applying moderate pressure, and measuring the amount of radioactive material on the wiping with an appropriate instrument of known efficiency. When removable contamination on objects of surface area less than 100 cm² is determined, the activity per unit area should be based on the actual area and the entire surface should be wiped. It is not necessary to use wiping techniques to measure removable contamination levels if direct scan surveys indicate that the total residual surface contamination levels are within the limits for removable contamination.

¹⁰ This category of radionuclides includes mixed fission products, including the Sr-90 which is present in them. It does not apply to Sr-90 which has been separated from the other fission products or mixtures where the Sr-90 has been enriched.

¹¹ Property recently exposed or decontaminated, should have measurements (smears) at regular time intervals to ensure that there is not a build-up of contamination over time. Because tritium typically penetrates material it contacts, the surface guidelines in group 4 are not applicable to tritium. The Department has reviewed the analysis conducted by the DOE Tritium Surface Contamination Limits Committee ("Recommended Tritium Surface Contamination Release Guides," February 1991), and has assessed potential doses associated with the release of property containing residual tritium. The Department recommends the use of the stated guideline as an interim value for removable tritium. Measurements demonstrating compliance of the removable fraction of tritium on surfaces with this guideline are acceptable to ensure that non-removable fractions and residual tritium in mass will not cause exposures that exceed DOE dose limits and constraints.

References:

Requirements:

Order DOE 5400.5, Radiation Protection of the Public and Environment, Department of Energy, Feb. 8, 1990, Revise Jan. 7, 1992 (change 2).

Surface Guidelines:

NRC publication, Guidelines for Decontamination and Decommissioning of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source or Special Nuclear Material, U.S. Nuclear Regulatory Commission, July 1984.

NRC Regulatory Guide 1.86, Termination of Operating Licenses for Nuclear Reactors, U.S. Nuclear Regulatory Commission, June 1976.

DOE/CH-8901, Manual for Implementing Residual Radioactive Material Guidelines - A Supplement to the U.S. Department of Energy Guidelines for Residual Radioactive Material at FUSRAP and SFMP Sites, Department of Energy, June 1989.

DOE Guidance Memorandum, "Unrestricted Release of Radioactively Contaminated Personal Property", J. Maher, DOE Office of Nuclear Safety, Mar. 15, 1984.

DOE Committee Report, Recommended Tritium Surface Contamination Release Guides, DOE Tritium Surface Contamination Limits Committee, Feb. 1991.

ALARA:

DOE Guidance: DOE Guidance on the Procedures in Applying the ALARA Process for Compliance with DOE 5400.5, Department of Energy, Office of Environmental Guidance, March 8, 1991.

ANL/EAD/LD-2, Manual for Implementing Residual Radioactive Material Guidelines Using RESRAD, Version 5.0, Chapters 1 and 5 and App. M, September 1993.

Measurement and Data Reporting:

DOE Manual for Use and Comment, Environmental Implementation Guide for Radiological Survey Procedures, Department of Energy, Office of Environmental Guidance, Nov. 1992

DOE/EH-0173T, Environmental Regulatory Guide for Radiological Effluent Monitoring and Environmental Surveillance, Department of Energy, Jan. 1991.

NUREG/CR-5849, Manual for Conducting Radiological Surveys in Support of License Termination, U.S. Nuclear Regulatory Commission, June 1992.

NUREG-1505, Draft Report, A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys, U.S. Nuclear Regulatory Commission, August 1995.

NUREG-1506, Draft Report, Measurement Methods for Radiological Surveys in Support of New Decommissioning Criteria, U.S. Nuclear Regulatory Commission, August 1995.

Dose Factors:

EPA-520-1-88-020, Federal Guidance Report No. 11, Limiting Radionuclide Intake and Air Concentrations and Dose Conversion Factors for Inhalation, Submersion and Ingestion, Environmental Protection Agency, Sept. 1988.

EPA 402-R-93-081, Federal Guidance Report No. 12, External Exposure to Radionuclides in Air, Water and Soil, Environmental Protection Agency, Sept. 1993.

Tools for Dose Assessment:

ANL/EAD/LD-2, Manual for Implementing Residual Radioactive Material Guidelines Using RESRAD, Version 5.0, Published by Argonne National Laboratory and prepared by ANL and DOE staff, September 1993.

ANL/EAIS-8, Data Collection Handbook to Support Modeling the Impacts of Radioactive Material in Soil, Argonne National Laboratory, April 1993.

ANL/EAIS/TM-103, A Compilation of Radionuclide Transfer Factors for Plant, Meat, Milk and Aquatic Food Pathways and Suggested Default Values for the RESRAD Code, Argonne National Laboratory, August 1993.

PNL-8724, Radiation Dose Assessments to Support Evaluations of Radiological Control Levels for Recycling or Reuse of Material and Equipment, Pacific Northwest Laboratory, July 1995.

ANL/EAD.LD-3, RESRAD-Build: A Computer Model for Analyzing the Radiological Doses Resulting from the Remediation and Occupancy of Buildings Contaminated with Radioactive Material, Argonne National Laboratory, November 1994.

NUREG-1500, Working Draft Regulatory Guide on Release Criteria for Decommissioning: NRC's Staff's Draft for Comment, U.S. Nuclear Regulatory Commission, August 1994.

NUREG-5512, Residual Radioactive Contamination From Decommissioning - Technical Basis for Translating Contamination to Annual Total Effective Dose Equivalent, Pacific Northwest Laboratory for the Nuclear Regulatory Commission, October 1992.

Appendix I Background and Summary Information

This guidance was prepared in response to several memorandum and phone requests from the field. Ultimately, EH will be responding to these as part of the promulgation of 10 CFR Part 834, "Radiation Protection for the Public and Environment." However, due to the delay in issuing the final rule, we are issuing this interim guidance for continued implementation of DOE 5400.5 until the rule becomes effective.

The Department's current requirements call for the establishment of DOE approved authorized limits for release of property containing residual radioactive material. The principal DOE 5400.5 requirements for the establishment of release limits are that the releases subject to the authorized limits not cause members of the public to receive doses in excess of the dose limits provided in the Order and that any doses be maintained as low as practicable as determined by the As Low As Reasonably Achievable (ALARA) process. These authorized limits must also be appropriately coordinated with the Nuclear Regulatory Commission (NRC) and Agreement States to ensure they are consistent with commercial standards.

The order established a procedure for developing authorized limits for soil and guidelines for surface contamination. However, because the Department had no procedures or specific criteria for property having contamination in mass, Order DOE 5400.5 required EH-1 approval of any authorized limits established for radionuclide contamination in mass. This was done to ensure DOE-wide consistency and to ensure processes resulted in protective requirements. Since 1990, EH in coordination with EM have developed tools and criteria to assist the field in developing such limits. Working together, EH and the field have gained considerable experience in implementing the process. As a result, this guidance permits the field to approve authorized limits and releases that meet DOE 5400.5 requirements without written EH-1 concurrence if the following conditions are also met:

1) Based on a realistic but reasonably conservative assessment of potential doses, it is demonstrated to the satisfaction of the responsible field office manager or the program office, that:

- o the release or releases of the subject material will not cause a maximum individual dose to a member of the public in excess of 1 mrem in a year or a collective dose of more than 10 person-rem in a year;
- o the releases and authorized limits will be appropriately documented; and

2) A copy of the authorized limits, measurement/survey protocols and procedures, supporting documentation including a statement that the ALARA process requirements have been achieved, and appropriate material documenting any necessary coordination with the State(s) or NRC are provided to the Office of Environment, EH-4, at least 40 working days prior to the authorized limits becoming effective.

EH has worked with the field on several efforts to establish authorized limits for release of recyclable property and the disposal of slightly contaminated material in DOE on-site landfills. These activities have provided a high degree of public protection and produced significant cost savings. Examples include:

- o Recycle of LBL Copper - Maximum dose to 0.15 mrem (less than 0.05% of the typical background dose and likely individual doses would be much less), collective dose 72 person-mrem, savings - \$247,000 plus a reduction in environmental impacts resulting from recycling.
- o Authorized limits for commercial reuse of explosives from DOE Pantex facility - Maximum dose 0.005 mrem, collective dose $<2 \times 10^{-3}$ person-mrem, savings \$1,000,000 annually plus reduction in annual emissions associated with alternative disposal process.
- o Disposal of roofing material in Hanford Central Landfill - Maximum dose to the public 0.001 mrem per year, qualitative estimate of collective dose was a few person-mrem per a few hundred years - savings \$345,000.

This process does not establish a de minimis for radioactivity in that the release of these materials will continue to be controlled by DOE field office personnel and will require their approval. It ensures protective and consistent application of the requirements by permitting the field offices authority to approve releases at very low doses. It provides EH time to intervene if a problem is identified. It permits establishment of a tracking system to allow EH to distribute useful information throughout the DOE complex and to provide better comments on related EPA efforts to developed national standards while reducing the review burden on the complex. It does not prohibit releases at protective but more cost-beneficial levels that are above the levels that the field may approve which are but instead, requires a greater level of review for such approvals. The process and requirements are consistent with Nuclear Regulatory Commission requirements and will be appropriately coordinated with external regulators. It will help streamline the regulatory function of the Department and reserve EH resources for only potentially higher risk issues.

In addition to the resources and cost savings associated with the revised process, a clear structured approach to control and release of material will result in improved environmental protection. The lack of adequate guidance and a clear process has been a root cause of previous incidents where DOE facilities have inappropriately released radioactive material. When consistent, clear and logical processes are not implemented, some facilities have developed their own procedures and policies that have resulted in inconsistent and undocumented releases. The guidance in this memorandum will further DOE efforts to resolve these problems and ensure that public protection is integrated into facility operations rather than addressed as an after thought.



UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

October 25, 1995

Dr. Andrew Wallo
U.S. Department of Energy
Forrestal Building
EH-09
Washington, DC 20545

Dear Dr. Wallo:

In response to your facsimile request, we have performed a review of the Department of Energy's (DOE) guidance to its field offices on the implementation of DOE release criteria set forth in Order 5400.5. As you know, the release of any contaminated material to an unlicensed party involves technical and policy concerns that are not completely addressed by existing Nuclear Regulatory Commission guidance.

Recognizing the incomplete nature of existing regulations and guidance, we agree that DOE field offices should coordinate with the Agreement States or NRC, as well as other applicable regulatory authorities (e.g., State permitting agencies) to ensure that the site specific release limits and the survey and review protocols are appropriate and acceptable. We also believe your application of the ALARA process is reasonable for establishing authorized limits for material either sent to a non-DOE landfill or transferred to the public. However, as you know, a 15 mrem/yr dose constraint is currently included within our proposed rule on radiological criteria for decommissioning. Your upper bound constraint level of 25 mrem/yr for release of property to a landfill or the public could be viewed as being inconsistent with this proposed value, even though we recognize that the calculated radiological impacts from actual releases are typically well below this value. With regard to the collective dose constraint imposed on the release of surface-contaminated property, we were not able to make a judgement, without further information on assessment methods, on the appropriateness of using the 10 person-rem as a threshold value, below which only qualitative screening would be required.

The case-by-case decisions that NRC has made in releasing land and structures for unrestricted public use have been typically related to NRC's Site Decommissioning Management Plan sites. Release criteria have been those in the "Action Plan to Ensure Timely Cleanup of Site Decommissioning Management Plan Sites" (57 FR 13389-13392) or have been justified as being consistent with these criteria (See SECY-94-145 enclosed). These criteria include surface contamination guidelines in Regulatory Guide 1.86. In the few instances where radioactive material has been or is being considered for transfer to unlicensed entities on a case-by-case basis, impact analyses have been performed and accepted by appropriate approval authorities. These analyses generally indicate extremely small (i.e., less than 1 mrem/year) annual individual exposures and minimal collective exposures.

A. Wallo

- 2 -

As you know, the criteria in NRC's proposed decommissioning rule, when finalized, will replace the above guidance, but only for release of lands and structures. Both EPA and NRC have then indicated that a proposed recycle rule will be developed.

As with the "decommissioning" rule, the applicable dose criteria will be the subject of considerable debate. It is likely that the criteria in the decommissioning rule could influence the selection of criteria for this recycle rule.

I hope this response has been responsive to your request.

Sincerely,



Michael F. Weber, Chief
Low-Level Waste and Decommissioning
Projects Branch
Division of Waste Management
Office of Nuclear Material Safety
and Safeguards

18



POLICY ISSUE (Information)

May 27, 1994

SECY-94-145

FOR: The Commissioners

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: INCREASE OF TRITIUM AND IRON-55 UNRESTRICTED USE LIMITS FOR
SURFACE CONTAMINATION AT SHOREHAM AND FORT ST. VRAIN

PURPOSE:

To inform the Commission of the staff's decision to increase certain unrestricted use limits for surface contamination on buildings, structures, and equipment for the decommissioning projects at Fort St. Vrain Nuclear Station (FSV) and Shoreham Nuclear Power Station (Shoreham). This increase applies only to fixed contamination from iron-55 (Fe-55) and tritium (H-3), and is a modification to one of the clean-up criteria that the Commission directed the staff to consider in the "Action Plan to Ensure Timely Cleanup of Site Decommissioning Plan Sites" (Action Plan) (i.e., Table 1 of Regulatory Guide 1.86 (RG 1.86)).

SUMMARY:

Shoreham and FSV are currently decommissioning their facilities with the goal of releasing the buildings, equipment, and grounds for unrestricted use in the short term (i.e., the DECON option). The major radionuclide identified in contaminated structures and systems during the characterization of these facilities was cobalt-60 (Co-60). However, during the dismantling of the facilities, both licensees identified concrete containing unexpectedly high concentrations of H-3 and Fe-55 in areas subjected to neutron radiation during operations. Shoreham also identified Fe-55 in steel used as a liner for the bioshield concrete. The total (fixed plus removable) surface contamination

Contact: David N. Fauver, NMSS
415-6625

SECY NOTE:

TO BE MADE PUBLICLY AVAILABLE IN 10 WORKING DAYS FROM THE DATE OF THIS PAPER.

Enclosure

levels of H-3 and Fe-55 on the concrete and steel exceed the current unrestricted use limits. Smaller concentrations of europium-152 (Eu-152) and Co-60 have also been identified on the activated material. Shoreham estimates that compliance with the current surface contamination limits for the H-3 and Fe-55 in the activated material will require the shipment of up to 73 additional cubic meters (2600 cubic feet) of slightly contaminated concrete and steel, above that required to comply with the Co-60 and Eu-152 surface contamination limits alone, to a licensed low-level waste facility at a cost of up to \$1 million. FSV has estimated that compliance with the current surface contamination limits would require the removal and shipment of up to 260 additional cubic meters (9300 cubic feet) of activated concrete, at a cost of up to \$4.5 million. Both licensees submitted requests to increase the unrestricted use limits for H-3 and Fe-55 surface contamination based on their conclusions that the risks from H-3 and Fe-55 are very small, at the current limits, and that application of the current limits to the activated material at their facilities is not in accordance with the as low as is reasonably achievable (ALARA) principle.

The staff evaluated the risk from H-3 and Fe-55 surface contamination relative to the other radionuclides listed in RG 1.86. The risks from Fe-55 and H-3 surface contamination at their respective RG 1.86 limits were found to be lower than the average risk from the other nuclides by at least a factor of 400. The risks are lower because Fe-55 emits only low-energy X-rays and H-3 emits only low-energy beta radiation. A detailed discussion of the relative risks, and additional background information, is contained in the enclosure.

Based on the magnitude of the disparities between risks, the staff concluded that it would be ALARA to increase the surface contamination limits for H-3 and Fe-55 at FSV and Shoreham, and that the magnitude of the increase could be limited to ensure that the risks from H-3 and Fe-55 remain consistent with the risks from the other nuclides in RG 1.86. The staff contemplated raising the total average surface contamination limits for H-3 and Fe-55 from the current 5,000 dpm/100 cm² limit to 2,000,000 dpm/100 cm². At this level, the risk from Fe-55 would be equal to the average risk from the other nuclides in RG 1.86, and the risk from H-3 would be less than the average. However, in consideration of ALARA, the staff selected 200,000 dpm/100 cm² as the new total average surface contamination limit for H-3 and Fe-55 at FSV and Shoreham.

RG 1.86 also contains maximum limits that are 3 times the average limit. Accordingly, the maximum limit for total H-3 and Fe-55 surface contamination was raised by a factor of 3 to 600,000 dpm/100 cm².

The existing limits for removable H-3 and Fe-55 surface contamination were also evaluated. Increasing the removable contamination limits is not considered ALARA, since standard remediation techniques are capable of lowering removable contamination to levels below the current limit. Therefore, the removable limits for H-3 and Fe-55 will remain at the current level of 1000 dpm/100 cm².

The primary bases for the staff's decision to increase the surface contamination limits for H-3 and Fe-55 were 1) that the risk from the increased limits are consistent with the risk from other nuclides, and 2) ALARA. However, to provide additional information, dose assessments were performed assuming that contamination is present at the increased limits

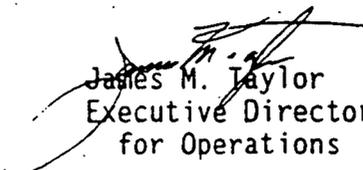
At 200,000 dpm/100 cm², the estimated doses from H-3 and Fe-55 surface contamination are 0.01 and 1.1 E-03 mSv (1.0 and 0.11 mrem)/y, respectively. These potential doses will decline as Fe-55 and H-3 decay with half-lives of 2.6 years and 12.2 years, respectively.

The above calculations assume that the dose from the activated concrete is from the inhalation and ingestion of material resuspended or removed from contaminated surfaces, which are considered the most probable exposure pathways. Although considered unlikely, the staff also estimated the potential dose assuming that the activated concrete is removed from the buildings and disposed without restrictions. For this case, the groundwater pathway, and the other exposure pathways in the residential farmer scenario, were evaluated. The resulting potential doses from the unrestricted disposal of the material at Shoreham and FSV were estimated as 0.012 mSv/y (1.2 mrem/y) and 0.019 mSv/y (1.9 mrem/y), respectively. These modeled doses decline rapidly with time following the maximum due to the conservative assumption that all of the H-3 is leached in the first year. For example, the fourth and sixth year doses at both facilities are less than 5E-03 and 2E-04 mSv/y (0.5 and 0.02 mrem/y), respectively.

The modified Fe-55 and H-3 limits discussed in this paper are based on ALARA considerations specific to the FSV and Shoreham decommissioning projects. However, volumetric contamination of activated materials at decommissioning power reactors is a generic issue and the staff anticipates that other decommissioning reactors will make similar requests for exceptions to RG 1.86 for H-3 and Fe-55, and that such requests would likely be approved. This issue should be addressed generically in the guidance developed to implement the rulemaking on Radiological Criteria for Decommissioning. This guidance would supersede or modify RG 1.86. The staff expects to notify Shoreham and FSV that the limits have been increased no later than June 3, 1994. This early date is associated with the time required to remove and ship low-level waste, if necessary, to Barnwell prior to the June 30, 1994, deadline. A meeting with the Commission assistants to discuss this matter would be useful.

COORDINATION:

The Office of the General Counsel has reviewed this paper and has no legal objection.


James M. Taylor
Executive Director
for Operations

Enclosure:
As stated

ADDITIONAL BACKGROUND AND TECHNICAL INFORMATION

In the Action Plan, the Commission directed the staff to consider existing guidance, and ALARA, when determining a site's suitability for unrestricted use pending the final rule on "Radiological Criteria for Decommissioning." Table 1 of RG 1.86, which contains surface contamination limits for buildings, structures, and equipment, was listed in the Action Plan as one of the existing criteria to consider. Accordingly, the staff approved the limits in RG 1.86 as the unrestricted use criteria for surface contamination at FSV and Shoreham.

The experience, to date, in applying the RG 1.86 surface contamination limits as unrestricted use criteria has been positive. In general, the limits are practical and ALARA, and have been applied by Nuclear Regulatory Commission licensees undertaking decommissioning. However, on December 23, 1993, the Public Service Company of Colorado (PSC) requested that NRC increase the surface contamination limits for Fe-55 and H-3 at FSV. The Fe-55 and H-3 contamination identified at FSV resulted from neutron activation and is primarily located in the concrete comprising the former prestressed concrete reactor vessel. PSC contends that the surface contamination limits for these two nuclides should be increased since: 1) the potential health and safety risk from these nuclides is very low; 2) the relative risk from these nuclides is very low compared to the other nuclides of concern at FSV, predominantly Co-60 and Eu-152; and 3) compliance with existing limits for these nuclides would cost an additional \$4.5 million above the cost required to comply with the surface contamination limits for Co-60 and Eu-152.

In addition, on April 22, 1994, Shoreham requested that NRC consider revising the release criteria for Fe-55 and H-3 because of the recent identification of concrete and steel, from the biological shield, that contains elevated levels of H-3 and Fe-55 as a result of neutron activation. Shoreham asserts that applying the current unrestricted release criteria for Fe-55 and H-3 would cost up to \$1 million without appreciable decrease in potential risk to public health and safety, and that Fe-55 and H-3 pose significantly lower risk than other nuclides in RG 1.86.

In response to the FSV and Shoreham requests, the staff reviewed the technical bases for Table 1 of RG 1.86 to determine if the limits for H-3 and Fe-55 are inconsistent with the other nuclides in Table 1. The RG 1.86 limits were developed using a semi-quantitative evaluation of relative risk based on the maximum permissible concentrations (MPC) for air and water listed in 10 CFR Part 20. The starting assumption in developing the limits was that licensees should not be expected to lower surface contamination below the existing environmental background levels caused by fallout from the atmospheric testing of nuclear devices. The predominant radionuclide found in the environment as a result of atmospheric testing is strontium-90 (Sr-90); the background level of Sr-90 was about 1000 dpm/100 cm² in 1974, when RG 1.86 was published.

The surface contamination limits in Table 1 of RG 1.86 were selected using the 1000 dpm/100 cm² environmental background level for Sr-90 as the baseline. Using the ratio of Sr-90 MPC's to the MPC's for the various nuclides as a measure of relative risk, the RG 1.86 surface contamination limits were generally set at one of three levels, i.e., 100, 1000, or 5000 dpm/100 cm². The upper limit of 5000 dpm/100cm² was set, in part, to limit direct radiation and, in part, as a level that seemed readily attainable. Note that the direct radiation exposure from both Fe-55 and H-3 is essentially zero. The ability to measure the contamination using standard industry instrumentation and methods was also considered in setting the RG 1.86 limits.

The developers of RG 1.86 were aware that disparities existed between the risks from the various nuclides at their respective RG 1.86 limits, but issued the guide as a matter of practicality, realizing that a method for more closely estimating the risk from surface contamination would not be accepted as a consensus in the near term. Before the FSV and Shoreham cases, there has not been a compelling reason to evaluate more closely the magnitude of the disparities. Note that the inconsistencies in RG 1.86 are being addressed generically in the rulemaking on "Radiological Criteria for Decommissioning."

To determine if the RG 1.86 limits for H-3 and Fe-55, i.e., 5000 dpm/100 cm², pose relative risks that are significantly lower than other nuclides, the staff evaluated the risk from each of the nuclides specifically listed in RG 1.86, as well as Co-60, Cs-137, Eu-152, C-14, Ni-59, Ni-63, H-3, and Fe-55. The risk from each nuclide was estimated by multiplying the RG 1.86 limit by the nuclide specific dose factors for surface contamination developed for the building occupancy scenario in NUREG/CR-5512, "Residual Radioactive Contamination from Decommissioning," October 1992. For example, the NUREG/CR-5512 dose factor for Sr-90 surface contamination is 1.51E-05 mSv/y (1.51E-03 mrem/y) per dpm/100 cm² and the RG 1.86 surface contamination limit for Sr-90 is 1000 dpm/100cm². Multiplying these two values results in a dose of 1.51E-02 mSv (1.51 mrem)/y. Note that the NUREG/CR-5512 dose factors are currently the staff's best estimate of dose from surface contamination. These dose factors were developed to support the ongoing rulemaking on "Radiological Criteria for Decommissioning."

The estimated doses for the 24 nuclides evaluated range from about 0.8 mSv/y (80 mrem)/y for uranium to about 2E-05 mSv (2E-03 mrem)/y for I-133. The resulting doses for Co-60, Fe-55, and H-3 are 0.14, 2.5E-04, 2.8E-05 mSv (14, 2.5E-02, and 2.8E-03 mrem)/y, respectively. The average dose for the 24 nuclides evaluated was about 0.1 mSv (10 mrem)/y.

Comparing the relative risks from Co-60 to both Fe-55 and H-3, it is seen that the risk from Co-60, at the RG 1.86 limit, is 5000 times greater than the risk from H-3 and 560 times greater than the risk from Fe-55. Because of the magnitude of these differences, and the estimated cost of compliance with the existing limits at FSV and Shoreham, the staff believes that it is appropriate to consider the surface contamination limits for H-3 and Fe-55 separately.

To determine the total surface contamination limits, the staff considered the average dose from the 24 nuclides at their respective RG 1.86 limits, and ALARA. In addition, to maintain simplicity in implementation, the limits for both H-3

and Fe-55 were both set at equivalent levels, using the dose from Fe-55 as the basis. This results in a more conservative dose for H-3. The estimated average dose from nuclides at the RG 1.86 limits, as evaluated above, is 0.1 mSv (10 mrem)/y, which translates to a limit of 2,000,000 dpm/100 cm². In consideration of ALARA, the limit finally selected for total surface contamination from Fe-55 and H-3 at FSV and Shoreham was 200,000 dpm/100 cm². At 200,000 dpm/100 cm², the estimated Fe-55 and H-3 doses are 0.01 and 1.1 E-03 mSv (1.0 and 0.11 mrem)/y, respectively. These potential doses will decline as Fe-55 and H-3 decay with half-lives of 2.6 years and 12.2 years, respectively.

The 200,000 dpm/100 cm² limit discussed above applies to average contamination levels. RG 1.86 also contains maximum limits that are 3 times the average limit. Accordingly, the maximum limit for H-3 and Fe-55 total surface contamination was raised to 600,000 dpm/100 cm².

The existing limits for removable H-3 and Fe-55 surface contamination were also evaluated. Increasing the removable contamination limits is not considered ALARA since standard remediation techniques are capable of lowering removable contamination to levels below the current limit. Therefore, the removable limits for H-3 and Fe-55 will remain at the current level of 1000 dpm/100 cm².

The above dose calculations assume that the exposure from the activated concrete is through the inhalation and ingestion of material resuspended or removed from contaminated surfaces, which are considered the most probable dose pathways. However, although considered unlikely, the staff also evaluated the potential dose assuming that the activated concrete is removed from the buildings and disposed without restrictions. The potential dose was estimated using the RESRAD environmental pathway and dose assessment code (ANL/EAD/LD-2). The resulting dose is considered conservative since, 1) the activity in the concrete and steel was assumed to be immediately available for uptake by plants and animals, and 2) all of the H-3 is assumed to leach from the concrete in the first year and migrate to groundwater. The resulting maximum potential doses from groundwater, plus the other exposure pathways in the residential farmer scenario, at Shoreham and FSV are 0.012 mSv/y (1.2 mrem/y) and 0.019 mSv/y (1.9 mrem/y), respectively. The modeled dose declines rapidly with time following the maximum due to the conservative assumption that all of the H-3 is leached in the first year. For example, the fourth and sixth year doses at both facilities are less than 5E-03 and 2E-04 mSv/y (0.5 and 0.02 mrem/y), respectively.

**Supplemental Information and Examples for
EH-41 Guidance on
"Application of DOE 5400.5 requirements for release and control of property
containing residual radioactive material" dated November 17, 1995**

Contents:

Summary of Examples:

Approval of Copper Recycle Action:

Approval of Authorized Limits for the Recycle of High Explosives:

Approval for On-site Disposal of Roofing Material:

Supplemental Information and Examples:

Summary of Examples:

This package contains examples of documentation and analyses supporting authorized limits and requests for release of material containing residual radioactive material that have been approved by the Office of Environment, Safety and Health. They are provided with the EH-41 November 17, 1995, guidance as examples of the type of analysis and documentation EH has found acceptable to justify authorized limits and releases. Under the November 17, 1995, guidance, each of these example actions and authorized limits could be approved by the responsible field office. The documents should not necessarily be considered templates but are provided to DOE field and program elements to assist them in reviewing requests under their jurisdiction. The packages represent three different situations and provide a general outline for issues that need to be considered. However, for some site-specific actions, these analyses may be too detailed and for others insufficient.

Copper Recycle:

The package contains the ALARA summary for the action, approval documentation and a fact sheet on the action. This action was supported by an environmental assessment (EA) which is not attached because of its volume. The data in the ALARA summary was based on analyses contained in the EA. The Fact Sheet summarizes the results of the EA.

The copper is from the windings of a cyclotron and the most highly contaminated portions were removed and disposed prior to the action to recycle the copper. As a result, the action was to determine if the remaining copper was acceptable for recycle rather than to establish authorized limits for the recycle of the copper. Had the more highly contaminated copper not been already disposed, the action would have required the analysis to determine authorized limits to define the portion of copper that could be recycled. However, given the concentration and quantity of residual radionuclides in the remaining copper that was not necessary.

Recycle of High Explosives:

This example provides the ALARA finding and supporting documentation and approval documents for the establishment of authorized limits to recycle high explosives containing residual tritium. An ALARA assessment should normally investigate the impacts and benefits of various authorized limits (e.g., 10,000 dpm/100 cm², 1000 dpm/100 cm² and 100 dpm/100 cm²). However, in this case, the individual and collective doses associated with the proposed authorized limit were so low that there was no value in assessing the lower limits and it was qualitatively determined that a higher limit would provide no significant cost savings. Hence, the field office recommended approval of the proposed value without quantitative ALARA comparisons of alternatives.

Disposal of Roofing Material in an On-site Landfill:

It is possible to establish waste acceptance criteria for material containing low concentrations of residual radioactive material for a landfill (authorized limits for disposal) consistent with the requirements of DOE 5400.5 and DOE 5820.2A. However, there were no site-wide criteria established for the landfill in this case. As a result, the field chose to review the specific waste to determine if it could be disposed in the on-site landfill. The package contains the approval documents and supporting analysis. EH had several comments (included in the package) regarding the conservatism of some of the assumptions and the collective dose assessment. However, given the low doses, and the results of independent analyses conducted by EH, it was determined that the changes would not impact the decisions and the action was approved.

EXAMPLE - Copper Release

United States Government

Department of Energy

memorandum

DATE: August 2, 1994

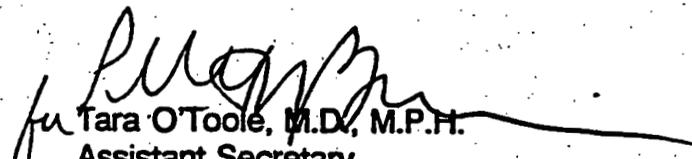
REPLY TO:
ATTN OF: Office of Environmental Guidance:Wallo:6-4996

SUBJECT: DOE Order 5400.5 Review and Approval of the Lawrence Berkeley Copper Recycle Action

TO: James F. Decker
Deputy Director
Office of Energy Research

The Office of Environment has reviewed the documentation for the subject action, including the attached As Low As Reasonably Achievable (ALARA) Summary Information and the supporting information contained in the Environmental Assessment and the finding of no significant impact, which I signed on June 20, 1994. We find that the action complies with the requirements set forth in Section II.5 (Release of Property Having Residual Radioactive Material) and with the ALARA process requirements of Section II.2 of DOE Order 5400.5 and associated guidance. Accordingly, I approve your request for release of the subject material, consistent with Section II.5.c(6) of DOE Order 5400.5.

If you have any questions please contact Mr. Andrew Wallo, Office of Environmental Guidance, at 202-586-4996.


Tara O'Toole, M.D., M.P.H.
Assistant Secretary
Environment, Safety and Health

SUMMARY OF ALARA ANALYSIS FOR THE DISPOSITION OF COPPER FROM LBL CYCLOTRON COIL WINDINGS

BACKGROUND

Lawrence Berkeley Laboratory (LBL) has 140 metric tons of copper which had become slightly activated from use as windings of the 184-inch cyclotron. The copper has been stored in 32 wood crates outdoor at a leased warehouse for several years and DOE would like to dispose of it. The amount of radioactive material is sufficiently low that the State of California, Department of Health has approved burial of the copper as ordinary waste, without regard the activity and found that the recycle of the material is acceptable under the practice of risk-based regulations. However, the copper is a valuable resource and could be sold for scrap for about \$0.80/lb (approximately \$247,000 for the 140 metric-ton lot) and recycled. DOE would like to make a final disposition of the copper and comply with the ALARA policy and requirements.

CONTAMINANTS

The high-purity (99.99%) copper has an average activity, principally Co-60 (half-life 5.26 years, beta and gamma emitter), of 3 pCi/g from activation and a maximum activity of 20 pCi/g. All of the copper with activity greater than 20 pCi/g has been disposed in Hanford. The total amount of Co-60 in the remaining copper is about 0.42 mCi. If the total amount of Co-60 in the 140 metric tons of copper (0.42 mCi), could be concentrated into a single small unshielded source, the dose rate at 1 ft from the source would be about 5.5 mrem/hr. About 1.5 pCi/g of Ni-63 (half-life 92 years, beta emitter) is also present, but it is of little radiological importance.

PROPOSED ACTION AND ALTERNATIVES

DOE proposed to recycle the copper by selling it to a local scrap metal dealer. Several local dealers are interested and the nearest is located within 10 miles of the warehouse. Five alternative actions also were considered and evaluated:

1. no action -- continue to store the copper at the warehouse [this would require implementation of DOE 5820.2A for long-term storage of low-level waste--the Co-60 activity would be undetectable through decay in about 50 years];
2. recycle at the Scientific Ecology Group (SEG) facility, located in Oak Ridge TN, for re-use at a DOE facility [the likely use would be as customized shielding blocks which eventually would be disposed as low-level waste];
3. recycle by selling or giving the copper to a foreign government [China is interested in using the copper in synchrotron accelerators--transportation would be by common carriers];
4. disposal at a local sanitary landfill [a local sanitary landfill is available but some additional testing would be required]; and
5. disposal at the Hanford Low-Level Waste Burial Facility [common carriers would be used to transport the copper to Hanford, Washington].

RADIOLOGICAL IMPACT

Members of the Public The uses by the public of the copper through recycling include home wiring, electronic components, and jewelry. A maximum collective population dose of 72 person-mrem was estimated from the reuse of the copper as jewelry. An additional 0.003 person-mrem would result from transportation to the recycle facility. The potential biological risk, assuming 500 radiation induced fatal cancers per million person-rem, would be about 0.00004 fatal cancers for the exposed public--in addition to the normal incidence. The normal incidence of cancer among individuals is about 1 in 3, about half of which are fatal.

Radiation Workers Transporting and recycling the copper were estimated to cause a collective doses of 0.0004 and 0.04 person-mrem, respectively, to workers. Potential fatal cancers would be 2×10^{-10} and 2×10^{-8} , respectively. Workers in the warehouse, for the storage option, would receive 0.1 person-mrem, with an associated fatal cancer incidence of 6×10^{-8} .

DOSE AND COST/BENEFIT SUMMARY

A summary of the doses and costs are presented in Table B-1.

TABLE B-1 COSTS AND DOSES FOR THE ALTERNATIVE COPPER DISPOSAL ACTIONS

Alternative Action	Maximum public individual dose, (mrem)	Collective dose public + worker (person-mrem)	Cost [saving] of alternative (\$1,000s)	Net cost [saving]c (\$1,000s)
Unrestricted use	0.15	72	[247]	[247]e
Storage [50 yr]	0.015	0.115	50/[247]bd	[197]e
Recycle @ SEG	a	0.14	323	323
Sale/gift-foreign	a	0.047	30	30
Disp./ Hanford	3×10^{-6}	0.0034	235	235
Sanitary fill	a	0.0034	4.2	4.2

- a Dose is essentially averted by alternative.
- b Assumes 50 years storage at \$1,000 per year. However, at that time the copper could be recycled and \$247,000 recovered for a net savings of about \$197,000.
- c A monetary equivalent of \$1,000 per person-rem collective dose (\$1 per person-mrem) was assumed in this summary. However, the collective dose is so small that there would be no significant change if \$10,000 per person-rem had been selected.
- d The interest considerations for cash received from the sale of the copper and payments for storage over the 50-year period were not included in this evaluation.
- e No attempt was made to assign a monetary value for the avoidance of environmental impacts from processing copper for which the reused copper is substituted, or other considerations such as those identified below.

The collective dose is so small that the choice of alternatives would not change if \$10,000 per person-rem were to be assumed (as was the case in the actual EA). A value of \$1,000 per person-rem has been used by the Nuclear Regulatory Commission. This value was chosen at a time when the risk coefficient for radiation was assumed to be about 200 fatal cancers per 1,000,000 person-rem. Based on the risk coefficient presently used by the Department (500 fatal cancers per 1,000,000 person-rem) an increase in monetary equivalent unit dose value may or may not be warranted. However, from a health-effect consideration, an assumption of \$10,000 per person-rem appears to be an excessive value for monetary equivalent unit dose unless other considerations are included. In any case, as noted above, the potential doses are so small the factor is not significant in the selection process.

OTHER CONSIDERATIONS

Additional benefits of the proposed recycling action would include:

1. environmental consequences, e.g., air emissions, water quality, energy use, and traffic, associated with the mining and processing of copper ore to produce an equivalent quantity of copper would be averted;
2. valuable, and expensive, low-level radioactive waste burial space for material that is actually classified as radioactive waste would be preserved;
3. valuable sanitary landfill space would be preserved;
4. currently used storage space would be released;
5. compliance with the DOE waste minimization and pollution prevention policy would be achieved; and
6. copper, a valuable resource, would be preserved.

In review of this action, potential impacts on special industries such as the electronics or photographic industry were considered and determined to be nonexistent. The levels of radioactive material in the subject material are too low to be of any concern. Furthermore, the relatively short half-life of Co-60 (5.2 years) ensures that there is no concern for buildup of this material in the metals pool.

DISCUSSION AND CONCLUSIONS

Clearly, the proposed recycle option is preferred from ALARA considerations, not only on the basis of cost considerations, but also in consideration of the "additional benefits," listed above. In this case, both the individual and collective doses to the public and to workers are too small to be a consideration in selecting any of the options.

EH-23 Routing and Transmittal

Date: Aug. 1, 1994

Name	Initials	Date
1) To:	AWZ	8/1/94
1) R. Pelletier, EH-23	RBB	8/2/94
2) R. Berube, EH-20		
3) P. Brush, EH-2		

4) T. O'Toole EH-1
 Subject: ER Request for DOE 5400.5 LBL Copper Recycle Approval.

Handwritten: 8/4/94

EH-1 signed the FONSI for this action on June 20, 1994. On July 6, 1994, ER sent a memo to EH-1 requesting EH approval required by DOE 5400.5. Note: The ER package was lost in distribution and we did not receive it until the week of July 25, 1994.

We have reviewed the package and supporting material and find that all the requirements of DOE 5400.5 relevant to this action have been satisfied. In general, the ER and LBL staff have done an excellent job on this project. We reviewed the initial proposal in 1993 and provided comments and suggestions. LBL staff implemented a program to make necessary measurements and collect the data we requested. It is rare that field and program offices respond so quickly and thoroughly. We intend to use this project as an example of an ALARA analysis in our next revision of the ALARA guidance for 10 CFR Part 834.

Recommendation: Sign memorandum approving action.

Handwritten signature of Andrew Wallo

Andrew Wallo, Director EH-232 Air, Water and Radiation Division	Room GA098 Phone, 202-586-4996 FAX, 202-586-3915 E-mail, andrew.wallo@hq.doe.gov
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Recycling of Irradiated Copper at Lawrence Berkeley Laboratory Fact Sheet

- o The proposed action is to release 310,000 pounds (140 metric tons) of irradiated copper coil windings from the decontamination and decommissioning of a cyclotron at Lawrence Berkeley Laboratory (LBL).
- o The irradiation resulted in internal contamination, primarily cobalt-60 (half-life 5.3 years).
- o The most contaminated copper has been separated and sent to disposal. The only portions that remain (and are being proposed for recycle) are those that were only slightly irradiated.
- o The average residual radioactivity level is 3 picocuries per gram.
- o The proposal is to release (sell) the copper to the open market for unrestricted use.
- o The market value of the copper is about \$250,000.
- o Alternatives considered include:
 - Continued storage.
 - Recycle:
 - Reutilization (as mixed metal) at another DOE facility.
 - Sale.
 - Disposal:
 - Landfill.
 - DOE low-level waste site.
- o Maximum radiological health impacts (Note: these are extremely low and hypothetical and of little national consequence):
 - Cumulative population over 30 years - 0.5 person-mrem (3×10^{-7} fatal cancers) (assuming use in home wiring) - for other uses range is from 0 to 72 person rem ($< 4 \times 10^{-5}$ fatal cancers).
 - Maximum individual lifetime - 0.15 mrem (8×10^{-8} risk).
 - Additional one-time population for transportation - 0.003 person rem (2×10^{-9}).
 - Worker risk 2×10^{-8} or less.
- o Environmental benefits (Note: these benefits are hypothetical and of little national significance):
 - AIR:
 - Avert 90,000 lbs of particulate emissions.
 - Avert 38,000 lbs of sulfur dioxide emissions.
 - WATER: Avert theoretical contamination of water from copper production (arsenic, lead, etc.)
 - ENERGY: More than a 90% energy savings.
 - LAND: Preserve disposal site space.
 - OTHER: Consistent with waste minimization policies.
- o DOE's requirements applicable for this release are:
 - DOE 5440.1E: Appropriate level of NEPA documentation - in this case staff recommended that the EA/FONSI is appropriate.
 - DOE 5400.5: Conduct and document ALARA analysis to support specific authorized limits for release or reuse, obtain State(s) (and where appropriate EPA and NRC) agreement and obtain EH approval.
 - DOE 5820.2A: Includes disposal and storage requirements for radioactive waste. These have been met for the contaminated portions of the copper and if recycle is not approved will be

implemented for the remaining fractions (140 metric tons).

- o **The State is aware of and agree to proposed action:**
 - The State has conducted confirmatory measurements and confirmed the DOE/LBL analyses.
 - The copper meets California Title 17 part 30104 requirements and was declared exempt from 10 CFR part 61 requirements.
- o **Release is consistent with NRC practice:**
 - NRC requirements: NRC (or its agreement states) conducts case-by-case reviews to determine if material can be released. LBL/DOE, through coordination with the State, have met these requirements.
- o **Release is consistent with international guidelines**
 - IAEA Safety Series (0.01 mSv [1 mrem] in a year and 1 person-Sv [100 mrem])
 - Complies with Draft (12/93) Clearance levels for solid materials (0.3 Bq/g [8 pCi/g])
 - Individual country's have standards that range from 0.3 Bq/g to 74 Bq/g for recycle.
- o **Rationale to approve proposed action:**
 - Cost effective.
 - Technical analysis is sound and all current DOE requirements met. LBL has been very responsive to EH staff in preparing analyses.
 - Comparable to NRC releases.
 - State agreement.
 - Hypothetical health impacts low.
 - Hypothetical adverse environmental impacts averted.

memorandum

DATE: 11/11 = 6 1994

REPLY TO
ATTN OF: Energy Research

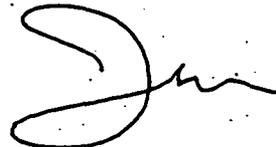
SUBJECT: Approval for Release from Lawrence Berkeley Laboratory Copper with Induced Volume Activation Under Department of Energy Order 5400.5

TO: Tara O'Toole, Assistant Secretary for Environment, Safety and Health

The Office of Energy Research proposes to recycle, for unrestricted use, 140 metric tons of slightly activated copper that is currently stored at Lawrence Berkeley Laboratory, Berkeley, California. This recycling proposal is consistent with the Department's August 20, 1992, Policy on Waste Minimization and Pollution Prevention. DOE Order 5400.5, Section I.5c(6), requires the approval of the Assistant Secretary for Environment, Safety and Health for release of material with volume activation for unrestricted use. I am hereby requesting your approval for the release of this material for recycling.

Attached to this request are an ALARA analysis, which was coordinated with EH-232, and an approved Environmental Assessment and Finding of No Significant Impact, which were approved by your Office for this recycling proposal on June 20, 1994. These documents provide the basis for the conclusion that the proposal will have no impact on workers, the public, or the environment, and that the proposal is on balance, beneficial, and cost-effective.

If there are any questions on this proposal, please direct them to Fred Koomanoff at 3-3298.



James F. Decker
Deputy Director
Office of Energy Research

Attachments

cc w/attachments:
T. Adduci, Oakland Operations Office

United States Government

Department of Energy

memorandum

DATE: June 20, 1994

REPLY TO
ATTN OF: Office of NEPA Oversight:Jessee:6-2410

SUBJECT: Review of the Environmental Assessment (EA) for the Recycling of Slightly Activated Copper at Lawrence Berkeley Laboratory (DOE/EA-0851)

TO: Martha A. Krebs
Director
Office of Energy Research

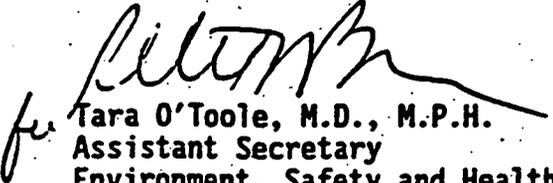
This is in response to an October 12, 1993, memorandum from James K. Farley, the Energy Research (ER) National Environmental Policy Act (NEPA) Compliance Officer (NCO), requesting approval of the subject EA. The EA was originally transmitted to our office via Dr. Happer's memorandum of February 18, 1993, requesting authorization to conduct State/tribal coordination of the EA with the State of California. Substantial analytical errors regarding inconsistent characterization of the irradiated copper and lack of evaluation of multiple exposures to the public were noted and discussed with your staff. The ER NCO transmitted a revised EA on June 10, 1993. The Office of NEPA Oversight (EH-25) authorized State/tribal coordination with provisos on July 27, 1993; copies of the revised EA were provided to the State of California on August 13, 1993. The State responded with no comment on September 22, 1993.

The Office of Environment, Safety and Health has reviewed the revised EA in accordance with our responsibilities under DOE Order 5440.1E regarding compliance with NEPA. Based on my staff's review and their recommendation, and after consultation with the Office of General Counsel, I have determined that the proposed action is not a major Federal action significantly affecting the quality of the human environment within the meaning of NEPA and its implementing regulations (40 CFR Parts 1500-1508). Therefore, the preparation of an environmental impact statement is not required, as described in the attached FONSI.

Accordingly, the EA is approved and I have signed the accompanying FONSI. The FONSI does not need to be published in the Federal Register since this is not an action with effects of national concern. However, the local public should be notified of the availability of the EA and FONSI in accordance with 40 CFR 1506.6 and DOE Order 5440.1E.

Please note that approval of the EA does not constitute approval for Release of Property Having Residual Radioactive Material under DOE 5400.5 Section II.5c(6), regarding radiation protection of the public and the environment. The request for EH-1 approval should be made separately.

Please provide the Office of NEPA Oversight with an electronic version of DOE/EA-0851, five hardcopies and a record of distribution of the EA and FONSI.


Tara O'Toole, M.D., M.P.H.
Assistant Secretary
Environment, Safety and Health

Attachment

cc: James K. Farley, ER-8.2
NEPA Compliance Officer

Tony Adduci, NEPA Compliance Officer
San Francisco Operations Office

U.S. Department of Energy
Finding of No Significant Impact
for the
Proposed Recycling of Slightly Activated Copper
at
Lawrence Berkeley Laboratory, Berkeley, California

AGENCY: U.S. Department of Energy

ACTION: Finding of No Significant Impact (FONSI)

SUMMARY: The Department of Energy (DOE) has prepared an Environmental Assessment (EA), DOE/EA-0851, evaluating the total impacts associated with the release of 140 metric tons (310,300 pounds) of high-purity activated copper to the open market for unrestricted reutilization. The $\frac{1}{2}$ " x 4" x 7' strips of copper coil windings were activated from bombardment with neutrons and other high energy particles during operation of the 184-Inch Cyclotron in use at Lawrence Berkeley Laboratory (LBL) from 1940-1988. This resulted in the internal production of radionuclides, primarily cobalt-60; the irradiated copper currently has an average residual activity of 3 picocuries per gram. The EA addresses both the potential environmental impacts from the activities encompassed within the proposed action, such as transportation, and incremental impacts when added to other actions such as air emissions from reprocessing. Assessment of the human health impacts focuses on radiological doses and protection of workers and the public. Alternatives were developed to meet DOE waste minimization requirements (DOE Orders 5400.1 and 5820.2A), and are consistent with other requirements for hazardous, radioactive, and mixed waste transportation, storage, treatment and disposal.

Based on the analyses in the EA, the DOE has determined that the proposed action does not constitute a major Federal action significantly affecting the quality of the human environment within the meaning of the National Environmental Policy Act of 1969 (NEPA). Therefore, the preparation of an Environmental Impact Statement is not required.

PUBLIC AVAILABILITY: Copies of this EA (DOE/EA-0851) are available from:

Terry Vaeth, Acting Manager
San Francisco Field Operations Office
U.S. Department of Energy
1301 Clay Street
Oakland, CA 94612-5208
(510) 637-1800 or -1801

For further information regarding the DOE NEPA process, contact:

Carol M. Borgstrom, Director
Office of NEPA Oversight
U.S. Department of Energy
1000 Independence Avenue, SW
Washington, D.C. 20585
(202) 586-4600 or (800) 472-2756

DESCRIPTION OF THE PROPOSED ACTION:

The proposed action is to recycle 140 metric tons of irradiated copper coil windings from the decontamination and decommissioning of the LBL 184-Inch Cyclotron. The activated copper would be released to the open market for unrestricted reutilization. It is probable that the copper would require reprocessing (i.e., re-melting) before reutilization. The activated copper could be melted with copper from other sources at the discretion of the purchasing reprocessor.

At least six domestic reprocessors of copper scrap metal have been identified in the eastern, midwestern and southwestern U.S. However, due to foreign and domestic copper market conditions, the industry practice for west coast scrap metal dealers is to ship the material directly to reprocessors in China or Taiwan. At this writing, the market value of copper scrap metal is estimated to be \$0.80/lb (i.e., \$246,960 for 140 metric tons).

ALTERNATIVES:

Five alternatives were considered: (1) reutilization at another DOE facility, (2) sale or gift of the activated copper to a foreign government, (3) disposal at a local sanitary landfill, (4) disposal at the Hanford Low-Level Waste Burial Site, and (5) no action.

Reutilization at Another DOE Facility:

The activated copper would be transported to the Scientific Ecology Group (SEG), a licensed radioactive scrap metal reprocessing facility located in Oak Ridge, Tennessee, for subsequent reutilization at another DOE facility. Safe operation of particle beam accelerators, synchrotrons and colliders requires many tons of high-density shielding. If reprocessed at the SEG facility, the activated copper coil windings from the BNL 184-Inch Cyclotron would be melted together with metals from other sources to meet shielding specifications for various DOE high-energy physics programs.

Sale/Gift to a Foreign Government:

The Institute of High Energy Physics in the People's Republic of China (PRC) has expressed interest in obtaining the copper for reutilization in synchrotron

accelerators. The activated copper would be transported by truck to the Port of Oakland from Emeryville, California, using licensed solid waste transporters in compliance with LBL procedures. The activated copper would then be transferred to a steamship carrier for transport to the receiving country. The copper would meet requirements for international shipping and would not require any special packaging for transport.

Disposal at a Sanitary Landfill:

The activated copper could be buried as ordinary non-radioactive waste at a designated California sanitary landfill. In a November 10, 1989, letter, the California Department of Health Services (DHS) independently validated the assumptions and calculations utilized by DOE in estimating the dose assessment for the recycling of the activated copper, and added estimates of the total potential dose resulting from reuse in various consumer products. The California DHS recognized that the fate of the recycled copper would not be immediate burial, but eventual disposal in a public landfill after its utility is diminished. The California DHS determined that the public health and safety risk from reutilization of the activated copper was acceptable under the practice of risk based calculations. The State determined that the copper meets the requirements of Title 17 part 30104, declared an exemption from licensed burial requirements under 10 CFR 61, and deemed that the copper may be disposed of as ordinary non-radioactive waste or recycled at DOE's option. A designated landfill in Livermore, California, will accept the activated copper.

Disposal at the DOE Hanford Low-Level Waste Burial Site:

The activated copper would be transported to and disposed of as low-level radioactive waste at the DOE Hanford Low-Level Waste Burial Site in Richland, Washington.

No Action:

The no action alternative is continued outdoor storage of the activated copper in 32 wooden crates under current LBL management practices at a leased warehouse in Emeryville, California. After 53 years of storage, the copper would still be radioactive, but at levels undetectable by current analytical techniques. Disposal of the radioactive copper would remain an eventuality.

ENVIRONMENTAL IMPACTS:

Human Health - Public

The radiological impacts to public health from the reutilization of the activated copper in home wiring/plumbing, in electronic components, and in jewelry were conservatively estimated. A maximum cumulative population dose equivalent of 0.5 person-mrem over 30 years was estimated to result from reuse of the copper in home wiring/plumbing; a maximum individual lifetime dose equivalent of 0.15 mrem was estimated to result from copper jewelry. An additional one time dose equivalent of 0.003 person-mrem to the public would result from transportation of the copper to reprocessing facilities. These dose equivalents correspond to an additional population fatal cancer risk of 3×10^{-7} , a maximum individual fatal cancer risk of 8×10^{-8} , and an additional one time population fatal cancer risk of 2×10^{-9} , respectively. In these and other respects that were

evaluated, radiation effects on public health from reuse of the copper would be inconsequential.

Human Health - Worker

The individual and cumulative radiological impacts to workers from reutilization of the activated copper were estimated for transportation and reprocessing and found to be inconsequential (risk of 2×10^{-8} or less).

Air Emissions

While the proposed action would result in particulate air emissions (approximately 640 pounds), the proposed action would have an overall beneficial impact on air emissions as a result of the indirect effect of averting larger particulate and sulfur dioxide emissions from mining, milling and smelting operations of copper from raw ore.

Water Quality

Water is not used directly in the reprocessing (i.e., recycling) of copper scrap metal. Wastewater, however, is produced indirectly from the cooling of furnaces, machinery and casting operations, and from boilers associated with power plants. Generally, such water is recirculated through cooling towers for reuse, and a small portion is discharged as blowdown. Any blowdown would meet EPA standards.

Land Use

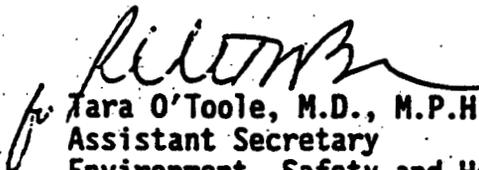
Although recycling of the activated copper would result in the generation of wastes (6,000 to 15,000 pounds), the proposed action may avert the

comparatively greater generation of wastes associated with the mining, milling, concentrating, and refining. In addition, the variety of solid and liquid waste streams generated from the three-stage refining process may be avoided.

DETERMINATION:

Based on the analyses in the EA, the DOE has determined that the proposed release of 140 metric tons of irradiated copper coil windings for unrestricted reutilization does not constitute a major Federal action significantly affecting the quality of the human environment within the meaning of the National Environmental Policy Act of 1969. Therefore, an Environmental Impact Statement on the proposed action is not required.

Issued in Washington, D.C., this 20th day of June, 1994


Tara O'Toole, M.D., M.P.H.
Assistant Secretary
Environment, Safety and Health

memorandum

VN
CH

DATE: APR 2 1993

REPLY TO
ATTN OF: EH-232

SUBJECT: ALARA analysis and approval of action to recycle cyclotron copper.

TO: J. Farley, ER-8.2

EH-232 has reviewed the draft EA for the subject action and provided comments to EH-25 for the consolidation with their comments. As you know, approval for the action must be requested under DOE 5400.5 under section IL5.c(6). The EA is generally acceptable as the background document for that request (assuming the comments we provided are addressed). However, we believe the request will be clearer and more concise if a specific ALARA summary is included with the request. Normally we would request that the field prepare the ALARA summary. However, all the information needed was well presented in the draft EA and to avoid unnecessary review of a draft ALARA document, we have prepared what we feel is an acceptable version of the ALARA summary document. You may use this document as part of your request or if you prefer, send your request with the EA and EH-232 will send the attached document with its recommendation. If you choose the latter please provide me comments on the summary report.



Andrew Wallo III
Director
Air, Water and Radiation Division

cc:
R. Strickler, EH-25
M. Kleinrock, EM-22
L. Stevens, EM-331

EXAMPLE - High Explosives Authorized Limits

memorandum

DATE: AUG 29 1995

REPLY TO
ATTN OF: Office of Environmental Policy and Assistance: Wallo: 202-586-4996

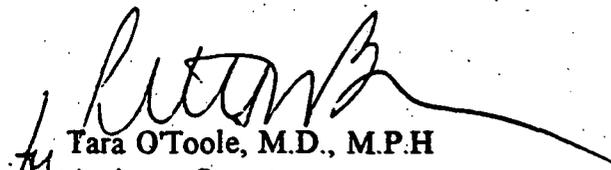
SUBJECT: Approval of Authorized Limit and Measurement Protocol for Release and Recycle of High Explosives Containing Residual Tritium.

TO: Nicholas S. Dienes, Assistant Manager for Technical Management Operations
Albuquerque Operations Office

My staff have reviewed your transmittal of May 30, 1995, and subsequent material provided by your staff pertaining to your request for approval of authorized limits for radiation protection and associated measurement protocol for the release and recycle of high explosives containing residual tritium. The propose criteria and the dose assessment are in compliance with the requirements of Order DOE 5400.5, "Radiation Protection of the Public and Environment," and have been adequately reviewed under the ALARA process to ensure that they are as low as is practicable.

We are aware that Department and contractor staff are completing coordination with the appropriate State representatives in Texas and California. On the condition that this coordination is completed and the proposed actions are acceptable to the States, the subject authorized limits and measurement protocol are approved for implementation. Records of the releases should be maintained consistent with the applicable requirements in DOE 5400.5 including Section II.5.c.(5).

If you have any questions regarding this approval or require further assistance please contact Mr. Andrew Wallo of my staff and (202)586-4996, email - andrew.wallo@hq.doe.gov.


Tara O'Toole, M.D., M.P.H.
Assistant Secretary
Environment, Safety and Health



EH-41 Routing and Transmittal

Date: 8/14/95

Name	Initials	Date
To: <i>[Signature]</i> Raymond Pelletier, EH-41 Raymond Berube, EH-4 Peter Brush, EH-1 Tara O'Toole, EH-1	<i>AWB</i> <i>UB</i>	8/15/95 8/29/95

Subject: Approval of authorized limits for release/recycle of high explosives containing residual tritium.

Consistent with the requirements of DOE 5400.5, Albuquerque Operations has requested approval of authorized limits for release of high explosives removed during the dismantlement of excess nuclear devices. The Department requires that authorized limits for release or recycle of material containing residual radionuclides be developed on the basis of the "ALARA" process and be as far below the applicable dose limit or dose constraint as is practicable. In addition, use of the limits is subject to appropriate coordination with and acceptance by state regulators.

We have reviewed the background material and worked with AL and Pantex staff and find the authorized limits and process acceptable.

SUMMARY:

The primary mission of Pantex Plant is to dismantle nuclear weapons that are no longer needed for the defense of the United States. These dismantlement operations produce high explosive (HE) material that may be slightly contaminated with tritium. Although much of the tritium contamination is on or near the surface of the HE, some of the contamination may have penetrated through the depth of the HE main-charges. Tritium diffusion into HE is similar to its diffusion into other materials such as metals and plastics.

Pantex Plant proposes to make this HE available for commercial use, rather than processing the HE on site by regulated open burning/open detonation. It is anticipated that the recycled HE will be sold to industrial users in the mining industry.

Recycling of this HE into commercial use will minimize waste and is the most cost effective and environmentally sound disposition of the explosive.

AL has proposed an authorized limit 2×10^{-3} microcuries of tritium oxide per gram of HE

($\mu\text{Ci HTO/g HE}$) be used to release the recycled main-charges for commercial use. They proposed that individual HE main-charges be screened for removable beta-gamma ($\beta\text{-}\gamma$) surface contamination and that a criterion of 1,000 disintegrations per minute over an area of 100 square centimeters ($\text{dpm}/100 \text{ cm}^2$) for removable ($\beta\text{-}\gamma$) surface contamination be used to verify compliance. The use of these criteria provides the most environmentally sound and cost-effective approach for the recycling of main-charge explosive.

Individual dose and dose limits:

Individual doses to members of the public or workers resulting from the proposed action is estimated to be less than 0.0001 mrem per year. This is more about a million times lower than the DOE all sources limit of 100 mrem/year and the upper dose constraint for release of property (30 mrem in a year). It is clearly insignificant in comparison to natural background doses of 300 mrem/year to a resident of the United States. (Individual risks associated with this incremental dose is well below the frequently use 10^{-4} to 10^{-6} lifetime risk range for cancer incidence used in many regulatory actions.)

Population dose and ALARA:

The radiological impacts from the recycle option are negligible. The collective EDE received by the estimated 75 members of the public, including Pantex Plant workers, for the recycling option is about 1.5×10^{-3} person-mrem. If one were to assume that even these low dose levels can cause cancer, the potential incremental fatal cancers associated with this population dose is about 0.000000001 (there is effectively zero risk of incremental cancers resulting from potential radiological exposures). For comparison, the radiological exposure from natural radiation background, i.e., 360 mrem per year for the "average" person in the United States, these members of the public would be collectively exposed to an EDE of 2.7×10^{-4} person-mrem.

It is estimated that the recycle of these high explosives will save the Department approximately \$1,000,000 annually.

This action will also reduce the quantity of high explosive currently under going open burning or detonation by 50,000 pounds per year.

RECOMMENDATION: Given the low doses/risks and the high pay back for this recycling effort, it is recommended that EH approve the authorized limits.

AL/AL contractors have been coordinating the action with the states and have general agreement; however, the approval should be on condition that the states find the action acceptable.

 Andrew Wallo EH-232 Air, Water and Radiation Division	DOE Forrestal BLDG - GA098 Phone, 202-586-4996 Fax, 202-586-3915 E-mail, andrew.wallo@hq.doe.gov
--	--

memorandum

DATE: **JUL 12 1995**
REPLY TO: **EPD:FHS**
ATTN OF:
SUBJECT: **Release Criteria for Volume Tritium Contamination in High Explosives**

TO: **Andrew Wallo, Director, Air, Water, and Radiation Division, EH-412, HQ**

Your comments of April 26, 1995, have been incorporated in the attached report, which should now be suitable for approval by the Office of Environment, Safety and Health (EH-1). Additionally, the following summary is provided to complete the As Low As Reasonable Achievable (ALARA) determination requirement of DOE Order 5400.5.

The purpose of this report was to request the exemption to release volume contaminated high explosives (HE) to be recycled. If approved, Pantex will recycle about 50,000 pounds of HE per year using 1000 disintegrations per minute per 100 centimeters square ($1000 \text{ dpm}/100 \text{ cm}^2$) as a standard for release.

This recycle effort is generally beneficial to the environment, saves the DOE resources through the sale of the explosives, and will only cause insignificant incremental doses to the public even under conservative assumptions. The dose to the maximally exposed worker or member of the public was estimated to be only about 5.0×10^{-3} millirem (mrem) and the collective doses are estimated to be less than 2.0×10^{-3} person-mrem from the activity at the proposed standard. The incremental doses associated with the activity are seven orders of magnitude lower than background and are far below all radiation protection requirements. Because potential doses are already extremely low, lowering the limit will have no impact on public health or welfare but would increase implementation difficulties substantially. Therefore, the proposed limit represents the appropriate ALARA-based value for this activity.

Pantex and the DOE Amarillo Area Office are coordinating this action with the appropriate state regulators.

Andrew Wallo

2

JUL 12 1995

This memorandum, with the revised report and disk, should address all the comments submitted by EH-41. If you have any questions, please contact Frank H. Sprague at (505) 845-4340.



Constance L. Soden
Director
Environmental Protection Division

Attachment

cc w/attachment (letter only):
M. S. Bange, WMD, AL

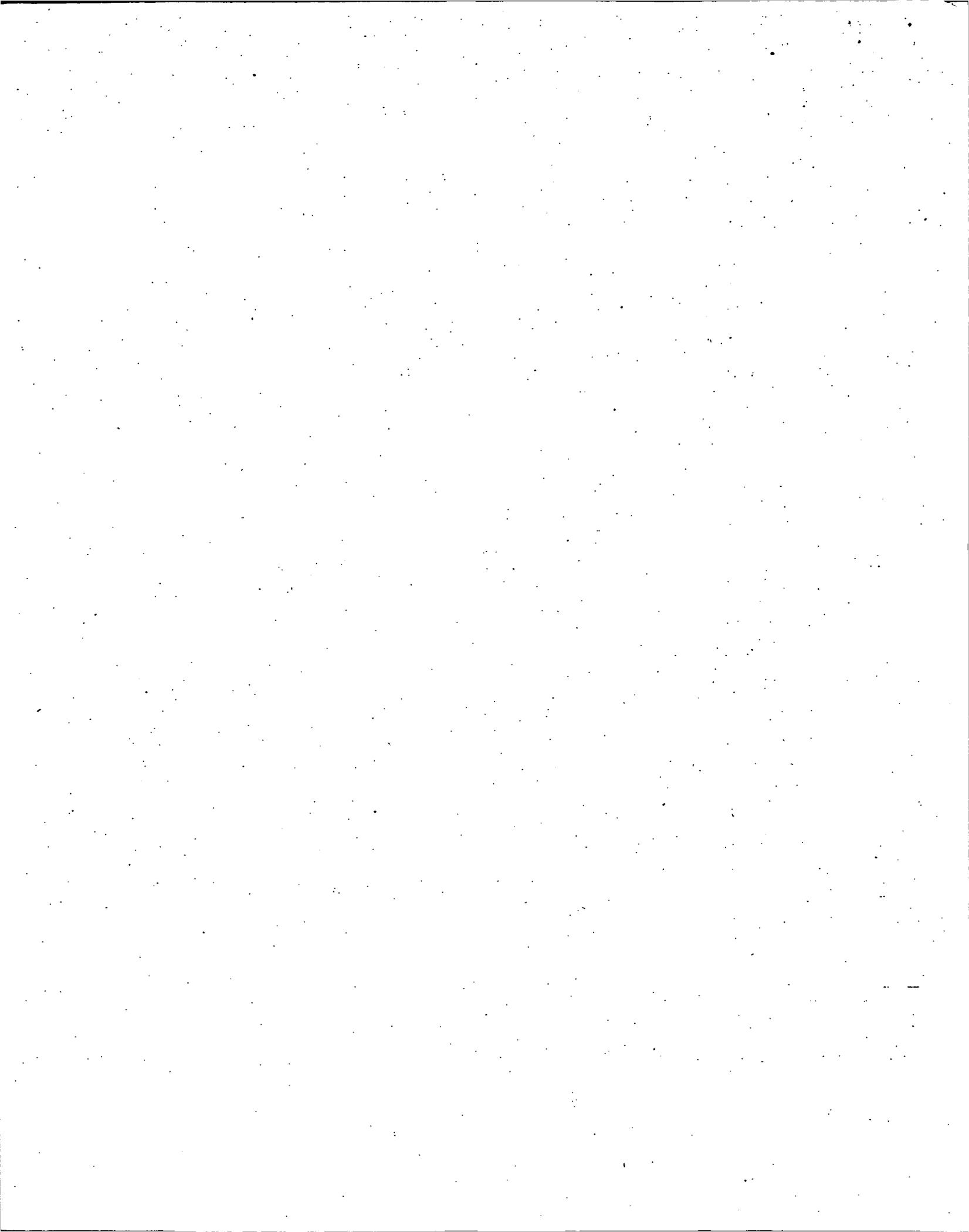
Analysis to Support
an
As Low As Reasonably Achievable
(ALARA) Determination
for
Disposition of Tritium-Contaminated High Explosives

July 7, 1995

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July 7, 1995



I. INTRODUCTION

There are two options being considered for the disposition of high explosive (HE) main-charges that are produced as a result of the dismantlement activities for nuclear weapons at Pantex Plant. The current method involves removing the HE part and treating the HE through open burning/open detonation (OB/OD) at the Pantex Plant's Burning Ground. This option will be referred to as the "open burning" option. The second option is to recycle the HE by making it available to commercial users. This option will be referred to as the "recycle" option.

This analysis considers the following factors for each option:

1. Radiation doses and risk,
2. Economic factors,
3. Operational constraints, and
4. Societal impacts and perceptions.

II. EVALUATIONS OF OPTIONS

A. CURRENT ACTION -- OPEN BURNING/OPEN DETONATION

Currently, Pantex Plant disposes of the HE main-charges by treating them via OB/OD at the Burning Ground. The Burning Ground is being operated under a Resource Conservation and Recovery Act (RCRA) interim strategy permit and written grant of authority issued by the Texas Natural Resource Conservation Commission (TNRCC) of the state of Texas. This activity releases small quantities of carbon monoxide (CO), carbon dioxide (CO₂), oxides of nitrogen (NO_x), fluorides (F⁻¹), chlorides (Cl⁻¹), and airborne tritium in the form of HTO. All releases of CO, CO₂, NO_x, F⁻¹, and Cl⁻¹ are in full compliance with applicable regulations. In addition, the release of the airborne tritium activity is in full compliance with Title 40 Code of Federal Regulations (CFR), Part 61, Subpart H "Environmental Protection Agency Regulations on National Emissions Standards for Hazardous Air Pollutants - National Emissions Standards for Emissions of Radionuclides Other Than Radon From Department of Energy Facilities."

Over 50,000 pounds of HE, containing an estimated 0.144-curies (Ci) of tritium in the form of tritiated water (HTO), were treated of by OB/OD during 1993. The OB/OD activities occurred at various times throughout 1993. The steps involved in the OB/OD option are listed below:

1. Transportation to the Burning Ground, and
2. Burning/detonation.

To determine the potential radiological exposures to the maximally exposed off-site person and to the nearby population, the annual and collective effective dose equivalents (EDEs) were calculated using the CAP-88 computer code. Although the OB/OD activities occurred at various times throughout 1993, a single diffuse HTO source term was assumed. The HTO source term was determined by multiplying the number of HE parts burned/detonated by an assumed HTO activity of 100 microcurie HTO per HE part ($\mu\text{Ci HTO/HE part}$).

The annual EDE to the nearest maximally exposed member of the public, located about 900 meters (m) northeast of the Burning Grounds, was 5.7×10^{-5} millirem (mrem). The collective EDE to the nearby population, estimated to be about 275,000 residing within a radius of 50 miles, was 1.0×10^1 person-mrem.

The disadvantages to this practice are as follows:

1. The OB/OD operation is permitted by the state of Texas under an interim agreement; this status may change at any time.
2. The OB/OD assures that all the tritium is converted to HTO form, resulting in the immediate release of all of the tritium into the environment.
3. The treatment of this HE is a loss of a valuable asset. If this material were to be replaced, there would be potential impacts on worker safety, environmental, energy, and resources.

B. PROPOSED ACTION -- RECYCLING OPTION

Pantex Plant, in conjunction with Los Alamos and Lawrence Livermore National Laboratories (LANL/LLNL), is proposing that all HE below a specified bulk tritium contamination of 2×10^3 microcuries of tritium oxide per gram of high explosives ($\mu\text{Ci HTO/g HE}$) be recycled to a commercial HE manufacturer for use in commercial explosives. The steps involved in the recycling option are listed below:

1. Size and configuration reduction to wet powder,
2. Analysis and packaging,
3. Storage,
4. Transportation to commercial HE manufacturer,
5. Storage of PBX powder,
6. Processing/fabrication at commercial HE manufacturer,
7. Packaging at commercial HE manufacturer,
8. Transportation to commercial end-user, and
9. Commercial use of explosive.

The first three steps of the recycling option have been (and will be if this option is approved) performed at Pantex Plant. Although no detailed exposure analysis was performed for these three steps, it is anticipated that radiological exposures to

individual workers would be comparable to those estimated for the storage, unpacking, and blending operations examined in scenarios #1, #2, and #4 in Appendix C of the main report, i.e., "Health-Based Release Criteria for Bulk Tritium-Contaminated High Explosives". Urinalyses data for workers who are currently performing such operations at Pantex Plant show no tritium above detection limits, i.e., 1.0×10^{-1} mrem.

Steps 4 through 9 of the recycling process are schematically represented in Figures 1 and 2 and in Processes 1 through 3. The analyses in Appendix C of the main report examined possible scenarios by which members of the public, including workers at Pantex Plant, the commercial HE manufacturer, and the explosives end-user, might receive a radiological exposure from the use of the recycled HE. It should be noted that these scenarios do not necessarily match the step-by-step process(es) shown in the flow charts, i.e., Figures 1 - 2 and Processes 1 - 3. The scenarios are intended to represent exposure possibilities that might exist at one or more of the processing steps.

The analyses in Appendix C of the main report examined two cases, i.e., a "worst-case" and a "realistic" case. Both of these cases used very conservative assumptions. The "worst-case" established upper limits of doses that might occur. For the purpose of this "realistic" analysis, a set of calculations was performed for each of the exposure scenarios using assumptions that were as realistic as possible. The exposure scenarios used were essentially the same; however, the assumptions and input parameters were fine-tuned to reflect anticipated exposure conditions during the "life-cycle" of the HE during "normal" operational conditions. The differences in the assumptions for the various scenarios used are summarized in Table 1.

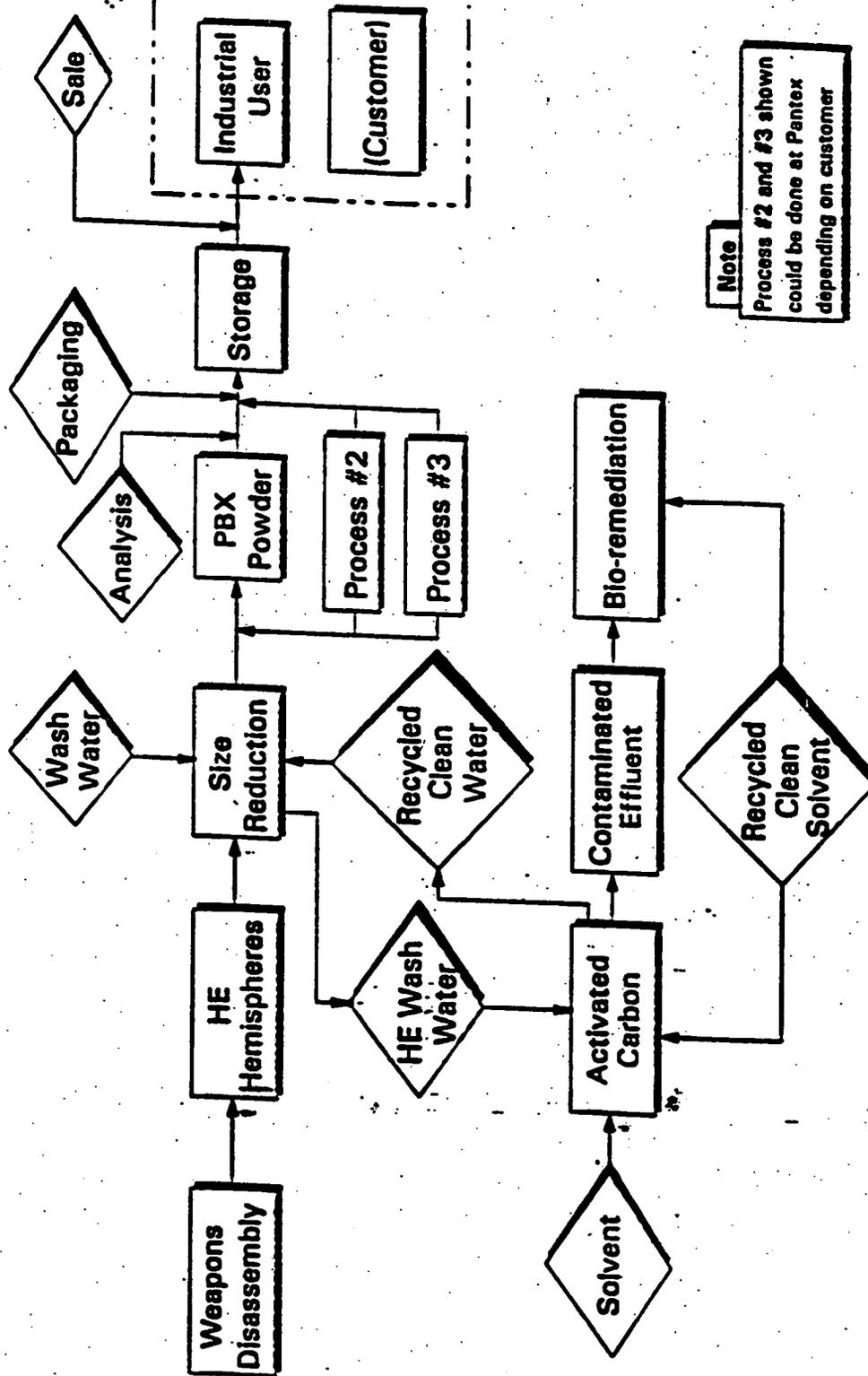
The maximum individual and collective EDEs from the radiological impacts for the recycle option are negligible. A comparison of the EDEs received by the maximally exposed members of the public, including Pantex Plant workers, for each scenario shown in Appendix C of the main report, is presented in Table 2. It should be noted that for this analysis the occupationally exposed workers at Pantex Plant are considered to be members of the public. Therefore, no attempt was made to exclude their occupational exposures from this option. For the "realistic" analysis, the collective EDE to the members of the public for the same scenarios is summarized in Table 3.

For the recycle option, the maximum EDE received by a worker at Pantex Plant was 3.8×10^{-5} mrem; it was 4.9×10^{-5} mrem for the maximally exposed off-site member of the public. These annual EDEs are much less than 1% of the applicable primary dose limit for members of the public, i.e., 100 mrem, promulgated in DOE 5400.5- "Radiation Protection of the Public and the Environment" and in Title 10 Code of Federal Regulation (CFR) Part 20 "Standards for Protection Against Radiation."

They are also well below the 30 mrem in a year DOE dose constraint for release of property.

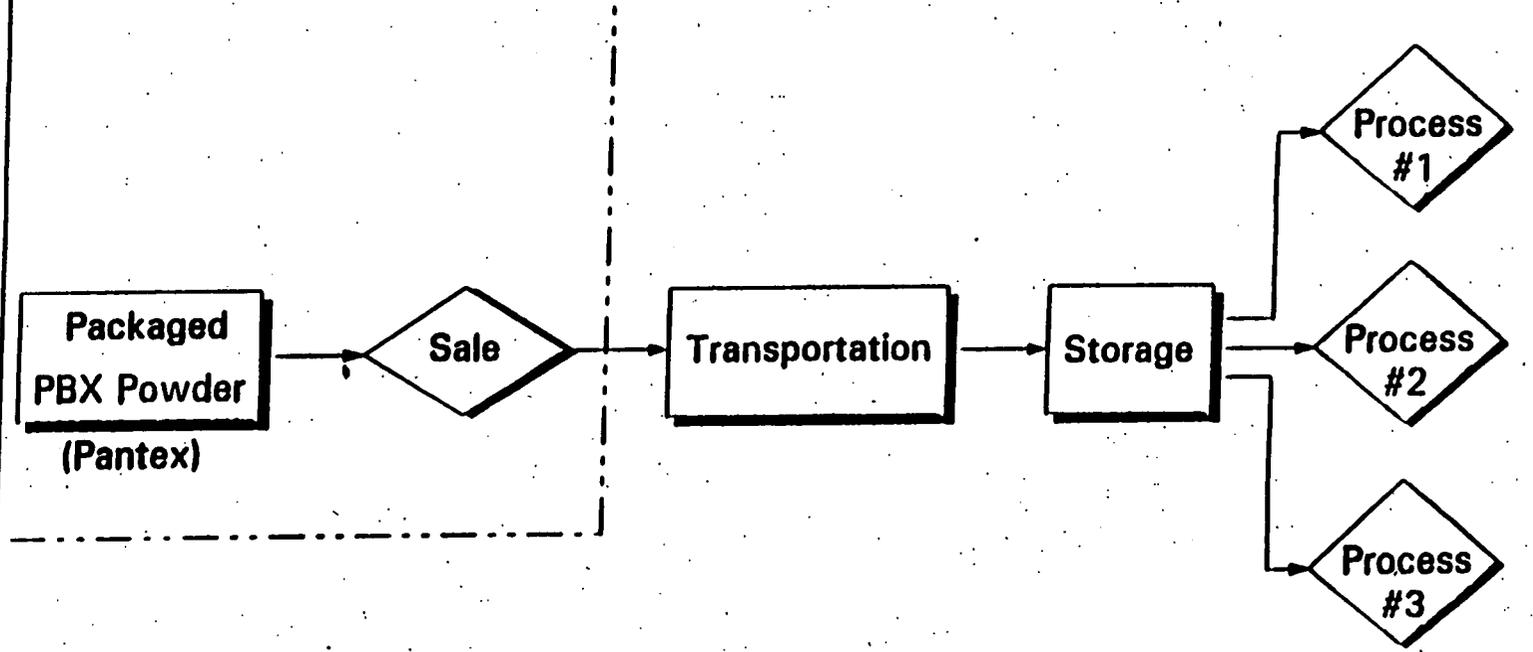
It is estimated that about 75 members of the public, including Pantex Plant workers, could receive some radiological exposures as a result of the recycle option. As shown in Table 3, the total collective effective dose equivalent for this option is about 1.5×10^{-3} person-mrem. By comparison, this same population is collectively exposed to $2.7 \times 10^{+4}$ person-mrem, assuming an "average" natural background radiation of 360 mrem.

Figure 1: Pantex Plant Operations

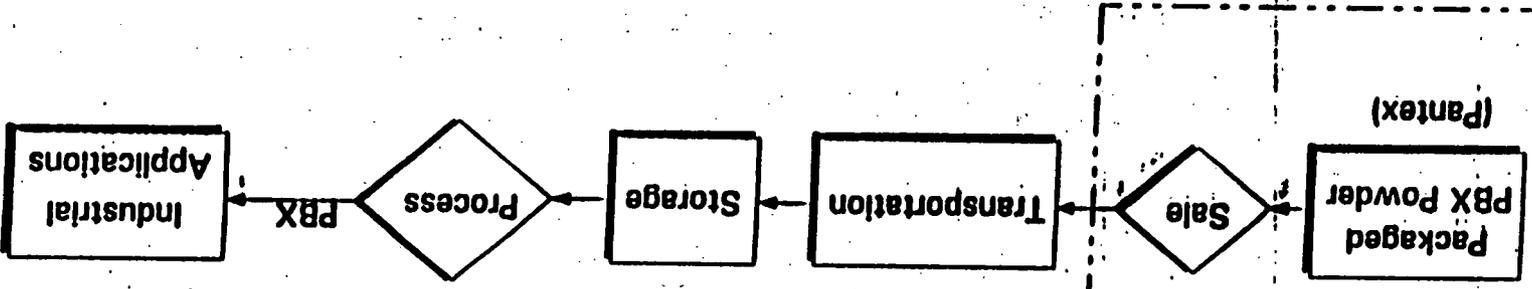


Note
Process #2 and #3 shown
could be done at Pantex
depending on customer

Figure 2: Industrial User Operations

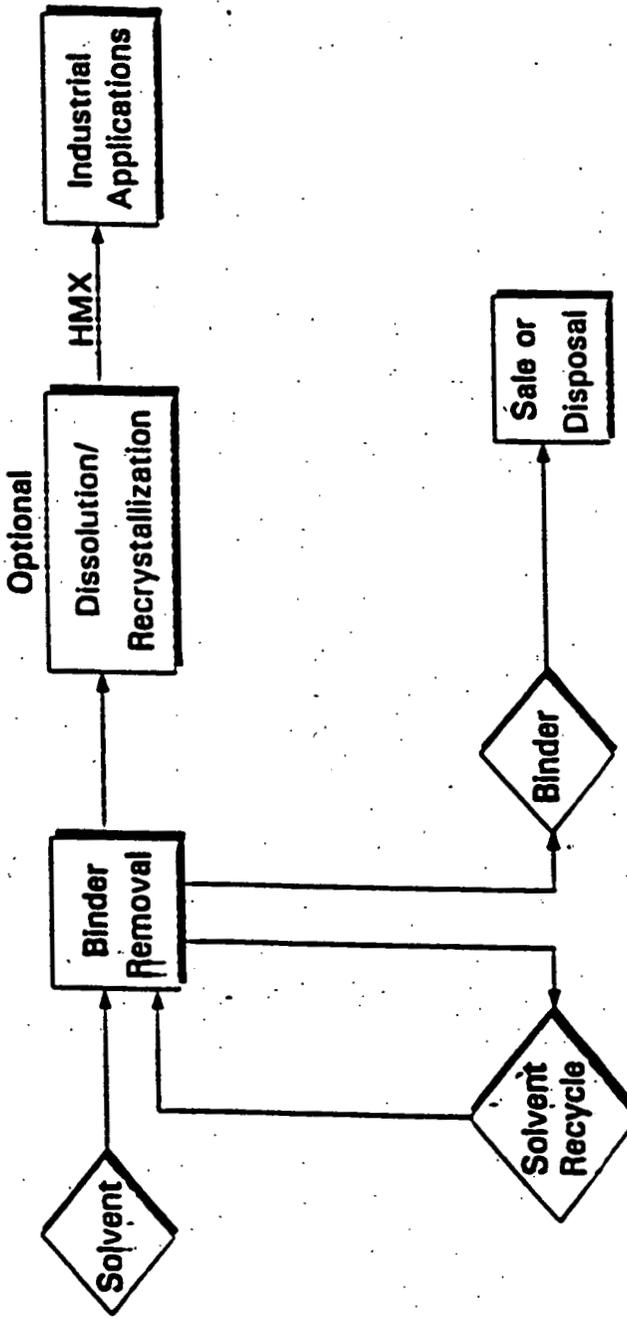


Process #1



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Process #2



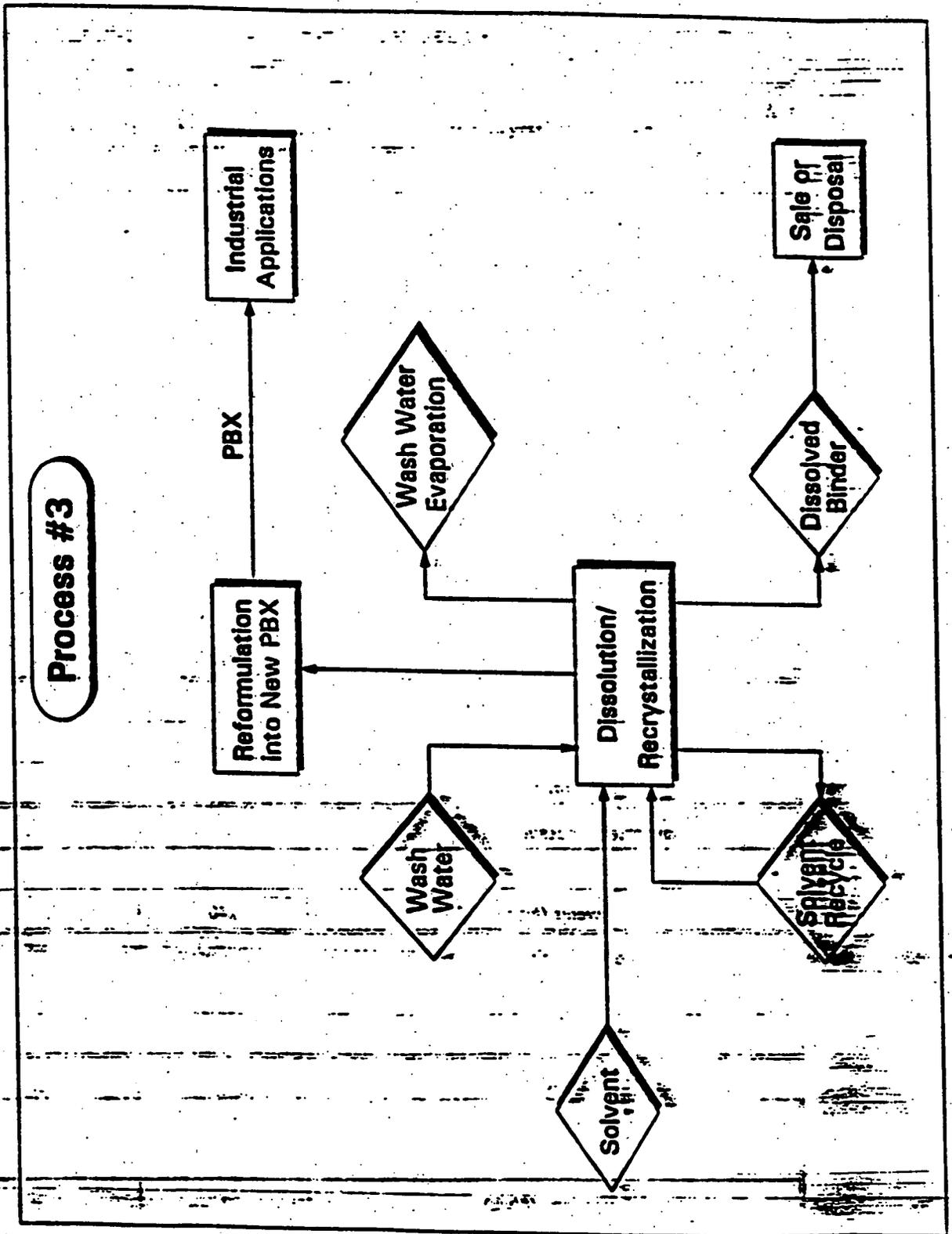


TABLE 1: SUMMARY OF PARAMETER CHANGES		
Scenario/Parameter	"Worst-Case"	"Realistic"
All Scenarios:		
Tritium ($\mu\text{Ci HTO/g HE}$)	2.0×10^{-4}	1.0×10^{-4}
Storage:		
Mass of HE Stored (lbs)	40,000	2,000
Ventilation Air Changes (#/d)	10	24
Exposure Period (hrs/d)	8	0.3
Manual Unpacking:		
Air Mixing Volume (m^3)	10	20
Bags of HE Opened (#/d)	10	1
Days Worked (d/yr)	240	40
Dust-Generating Operations:		
Airborne Dust Loading (mg/m^3)	10	1
Respiratory Protection Factor (RPF)	1	10
Exposure Period (hrs/d)	8	1
Days Worked (d/yr)	240	40
Adding Wet PBX to Slurry:		
Exposure Period (hrs/d)	8	0.33
Days Worked (d/yr)	240	100
Inadvertent Ingestion of Slurry:		
Quantity Ingested (g/d)	0.01	0.001
Events (#/yr)	240	12
Skin Exposure to Splashed Slurry:		
Exposure Time (min)	10	5
Days Worked (d/yr)	240	24
Dust Plume from HE Explosion:		
Mass of HE Used (lbs)	100,000	1,000
Wind speed (m/sec)	2	4.5
Explosions (#/wk)	1	7

NOTE: Please see Appendix C of the main report for details of the other parameters used in these analyses.

TABLE 2: DOSE EQUIVALENTS FOR ALL SCENARIOS

Scenario	"Worst-Case" (mrem)	"Realistic" (mrem)
A: Pantex Preparation (See Note)	6.8×10^3	3.8×10^5
1: Warehouse Storage	1.4×10^3	5.5×10^7
2: Unpacking of HE	2.1×10^3	8.7×10^6
3: Mixing/Crushing (Airborne)	4.6×10^4	4.6×10^6
4: Adding Wet PBX to Slurry	3.3×10^3	2.9×10^6
5: Ingestion of Slurry	9.2×10^6	4.6×10^7
6: Skin Splashed with Slurry	1.5×10^3	4.9×10^5
7: Dust Plume from Explosive Use	9.6×10^5	1.5×10^6

NOTE: With the exception of the "worst-case" dose equivalent, the values used for "Pantex Preparation" are simply the sum of the values for scenarios #1, #2, and #4 of Appendix C.

TABLE 3: COLLECTIVE DOSE EQUIVALENTS -- REALISTIC ANALYSIS

Scenario	Annual Individual EDE (mrem)	Sites	Workers per Site	Annual Collective EDE (man-mrem)
A: Pantex Preparation	3.8×10^6	1	10	3.8×10^6
1: Warehouse Storage	5.5×10^7	2	10	1.1×10^9
2: Unpacking of HE	8.7×10^6	2	10	1.7×10^8
3: Mixing/Crushing (Airborne)	4.6×10^6	2	5	4.6×10^7
4: Adding Wet PBX to Slurry	2.9×10^6	2	5	2.9×10^7
5: Ingestion of Slurry	4.6×10^7	2	5	4.6×10^8
6: Skin Splashed with Slurry	4.9×10^5	2	5	4.9×10^6
7: Dust Plume from Explosive Use	1.5×10^6	5	25	1.9×10^8
Total Collective EDE (person-mrem)				1.5×10^9

III. RADIOLOGICAL IMPACTS

The radiological impacts from the recycle or OB/OD options are negligible. It is estimated that about 75 members of the public will be directly involved with the recycle option; there are about 275,000 members of the public within a 50 mile radius of the OB/OD activities. The collective EDE for members of the public from the recycle option is estimated to be about 1.5×10^3 person-mrem. It is estimated to be about 1.0×10^1 person-mrem for the OB/OD option. If it is assumed that the "average" person in the United States receives about 360 mrem per year from background radiation, the members of the public for the recycle option could be collectively exposed to an EDE of $2.7 \times 10^{+4}$ man-mrem from natural background radiation. Similarly, the members of the public within a 50 mile radius of the OB/OD activities could be collectively exposed to an EDE of $9.9 \times 10^{+7}$ man-mrem from natural background radiation.

For low level exposure to tritium in the form of tritiated water, i.e., HTO, the use of Straume's "most probable risk" of cancer mortality, i.e., 8.1×10^5 per milliGray (mGy) [8.1×10^7 per mrem, assuming a quality factor (Q) of 1], is appropriate. Using this risk factor and the collective EDE for the OB/OD option, i.e., 1.0×10^1 person-mrem, a lifetime cancer mortality risk to a member of the public is estimated to be $8.7 \text{ E-}08$, i.e., 1 in 10 million. Likewise, using Straume's risk factor and the collective EDE of 1.5×10^3 person-mrem for the recycle option, a lifetime cancer mortality risk to a member of the public is estimated to be $1.2 \text{ E-}09$, i.e., 8 in 100 million. By comparison, the normal risk of cancer mortality from all carcinogenic sources is about 0.25 to 0.33, i.e., 1 in 4 to 1 in 3.

IV. ECONOMIC COSTS

The cost of the OB/OD option is based on the labor required to transfer the HE to the Burning Ground and to burn/detonate the HE. This cost is estimated to be \$9.75 per pound. Assuming 50,000 pounds of waste HE per year are treated by this option, it is estimated that Pantex Plant would incur a cost about \$487,500 per year for the OB/OD option.

For the recycling option, it is assumed that Pantex Plant recycles 50,000 pounds of HE per year. The assumed selling price of the recycled HE is estimated to be \$15.00 per pound. Therefore, the net annual cost benefit to Pantex Plant is estimated to be about \$1,000,000 per year. This net gain was calculated as follows:

$$\begin{aligned} \text{Net Gain/yr} &= 50,000 \text{ lbs/year} [\text{Sale value/lb} + \text{Avoided} \\ &\quad \text{Treatment Costs/lb} - \text{Recycle Cost/lb}] \\ \$1,000,000/\text{yr} &= [50,000\text{-lbs/yr}] [\$15/\text{lb} (\text{sale price}) + \$9.75/\text{lb} \\ &\quad (\text{treatment cost}) - \$4.75/\text{lb} (\text{recycle cost})] \end{aligned}$$

V. OTHER CONSIDERATIONS

There are other, less tangible, but nevertheless important, factors that should be taken into account when considering the costs, risks, and benefits of these options. The OB/OD option presents a number of potential problems in addition to the emission of HTO from this activity. There are also some chemical constituents, i.e., CO, CO₂, NO_x, F⁻¹, and Cl⁻¹, that are released during the OB/OD process.

The OB/OD of chemical and/or radioactive waste material, however innocuous, is distasteful to members of the public. If the OB/OD option becomes unavailable, disposal of the HE as radioactive (or perhaps mixed) waste would be very expensive.

Treatment of the HE by burning/detonating it is a loss of a valuable resource. Sales of recycled HE would be an important building block in the Pantex Plant Technology Transfer Program. It would also strongly support the Secretary of Energy's Economic Competitiveness Program. Also, the availability of recycled HE would reduce or eliminate the need for HE users to purchase this material from the People's Republic of China.

Finally, there are small but finite risks to the workers and to the environment, and resource and energy costs that would result from the chemical processing needed to manufacture replacement HE.

VI. SUMMARY AND RECOMMENDATIONS

The radiological dose equivalents and the economic costs to members of the public are summarized in Table 4 below.

Option	Maximum EDE to Individual (mrem)	Collective EDE to Population (person-mrem)	Economic Costs (\$1,000)
Open Burning	5.7×10^{-5}	1.0×10^{-1}	-487.5
Recycle/Sale	5.2×10^{-5}	1.5×10^{-3}	+1,000

The individual and collective EDEs for either the OB/OD or recycle options are negligible. However, the EDEs from the OB/OD option are likely to be less acceptable to the public perception than those from the recycling option. Economic costs, assuming that the OB/OD option remains available, strongly favor the recycling effort. With OB/OD option, it would cost Pantex Plant about \$487,500 per year to

dispose of 50,000 pounds of waste HE. If Pantex Plant were to recycle the same amount, it would realize a net gain of about \$1,000,000. This net gain includes a cost avoidance of about \$487,500 that would not have to be spent on the OB/OD option.

Therefore, Pantex Plant, in conjunction with LANL and LLNL, recommend that HE resulting from the dismantlement activities of nuclear weapons at Pantex Plant be released to the commercial sector. The HE would be released to the commercial sector only if the sampling and analysis of the HE showed that it had a bulk tritium contamination less than $2 \times 10^{-3} \mu\text{Ci HTO/g HE}$ and had removable tritium contamination less than 1,000 disintegrations per minute over an area of 100 square centimeters (dpm/100 cm²).

**HEALTH-BASED RELEASE CRITERIA
FOR
BULK TRITIUM-CONTAMINATED HIGH EXPLOSIVES**

July 7, 1995

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July 7, 1995

EXECUTIVE SUMMARY

The primary mission of Pantex Plant is to dismantle nuclear weapons that are no longer needed for the defense of the United States. These dismantlement operations produce high explosive (HE) material that may be slightly contaminated with tritium. Although much of the tritium contamination is on or near the surface of the HE, some of the contamination may have penetrated through the depth of the HE main-charges. Tritium diffusion into HE is similar to its diffusion into other materials such as metals and plastics.

Pantex Plant proposes to make this HE available for commercial use, rather than processing the HE on site by regulated open burning/open detonation. It is anticipated that the recycled HE will be sold to industrial users in the mining industry. The industrial users, depending upon their requirements, may require further processing of the HE into products required for their application(s).

Recycling of this HE into commercial use will minimize waste and is the most cost effective and environmentally sound disposition of the explosive. Pantex Plant, in conjunction with Los Alamos and Lawrence Livermore National Laboratories (LANL/LLNL), has developed dose-based release criteria for tritium-contaminated HE. This document presents the technical basis for release criteria and associated measurement techniques.

Pantex Plant, LANL, and LLNL jointly recommend that a release criterion of 2×10^3 microcuries of tritium oxide per gram of HE ($\mu\text{Ci HTO/g HE}$) be used to release the recycled main-charges for commercial use. We also recommend that individual HE main-charges be screened for removable beta-gamma (β - γ) surface contamination. It is recommended that the release criterion of 1,000 disintegrations per minute over an area of 100 square centimeters ($\text{dpm}/100 \text{ cm}^2$) for removable (β - γ) surface contamination, as promulgated in DOE 5400.5 "Radiation Protection of the Public and the Environment," be used to screen the main-charges. No recycled HE with an average bulk-tritium activity greater than $2 \times 10^3 \mu\text{Ci HTO/g HE}$ and/or removable (β - γ) surface contamination in excess of $1,000 \text{ dpm}/100 \text{ cm}^2$ would be released to the commercial market. The use of both criteria provides the most environmentally sound and cost-effective approach for the recycling of main-charge explosive.

The sampling protocol and methodology used by Pantex Plant have demonstrated that the removable β - γ surface release criterion of $1,000 \text{ dpm}/100 \text{ cm}^2$ surface contamination should be used as an indicator of bulk tritium activity in HE main-charges. Depth profile experiments have indicated that, if the removable tritium surface contamination is less than $1,000 \text{ dpm}/100 \text{ cm}^2$, the volume contamination at depth does not show a bulk tritium activity greater than $2 \times 10^3 \mu\text{Ci HTO/g HE}$. Assuming that the entire HE main-charge has bulk tritium activity at this value, all of the radiation exposure scenarios evaluated by this group indicate that the maximum annual effective dose equivalent (EDE) (H_p) for a member of the public is much less than 1 millirem (mrem). The primary dose limit from all radiation

sources and pathways for members of the public, as promulgated in DOE 5400.5 and in Title 10 Code of Federal Regulation (CFR) Part 20 "Standards for Protection Against Radiation," is 100 mrem annual EDE. Included in this primary dose limit is the exposure limits from all sources of DOE airborne emissions, i.e., 10 mrem, and all radionuclides in the drinking water pathway, i.e., 4 mrem.

Based upon the analysis to support an As Low As Reasonably Achievable (ALARA) determination, the maximum annual EDE received by a worker at Pantex Plant was 3.8×10^{-5} mrem while it was 4.9×10^{-5} mrem for the maximally-exposed member of the public i.e., industrial user. It should be noted that for this analysis the occupationally exposed workers at Pantex Plant are considered to be members of the public. Therefore, no attempt was made to exclude their occupational exposures from the proposed recycling option.

The radiological impacts from the recycle option are negligible. The collective EDE received by the estimated 75 members of the public, including Pantex Plant workers, for the recycling option is about 1.5×10^{-3} person-mrem. When compared to the radiological exposure from natural radiation background, i.e., 360 mrem per year for the "average" person in the United States, these members of the public would be collectively exposed to an EDE of 2.7×10^{-4} person-mrem.

If the assumption is made that some risk of radiation induced cancer occurs even at such low dose levels, the best estimate of the lifetime risk of fatal cancer for low level exposure to tritium in the form of HTO is 8.1×10^{-7} per millirad (mrad) [8.1×10^{-7} per mrem (assuming a quality factor of 1)]. Using this risk coefficient, the lifetime cancer mortality risk factor yields a collective risk of cancer mortality of 1.2×10^{-9} , i.e., 8 in 100 million. By comparison, the normal risk of cancer mortality from all sources is about 0.25 to 0.33, i.e., 1 in 4 to 1 in 3.

The Department of Transportation (DOT) has not imposed any special regulations for tritium at this low level. Other than the normal posting requirements for a contamination area (if the removable tritium contamination is greater than 10,000 dpm/100 cm²) and required bioassay sampling for tritium, special handling, labeling, and/or posting requirements are not necessary for work activities on HE main-charges.

The National Council on Radiation Protection and Measurements (NCRP) in NCRP Report No. 91 "Recommendations on Limits for Exposure to Ionizing Radiation," recommends recognition of a level of annual EDE below which further effort to reduce individual exposure from a practice is unwarranted. This level of risk is defined as the Negligible-Individual Risk Level (NIRL) and corresponds to an annual EDE of 1 mrem. The annual EDE for all scenarios evaluated for members of the public was much less than 1 mrem. Therefore, potential exposures to the public from the recycling of HE with a maximum bulk tritium activity of 2×10^{-3} μ Ci HTO/g HE should be excluded from further consideration.

CONCLUSION

Pantex Plant, LANL, and LLNL conclude from the experimental data that the volume contamination at depth for all HE main-charges examined do not show a bulk tritium activity greater than $2 \times 10^{-3} \mu\text{Ci HTO/g HE}$. Even if tritium were uniformly distributed throughout the HE main-charge at $2 \times 10^{-3} \mu\text{Ci HTO/g HE}$, it is conservatively estimated that the annual EDE for a member of the public would be much less than 1 mrem. The primary dose limit from all radiation sources and pathways for members of the public, as promulgated in DOE 5400.5 and in Title 10 CFR Part 20, is 100 mrem annual EDE. This annual dose limit includes the exposure limits from all sources of DOE airborne emissions, i.e., 10 mrem; and all radionuclides in the drinking water pathway, i.e., 4 mrem. The potential doses to members of the public from all activities associated with the recycle of HE are well below these limits.

The vast majority (96%) of HE main-charges that were measured for removable surface tritium contamination were below the removable β - γ surface release criterion set by DOE 5400.5. In fact, the majority (66%) of the HE main-charges had surface tritium contamination levels that were indistinguishable from the background measured in virgin, uncontaminated HE. Based on this data, this will allow over 95% of the Pantex Plant HE main-charges, obtained from weapon dismantlement, to be recycled and sold to commercial industry.

RECOMMENDATIONS

Pantex Plant, LANL, and LLNL recommend that the removable β - γ surface release criterion of 1,000 dpm/100 cm², promulgated in DOE 5400.5, be used as the screening criterion for HE main-charges. This unrestricted release criterion, in conjunction with a bulk activity limit of $2 \times 10^{-3} \mu\text{Ci HTO/g HE}$, should be used as an indicator of bulk tritium-contamination in HE resulting from nuclear weapons dismantlement. In addition, to verify the bulk tritium contamination in HE, it is recommended that a tritium assay be performed on the machined HE powder on a lot-by-lot basis to verify bulk tritium activity. Finally, it is recommended that all stored batches of HE be statistically re-sampled and analyzed before off-site release for commercial use to confirm that the bulk tritium activity is less than $2 \times 10^{-3} \mu\text{Ci HTO/g HE}$. The use of both criterion provides a substantial safety margin for the release of this material and is the most environmentally sound and cost-effective approach for the disposition of bulk tritium-contaminated HE.

INTRODUCTION

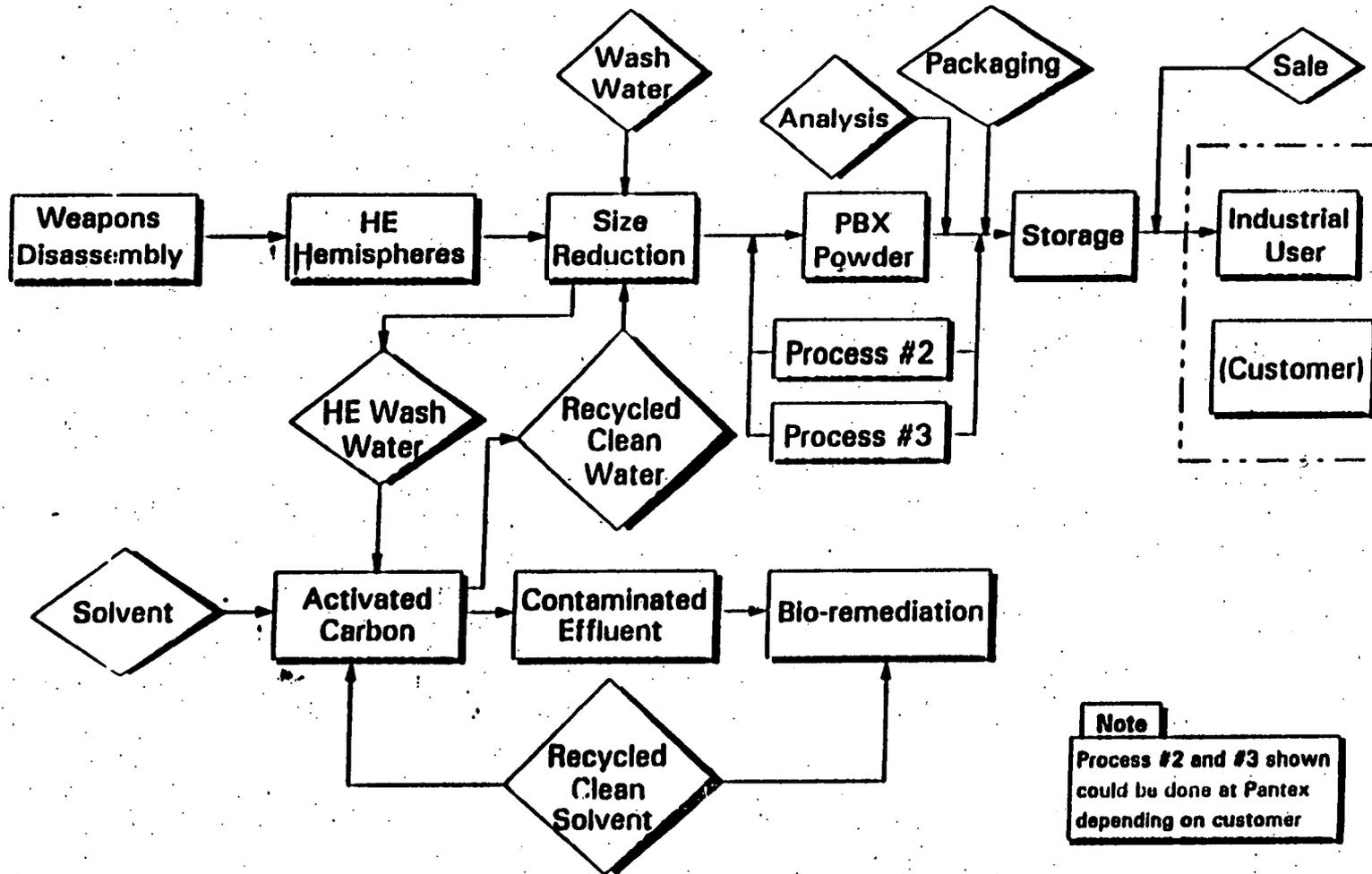
At the present time, the primary mission of Pantex Plant is to dismantle nuclear weapons that are no longer needed for the defense of the United States. Dismantlement activities are in support of the Intermediate-Range Nuclear Forces (INF) Treaty and the Strategic Arms Reduction Treaty (START) between the United States and the former Union of Soviet Socialist Republic (USSR) to reduce their respective stockpiles of nuclear, conventional, and chemical weapons.

Dismantlement activities produce a variety of materials, including high explosives (HE), that could be salvaged and released to the commercial market for recycling. Some of these materials may be contaminated with residual radioactivity, either on the surface of the materials or in bulk. At the present time, the only residual radioactivity criteria for the release of materials to the public are promulgated in Department of Energy (DOE) 5400.5 "Radiation Protection of the Public and the Environment." The release criteria apply only to radioactive surface contamination. The order further states that "... material that has been contaminated in depth ... may be released if criteria and survey techniques are approved by EH-1."

A number of different materials are released by Pantex Plant for recycling after the surface release criteria of DOE 5400.5 are met. Currently, there are no mechanisms for the salvage or recycle of HE from DOE nuclear stockpiles. Pantex Plant, in cooperation with Los Alamos and Lawrence Livermore National Laboratories (LANL/LLNL), proposes to expand the materials recycling program to include HE. This recycling effort would significantly reduce the cost of treatment and would provide a mechanism for capital recovery of a national asset. Through the year 2000, DOE will generate several hundred thousand pounds of excess high quality HE from the dismantlement of its nuclear weapons.

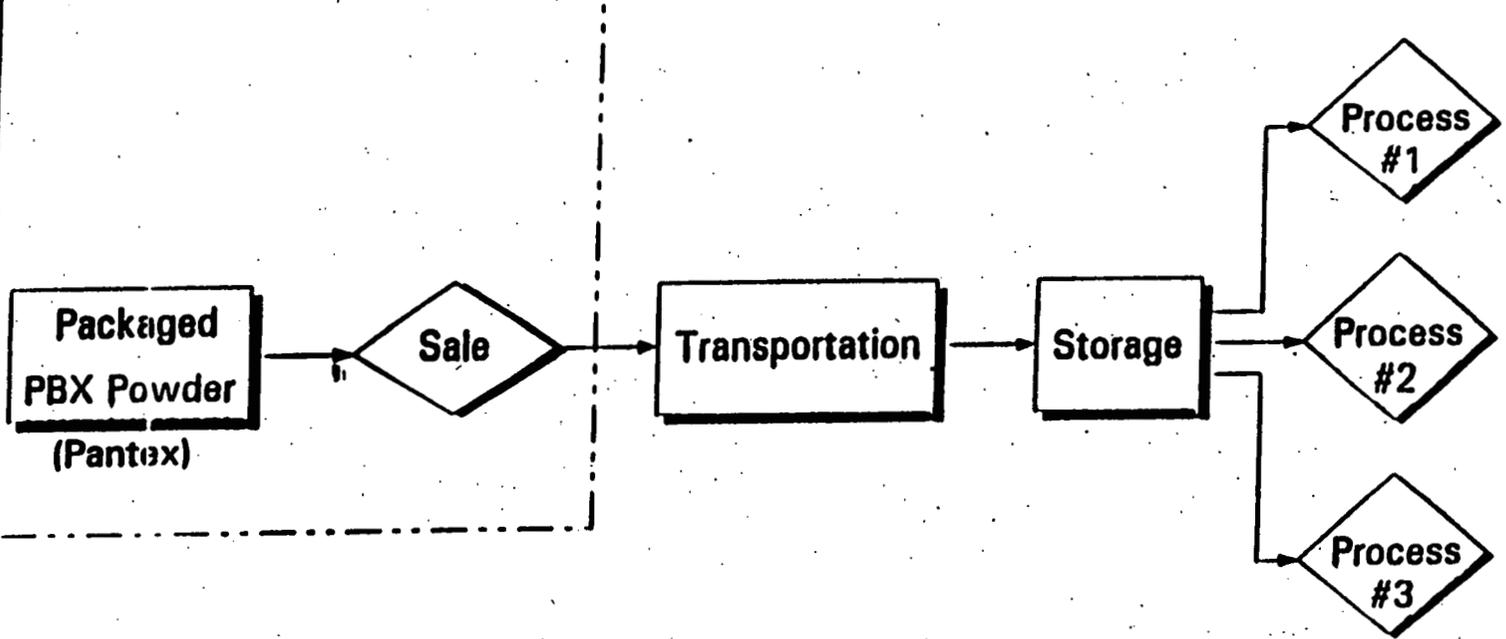
It is anticipated that the recycled HE will be sold to industrial users in the mining industry. Figure 1 shows the overall HE recycling activities that are anticipated to occur at Pantex Plant. Figure 2 provides a breakdown of the activities that are anticipated to occur at the industrial user's facilities. The industrial users, depending upon their requirements, may or may not reprocess the HE into products required for their application(s). Process #1 through #3 show the anticipated processes that may occur at the industrial user's facilities. Process #1 involves the use of the PBX as received from Pantex Plant. In Process #2, the PBX binder would be removed so that the industrial user could obtain the HMX. Process #3 involves the dissolution and recrystallization of the PBX received from Pantex Plant with subsequent reformulation into new PBX. As indicated in Figure 1, it should be noted that some reprocessing of the HE may occur at Pantex Plant. If this is required, Pantex Plant will use existing facilities and processes to perform this task. Therefore, no new facilities and/or processes are anticipated at Pantex Plant.

Figure 1: Pantex Plant Operations

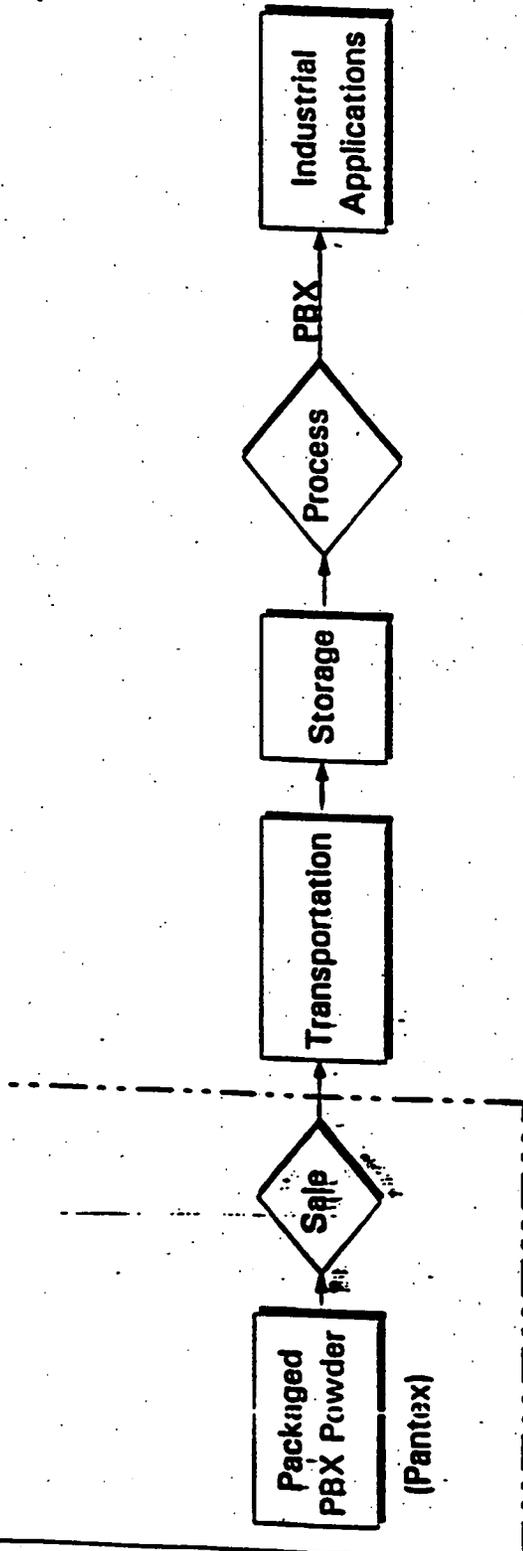


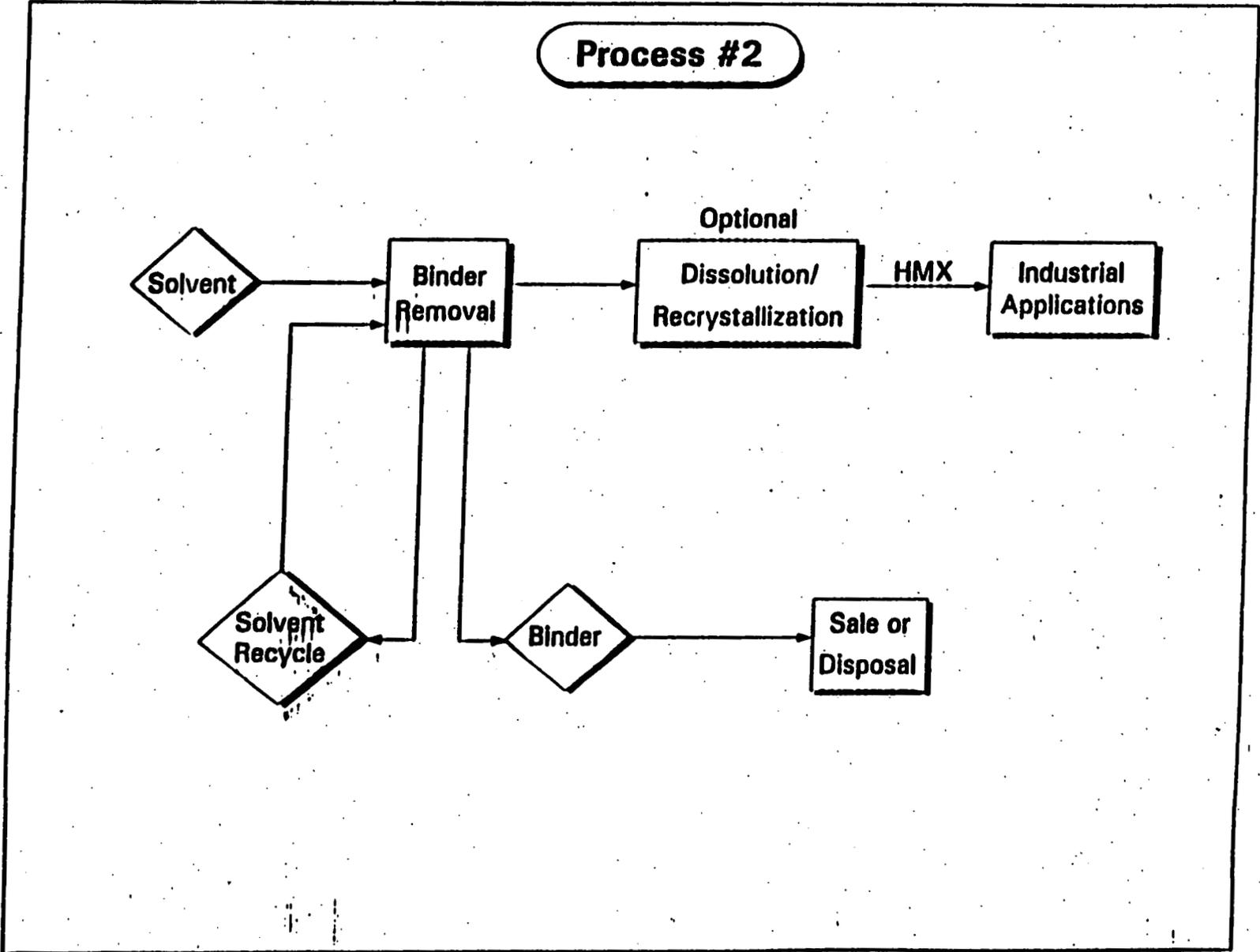
Note
 Process #2 and #3 shown
 could be done at Pantex
 depending on customer

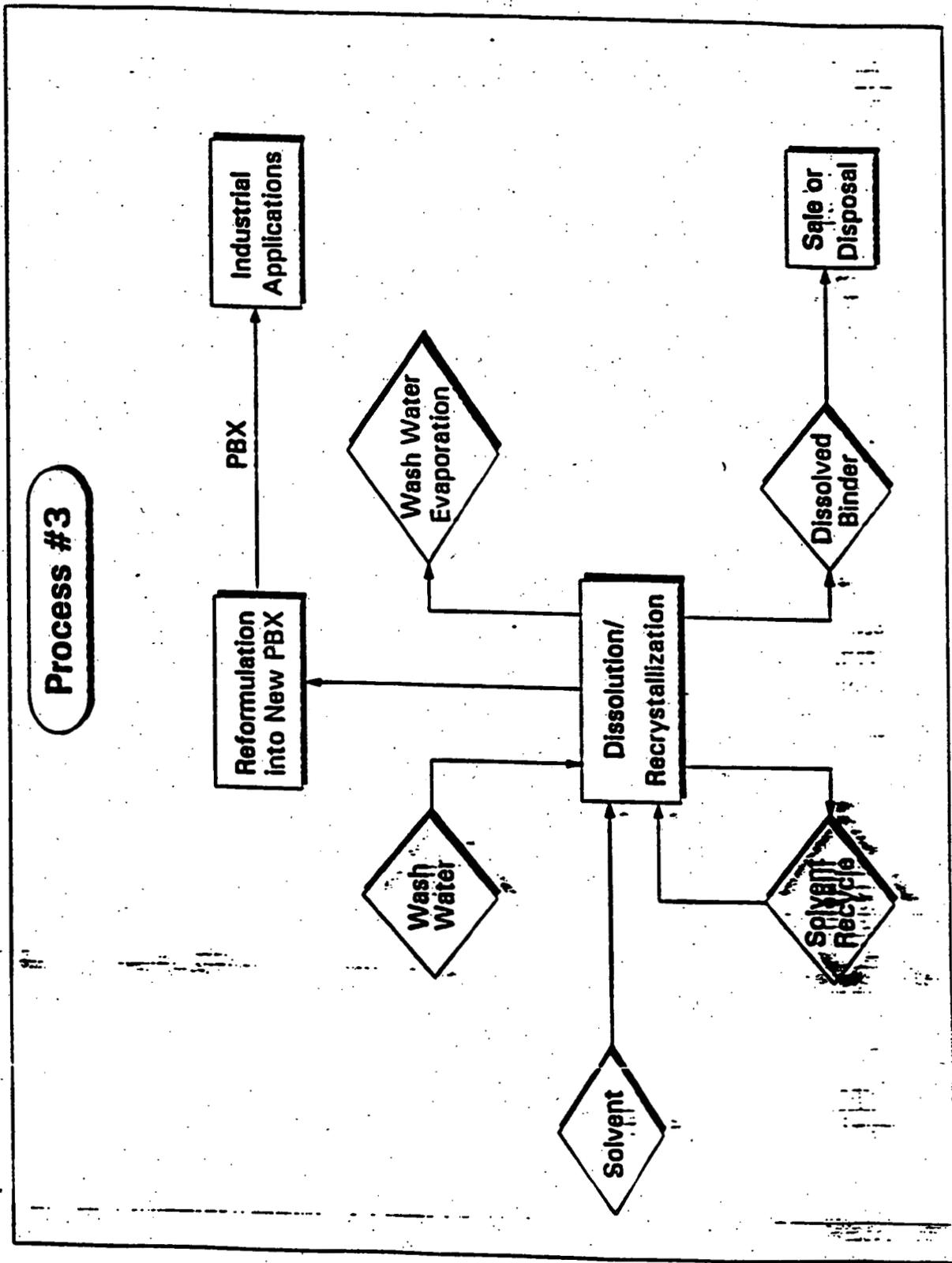
Figure 2: Industrial User Operations



Process #1







At Pantex Plant, some of the HE is slightly contaminated in bulk with tritium as a result of the diffusion of tritium from the reservoir contained in many nuclear devices. The amount and rate of diffusion depends upon the design, age, and storage conditions of the nuclear device. HE is not unique in this respect. Tritium is known to penetrate metals and plastics in a similar manner.

Historically, the surplus HE from nuclear dismantlement activities is demilitarized and sanitized, if necessary, and then treated and managed as a waste, not a national asset. Surplus DOE explosives that have been classified as waste are treated by open burning/open detonation (OB/OD). This activity releases small quantities of carbon monoxide (CO), carbon dioxide (CO₂), oxides of nitrogen (NO_x), fluorides (F⁻), chlorides (Cl⁻), and airborne tritium in the form of HTO.

All releases of CO, CO₂, NO_x, F⁻, and Cl⁻ are in full compliance with the Resource Conservation and Recovery Act (RCRA) interim strategy permit and written grant of authority issued by the Texas Natural Resource Conservation Commission (TNRCC) of the state of Texas. In addition, the release of the airborne tritium activity at Pantex Plant is in full compliance with Title 40 Code of Federal Regulations (CFR), Part 61, Subpart H "Environmental Protection Agency Regulations on National Emissions Standards for Hazardous Air Pollutants - National Emissions Standards for Emissions of Radionuclides Other Than Radon From Department of Energy Facilities." This regulation limits the annual effective dose equivalent (EDE) (H_p) [End Note 1] for all airborne radioactive emissions from a DOE facility to 10 millirem per year (mrem/yr) to the nearest member of the general public. All airborne radioactive emissions, either individually or in combination, arising from normal operations at Pantex Plant do not exceed levels which could cause an annual EDE to the nearest member of the general public in excess of 0.1 mrem.

Pantex Plant, LANL, and LLNL propose to use the existing removable surface contamination criterion of DOE 5400.5, namely, 1,000 disintegrations per minute over a 100 square centimeter area (dpm/100 cm²) for beta-gamma (β-γ) emitters, as the screening basis for recycling HE main-charges. This is a factor of 10 below the recommendation for removable tritium surface contamination found in DOE Handbook "Primer on Tritium Radiological Fundamentals" (FSC-6910, July 1993).

All HE main-charges that show removable tritium surface contamination less than or equal to the DOE 5400.5 β-γ release criterion would be processed by fluid-jet and/or wet-machining to produce a wet powder slurry. Each batch of wet powder slurry would be sampled and analyzed to confirm that the bulk tritium activity does not exceed 2×10^{-3} microcuries of tritium per gram of HE (μCi HTO/g HE). If the bulk tritium activity is less than or equal to 2×10^{-3} μCi HTO/g HE, the wet HE would be packaged in plastic bags and approved shipping containers. The HE would then be released to the commercial market. All HE main-charges that exceed the removable β-γ surface release criterion in DOE 5400.5 would be disposed of by alternate processes. It should be noted that the machining, sampling and

analysis, and packaging operations for the proposed recycling of the HE are existing operations at Pantex Plant.

If Pantex Plant is not allowed to use the existing removable surface contamination criterion promulgated in DOE 5400.5, namely, 1,000 dpm/100 cm² for β - γ emitters, costly alternative disposal (or treatment) processes, in lieu of OB/OD, must be developed and brought on line quickly to prevent undesirable HE accumulation. This proposal is consistent with the desire of Pantex Plant, LANL, and LLNL to improve disassembly processes, minimize waste, recycle our national resources, and protect the public and the environment.

PROPERTIES OF TRITIUM

The element hydrogen (¹H₁, commonly referred to as protium) has an atomic number of 1 and an atomic weight of 1 [End Note 2]. It has two heavier isotopes associated with it, namely, deuterium (D or ²H₁) and tritium (T or ³H₁). All forms of hydrogen have 1 proton in their nuclei. The difference in the nuclear structure between a protium, deuterium, or tritium atom is due to the number of neutrons present in their respective nuclei. Protium has 0 neutrons, D has 1 neutron, and T has 2 neutrons. Protium and deuterium are stable while tritium is radioactive. Tritium is a radionuclide that occurs naturally in the environment and is also produced by man-made processes (see Appendix A).

Tritium decays to non-radioactive helium-3 (³He₂) by emitting a very weak, low-energy beta particle (β^-) that has a maximum energy of 18.6 kiloelectron volts (keV) and an average energy of 5.7 keV. It does not emit any other form of radiation. Tritium has a radioactive half-life of 12.26 years; therefore, half of the radioactivity of any initial amount of tritium will decay naturally to ³He₂ in 12.26 years.

Because tritium is an isotope of hydrogen, its chemical properties and distribution in nature are the same as hydrogen. Since tritium reacts with substances in the same manner as hydrogen, it incorporates into the water present in the atmosphere to form tritiated water (HTO). The HTO falls to earth with rain and follows the same environmental pathway as regular water (H₂O). Nearly all tritium present in our environment is in the form of HTO.

BULK TRITIUM ACTIVITY IN HIGH EXPLOSIVE MAIN-CHARGES

Dismantlement of nuclear weapons at Pantex Plant removes the HE main-charges from the physics packages. Some of the weapon designs result in the HE main-charge becoming contaminated with low levels of tritium in the form of tritium oxide (HTO). Although much of this tritium contamination is at or near the surface of the main-charge, some of the tritium activity has penetrated into the depth of the HE. Accordingly, a number of measurements have been taken to determine the amount of removable tritium surface contamination on the

HE main-charge and the level of tritium activity as a function of depth in the volume of the HE.

The surfaces of the HE main-charges from various weapons designs have been monitored routinely for the presence of removable tritium surface contamination. The W68 program was chosen for additional study because of its history for HE contamination from the tritium reservoir. At least two areas with the highest suspected tritium contamination were measured for removable tritium surface contamination using a standard wet swipe technique. A total of 123 W68 HE main-charges were surveyed for removable tritium surface contamination. Table 1 shows the results of the wet surface swipes. An analysis of the data showed that the data were log normally distributed. About 96% of the HE main-charges had removable tritium surface contamination less than or equal to 1,000 dpm/100 cm².

Range (dpm/100 cm ²)	Total	Percent
0 - 42	82	66.7
43 - 100	29	23.6
101 - 500	5	4.1
501 - 999	2	1.6
> 1000	5	4.1

The analytical method, i.e., liquid scintillation counting, used to detect tritium activity on wet swipes is subject to some interference due to the chemicals in the HE and to electronic "noise" in the detection system. In order to account for these interferences, a series of wet swipes were taken on virgin HE of the same composition as that used in the W68 program. Though this was virgin, uncontaminated HE, wet swipe results fell in the range of 0 to 42 dpm/100 cm² above the levels measured in blank samples that contained only liquid scintillation cocktail. These values are very close to the lower limit of detection of the method and probably reflect chemiluminescence and electronic noise in the measurement. Interestingly, 66.7% of the values from the 123 W68 dismantlement main-charges fell in the same range. This analysis suggests that the majority of HE from disassembly will be indistinguishable from the "background" levels measured in virgin, uncontaminated HE.

Tritium, by its nature, will migrate into HE main-charges as it does with metals, plastics, and other hydrogenous materials, thus causing the HE main-charge to become contaminated in bulk with tritium. To measure the depth of penetration, duplicate depth profile measurements on two HE main-charges with appreciable removable tritium surface contamination were performed. A series of concentric rings of HE, about 0.1-inch (in.) in thickness by 0.3-in. high, were dry machined from the equator of each HE main-charge.

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The material from each concentric ring was collected, dissolved in dimethylsulfoxide (DMSO), and analyzed for tritium using liquid scintillation counting.

In addition, virgin W68 HE was sampled, dissolved in DMSO, and analyzed by liquid scintillation to establish a "background" for the virgin HE. Depending upon the "background" counts for the virgin HE, the sample size, counting time, and detector counting efficiency, the "background" level found in the virgin HE ranged from 1.9×10^{-5} to 2.9×10^{-5} $\mu\text{Ci HTO/g HE}$. The virgin HE "background" was subtracted from the gross HE analyses to determine the actual tritium concentration in the contaminated HE main-charges. The highest bulk tritium activity was measured near the outer surface of the HE main-charge, with the bulk tritium activity decreasing fairly uniformly with depth. The bulk tritium activity ranged from 9.7×10^{-5} to 1.0×10^{-3} $\mu\text{Ci HTO/g HE}$ above the "background" for the virgin HE.

Additional depth information was obtained by analyzing machining scrap. Six HE main-charges from the W68 program were wet-machined, using a steel cutter cooled with water, to produce a wet, spongy, flocculent powder. The wet powder was sampled, dissolved in DMSO, and analyzed using liquid scintillation counting. In a similar manner, virgin W68 HE was sampled, dissolved in DMSO, and analyzed by liquid scintillation to establish a "background" for the virgin HE. This "background" was subtracted from the gross HE analyses for the contaminated HE main-charges. As shown in Table 2, the six wet powders showed bulk tritium activity that ranged from 3.7×10^{-5} to 2.2×10^{-4} $\mu\text{Ci HTO/g HE}$ above the "background" for the virgin HE. The geometric mean (m_g) of the wet samples shown in Table 2, where $m_g = \text{antilog } \Sigma \log x_i / N$, was 9.6×10^{-5} $\mu\text{Ci HTO/g HE}$.

Sample drying also effects the tritium content of the contaminated HE wet powders. Portions of the six wet powders were dried at 200°F under vacuum and analyzed using liquid scintillation counting. As above, the "background" for the virgin W68 HE was subtracted from the gross analytical results for the dry contaminated HE samples. As shown in Table 2, the bulk tritium activity for the dry samples of HE ranged from 3.5×10^{-5} to 1.0×10^{-4} $\mu\text{Ci HTO/g HE}$. This bulk tritium activity for the HE samples that were dried is about 5 to 60% lower than the wet samples above the "background" of the virgin W68 HE. The geometric mean (m_g) for the samples that were dried was 6.0×10^{-5} $\mu\text{Ci HTO/g HE}$.

Part Number ⁽¹⁾	Wet Samples ($\mu\text{Ci HTO/g HE}$)	Dry Samples ($\mu\text{Ci HTO/g HE}$)
68-XXX-XX-28 ⁽²⁾	2.2×10^{-4}	1.0×10^{-4}
68-XXX-XX-04	8.2×10^{-5}	4.5×10^{-5}
68-XXX-XX-44	1.5×10^{-4}	1.0×10^{-4}
68-XXX-XX-20	7.2×10^{-5}	5.1×10^{-5}
68-XXX-XX-41	3.7×10^{-5}	3.5×10^{-5}
68-XXX-XX-49	1.1×10^{-4}	5.5×10^{-5}

⁽¹⁾The actual serial numbers for these parts are classified.

⁽²⁾The serial number for this part is not the same as the serial number for the similar part in Table 3.

Fluid-jet machining is another method that may be used to reduce HE main-charges to powder. In this process, high-pressure water is used as the cutting medium instead of the steel cutter used in the wet-machining process. This process is faster, uses substantially less water, and produces a wet slurry of HE with particle sizes ranging up to 7 millimeters (mm) in diameter.

Three HE main-charges from the W68 weapons program were fluid-jet machined. Multiple samples of the slurry from each HE main-charge were dissolved in DMSO and analyzed by liquid scintillation counting. DMSO solutions of virgin W68 HE were also analyzed by liquid scintillation counting to establish a "background" for the virgin HE. This "background" was subtracted from the gross HE results for the contaminated HE main-charges. As shown in Table 3, the contaminated HE samples showed bulk tritium activity that ranged from 9.5×10^{-5} to 1.8×10^{-3} $\mu\text{Ci HTO/g HE}$ above the virgin HE "background." The geometric mean (m_g) of the samples shown in Table 3 was 2.9×10^{-4} $\mu\text{Ci HTO/g HE}$.

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Table 3: Tritium Assay Results for Fluid-Jet Machining		
Part Number ⁽¹⁾	Sample Number	Wet Samples ($\mu\text{Ci HTO/g HE}$)
68-XXX-XX-35	1	1.3×10^{-3}
	2	1.6×10^{-3}
	3	1.8×10^{-3}
68-XXX-XX-28 ⁽²⁾	1	9.5×10^{-5}
	2	1.1×10^{-4}
	3	1.4×10^{-4}
68-XXX-XX-23	1	1.6×10^{-4}
	2	1.3×10^{-4}
	3	1.3×10^{-4}

⁽¹⁾The actual serial numbers for these parts are classified.

⁽²⁾The serial number for this part is not the same as the serial number for a similar part in Table 2.

The analytical results for bulk tritium activity from the fluid-jet and wet-machining processes confirm the results of the depth profile analyses. The highest wet bulk tritium activity found to date is $1.8 \times 10^{-3} \mu\text{Ci HTO/g HE}$ with the geometric mean (m_g) for the different machining processes ranging from 9.6×10^{-5} to $2.9 \times 10^{-4} \mu\text{Ci HTO/g HE}$.

These measurements clearly show that, even for HE main-charges with the highest wet surface swipe levels for removable tritium, the amount of tritium in the bulk material is very low. In all cases, the bulk tritium activity has been less than $2 \times 10^{-3} \mu\text{Ci HTO/g HE}$ with the average geometric mean (m_g) being $1.9 \times 10^{-4} \mu\text{Ci HTO/g HE}$. Based on these measurements, most of the bulk tritium activity is expected to be on the order of about 1.0×10^{-5} to $1.0 \times 10^{-4} \mu\text{Ci HTO/g HE}$.

As part of Pantex Plant dismantlement quality control (QC) system, surface contamination measurements will continue to be made. Any HE that shows removable tritium surface contamination greater than 1,000 dpm/100 cm² will not be sold to commercial industry. Further, bulk tritium analyses will be performed on a "batch-by-batch" basis to ensure that the HE destined for sale contains bulk tritium activity less than or equal to

$2 \times 10^{-3} \mu\text{Ci HTO/g HE}$. Appendix B shows the procedures that are and will be used to determine the removable tritium surface contamination and the bulk tritium activity of the HE main-charges.

PATHWAY ANALYSES FOR UNRESTRICTED RELEASE

To determine the potential exposure to members of the public from the unrestricted release of the tritium-contaminated HE, a number of scenarios were examined. In each scenario examined, pathways by which tritium could be inhaled, ingested, or absorbed through the skin were examined. Each scenario included assumptions that resulted in calculated results based on a "worst" case and a "realistic" case. The exposure scenarios included:

1. Storing HE in a poorly ventilated warehouse,
2. Unpacking sealed containers of HE in a poorly ventilated area,
3. Mixing or crushing operations for dry HE,
4. Adding wet PBX to a HE water slurry,
5. Inadvertent ingestion of HE water slurry,
6. Splashing of HE water slurry, and
7. Dust plume exposure from exploding HE.

Very conservative assumptions were used to model the possible transfer of tritium activity from the HE to a hypothetical member of the public. The assumptions in the "worst-case" provide the upper bounds on the doses that may result from the unrestricted use of the HE. However, the "realistic" assumptions reflect the anticipated exposure conditions during the "life-cycle" of the HE during "normal" operational conditions. The results from "realistic" cases will be used to determine the most probable exposures to members of the public. Appendix C provides the assumptions and conditions for the pathway analyses.

The results of the pathway analyses may be found in Appendix C. The 50-year committed effective dose equivalents (CEDEs) [End Note 3] ranged from 4.6×10^{-5} to 4.9×10^{-5} millirem (mrem) for the maximally exposed individual. However, since the 10 day effective half-life [End Note 4] of tritium results in the 50-year CEDE being delivered within about 100 days, there is no significant difference between the first-year EDE (H_p) and the 50-year CEDE. Therefore, the 50-year CEDE shall be referred to hereafter as the annual EDE.

The maximum annual EDE for a worker at Pantex Plant, i.e., 3.8×10^{-5} mrem, resulted from exposure to HTO during the preparation of the HE for recycle by Pantex Plant. It should be noted that the occupationally exposed workers at Pantex Plant are considered to be members of the public. Therefore, no attempt was made to exclude their occupational exposures from the recycle effort. For a manufacturer's worker, the highest credible annual EDE received was about 4.9×10^{-5} mrem. This anticipated exposure resulted from the skin

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being splashed with HE slurry (scenario #6). These annual EDEs are much less than 1% of the applicable primary dose limit for members of the public, i.e., 100 mrem, promulgated in DOE 5400.5 and in Title 10 Code of Federal Regulation (CFR) Part 20 "Standards for Protection Against Radiation." They are well below the 30 mrem in a year DOE dose restraint for release of property (RESRAD).

The establishment of risk factors for doses hundreds of times higher than those discussed in this document is complex and incompletely understood. Thus, the estimation of the risk of radiation-induced cancer from such low doses is subject to great uncertainty.

The National Research Council's committees on the Biological Effects of Ionizing Radiations (BEIR) in its BEIR V report "Health Effects of Exposures to Low Levels of Ionizing Radiation" (BEIR 1990), stated that "... the risk depends on the particular kind of cancer; on the age and sex of the person exposed; on the magnitude of the dose to a particular organ; on the quality of the radiation, i.e., quality factor; on the nature of the exposure, whether brief or chronic; on the presence of factors such as exposure to other carcinogens and promoters that may interact with the radiation; and on individual characteristics. The estimates of the risk of cancer must rely on observations of cancers of different kinds that arise in the irradiated groups, combined with biological knowledge and theories of possible cancer induction mechanisms."

BEIR V also states that "Moreover, epidemiologic data cannot rigorously exclude the existence of a threshold [for radiation health effects] in the milliSievert (mSv) [100 mrem] dose range. Thus, the possibility that there may be no risks from exposures comparable to external natural background radiation cannot be ruled out. At such low doses and dose rates, it must be acknowledged that the lower limit of the range of uncertainty in the risk estimates extends to zero."

If one makes the assumption that there is some risk of radiation-induced cancer at dose levels that are a small fraction of natural background radiation dose levels, the best estimate of the lifetime risk of a fatal cancer from exposure to tritium is probably that of Straume (Straume 1993). Straume's "most probable risk" of cancer mortality for low level exposure to tritium in the form of tritiated water (HTO) is 8.1×10^{-5} per milliGray (mGy) [8.1×10^{-7} per mrem assuming a quality factor (Q) [End Note 5] of 1]. Since a Q of 1 was used in the calculation of the dose equivalents in these scenarios, the calculated dose equivalents (mrem) are equal to the doses (mrad). Thus, the above risk coefficient can be directly applied to the dose equivalent values.

It is estimated that about 75 members of the public, including Pantex Plant workers, would be involved in the recycling effort for the HE. The collective EDE to these members of the public is 1.5×10^3 person-mrem. Using Straume's risk coefficient of 8.1×10^{-7} per mrem and the collective EDE of 1.5×10^3 person-mrem, the lifetime cancer mortality risk to a member of the public is 1.2×10^{-9} , i.e., 8 in 100 million, for the recycling option. By

comparison, the normal risk of cancer mortality from all sources is about 0.25 to 0.33, i.e., 1 in 4 to 1 in 3.

Straume's risk estimate is based on the BEIR V risk estimates for an adult population that is corrected with a realistic Dose Rate Effectiveness Factor (DREF) [End Note 6] of 4 (BEIR, 1990) and a frequency distribution of Relative Biological Effect (RBE) [End Note 7] values specific to tritium exposures, giving an approximate RBE value of about 2.5. As stated by Straume, previous risk estimates (BEIR, 1990; ICRP 60, 1991; and UNSCEAR, 1988) used DREFs in the 1 to 2 range which are unrealistically small compared with the observed values of 2 to 10 in animal studies. Also, the previous risk estimates used a RBE for HTO of 1.0, which is smaller by a factor of 2 to 3 when the comparison is with gamma rays. In the calculation of the risk factor, the RBE of 2.5 for HTO is roughly cancelled by the similar factor increase in the DREF.

WORK PLACE CONTAMINATION SURVEYS/BIOASSAY SAMPLING

Experience at Pantex Plant shows that the handling of tritium-contaminated HE main-charges does not present a significant potential for spread of contamination or uptakes of tritium. This conclusion is supported by the results of routine contamination surveys taken in HE work areas and by the results of bioassays taken from personnel who work with the HE.

The Radiation Safety Department (RSD) provides radiation protection services at Pantex Plant. These services include radiological surveys in areas with the potential for radioactive contamination and/or penetrating radiation and bioassay services for the determination of potential internal contamination. Radiation protection activities are conducted in compliance with DOE/EH-256T "Radiological Control Manual" and Title 10 CFR Part 835 "Occupational Radiation Protection."

The RSD has established removable surface contamination limits for unrestricted release of materials from a controlled area at Pantex Plant. Plant Standard (STD) 1451 "Unrestricted Release of Material and Equipment to the Public," promulgates these limits. The unrestricted removable surface contamination limit for tritium is 1,000 dpm/100 cm², which corresponds to the DOE unrestricted release criterion for beta-gamma emitters.

Routine surveys for removable surface contamination are performed in the work place and other areas of Pantex Plant. Removable surface contamination surveys are performed by taking one wet swipe and one dry swipe at each location. The wet swipes are analyzed by liquid scintillation counting for tritium, while the other is analyzed for gross α and/or gross β . The results of the tritium analyses are compared to an authorized limit for removable surface contamination consistent with DOE 5400.5 requirements. For calendar years 1991 and 1992, 12,343 and 34,142 swipes for removable tritium surface contamination were collected and analyzed, respectively. Only one swipe in each year exceeded the criterion for removable tritium of 1,000 dpm/100 cm².

The RSD collects and analyzes routine bioassay samples for Pantex Plant employees who work on nuclear devices with the potential for tritium contamination. These workers are monitored monthly for tritium through the bioassay program. About 20 persons from the Applied Technology Division (ATD), formerly called the Explosives Technology Department (ETD), participated in the research activities to determine the level of bulk tritium activity in HE main-charges. These close-contact activities lasted about one year. No measurable tritium activity was detected in any of the collected bioassay samples for the ATD personnel. This trend has continued through March 1995. The above data provide measured evidence that the activities involved in HE main-charges do not provide a viable source of tritium contamination or biological uptake for Pantex Plant workers.

Although the above information may not be strictly applied to tritium-contaminated HE main-charges, it does provide an indirect indication that the HE main-charges are not contaminated with excessive levels of tritium. Other than the normal posting requirements for a contamination area (if the removable tritium contamination is greater than 10,000 dpm/100 cm²) and required bioassay sampling for tritium, special handling, labeling, and/or posting requirements are not necessary for work activities on HE main-charges. In addition, the Department of Transportation (DOT) has not imposed any special regulations for tritium at this low level.

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- Plant Standard, Pantex Plant, "Unrestricted Release of Material and Equipment to the Public," August 18, 1993
- Straume, T., "Tritium Risk Assessment," *Health Physics* Volume 65, No. 6, pp. 673-682, December 1993
- Title 10 CFR Part 20, *Standards for Protection Against Radiation*, January 1, 1995
- Title 10 CFR Part 835, *Occupational Radiation Protection*, December 14, 1993
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- United Nations Scientific Committee on the Effects of Atomic Radiation, *Sources, Effects, and Risks of Ionizing Radiation*, UNSCEAR, 1988
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- U.S. Department of Energy, *Manual for Implementing Residual Radioactive Material Guidelines Using RESRAD, Version 5.0*, ANL/EAD/LD-2, September 1993 (Working Draft for Comment)
- U.S. Department of Energy, *Radiological Control Manual*, DOE/EH-0256T, April 1994
- U.S. Department of Energy, *Radiation Protection for Occupational Workers*, DOE 5480.11, December 21, 1988
- U.S. Department of Energy, *Radiation Protection of the Public and the Environment*, DOE 5400.5, February 8, 1990

END NOTES

- 1: The effective dose equivalent (H_E) is the quantity that is used to assess the risk to an individual from both uniform whole body and non-uniform partial body exposures. It is the sum of the weighted dose equivalents for irradiated tissues or organs, i.e., $H_E = \sum W_T H_T$, where W_T is the weighting factor for tissue T and H_T is the average dose equivalent received by tissue T.
- 2: The chemical symbol (X), atomic weight (A), and atomic number (Z) of all elements/isotopes are represented as follows: ${}^A_Z X$.
- 3: The CEDE is the sum of the committed dose equivalents (CDEs) to various tissues in the body ($H_{T,50}$), each multiplied by the appropriate weighting factor (W_T) — that is, $H_{E,50} = \sum W_T H_{T,50}$. The CDE is the dose equivalent calculated to be received by a tissue or organ over a 50-year period after intake of the radionuclide into the body. The CDE does not include any contributions from radiation sources external to the body.
- 4: The effective half-life (T_E) is the time required for a radioactive element deposited in the body to be diminished by 50 percent as the result of the combined action of radioactive decay (T_R) and biological elimination (T_B). It is calculated as:
$$T_E = (T_R \times T_B) / (T_R + T_B)$$
- 5: The quality factor is used in radiation protection in an attempt to account for the effectiveness of different radiations to produce biological effects. Q is the factor by which the radiation absorbed dose (rad) is multiplied to yield the dose equivalent (rem) of the exposure.
- 6: The DREF is defined as the slope of the linear curve fitted to the high dose/high dose-rate data divided by the slope of the curve fitted to the low dose/low dose-rate data for the same endpoint and the same radiation.
- 7: The RBE is the experimentally determined ratio of an absorbed dose of a radiation in question to the absorbed dose of a reference radiation required to produce an identical biological effect in a particular experimental organism or tissue. *The RBE should not be used in radiation protection. (See Quality Factor.)*

APPENDICES

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APPENDIX A

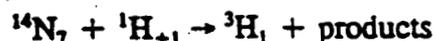
TRITIUM SOURCES/BIOLOGICAL EFFECTS

NATURAL AND MAN-MADE SOURCES OF TRITIUM IN THE ENVIRONMENT

Tritium has been produced naturally in the environment since the earth's formation. All forms of life have thus evolved in an atmosphere that has included tritium. Numerous sources contribute to the amount of tritium currently in the environment:

1. Natural production in the atmosphere (from the interaction of cosmic rays with oxygen and nitrogen)
2. Nuclear weapons testing
3. Nuclear power stations/nuclear weapons production facilities
4. Consumer products (i.e., luminous dials in watches, exit signs, airport runway signs, electronic devices filled with tritium gas, etc.)

Naturally-occurring tritium is produced in the upper atmosphere from nuclear reactions between atoms of oxygen-16 ($^{16}\text{O}_8$) and nitrogen-14 ($^{14}\text{N}_7$) and cosmic neutrons ($^1\text{n}_0$) and protons ($^1\text{H}_{+1}$) to produce tritium ($^3\text{H}_1$) [End Note 1]. The primary reactions are:



According to the National Council of Radiation Protection and Measurements (NCRP) in NCRP Report No. 62 "Tritium in the Environment," the world inventory of natural tritium due to cosmic ray interactions is estimated to be 70 million (7×10^7) curies (Ci), corresponding to a world production rate of 4 million (4×10^6) curies per year (Ci/yr). The rate of oxidation of tritium to HTO is less than 1% of the tritium released to the atmosphere (NCRP Report No. 62).

Commencing with the atomic bomb tests in 1945 and continuing through 1963, various radionuclides, including tritium, were injected into the atmosphere. The total amount of tritium injected into the earth's atmosphere by 1963 was estimated to be 185-240 times the natural level of tritium in the atmosphere. Atmospheric testing of nuclear devices was terminated in 1963. Most of the tritium activity has since been removed from the earth's atmosphere by radioactive decay and by natural processes. Assuming that HTO is removed

from the atmosphere by precipitation scavenging with an effective removal half-life of 3 years (NCRP Report No. 94 "Exposure of the Population in the United States and Canada from Natural Background Radiation," the current HTO concentration in the atmosphere is about 18 pCi/L. Most of the HTO in our environment is due to the natural production of tritium in the atmosphere. The tritium inventory due to weapons testing will decay to the natural level of 70 million (7×10^7) Ci approximately by the year 2030 (NCRP Report No. 62).

TRITIUM IN WATER

Tritium easily incorporates into water molecules in the atmosphere to form HTO. The HTO falls to earth with rain and follows the same environmental pathway, or hydrological cycle, as water. Thus, it is often found in surface and underground waters, which are the sources of mankind's drinking water supplies. About 3 to 16 pCi/L of HTO are present in environmental waters (NCRP Report No. 62). As a result, the Environmental Protection Agency's (EPA) Safe Drinking Water Act set the safe drinking water level at 20,000 pCi/L. This level was selected by EPA so that an individual who drinks 2 L (about half a gallon) of water daily with this concentration of tritium would receive an annual EDE of only 4 millirem (mrem). This is about the same amount of radiation dose one gets from cosmic sources while flying round-trip from Los Angeles to New York.

BIOLOGICAL EFFECTS OF TRITIUM

Tritium is a low-energy beta emitter. The beta particle emitted by tritium cannot penetrate deeply into tissue or travel far in air. By definition, there is no potential for external exposure to the whole body. The only potential for exposure from tritium arises from tritium absorption by the body.

The rate of uptake of tritium by the body is greatly dependent upon the chemical form of the tritium. Pure tritium gas, either as HT or TT, is not readily taken up by the body. If inhaled, HT and TT gas is only slightly incorporated into the body because it is rapidly removed through exhalation. About 99.9% of the HT and TT gas is exhaled along with the normal gaseous waste products, carbon dioxide (CO_2) and H_2O vapor. Uptake of HT or TT gas by absorption through the skin is negligible compared to inhalation. Therefore, HT and TT present a minimal radiological hazard.

Most of the tritium associated with the contaminated HE will be in the form of HTO, either as a liquid or a vapor. Tritium in the form of HTO is readily taken into the body. HTO may enter the body via inhalation, ingestion, or absorption through the skin. Exposure to HTO in air results in uptakes via inhalation and absorption through the skin. Essentially 100% of inhaled HTO vapor is assumed to be absorbed by the lungs. The rate of uptake of HTO via skin absorption is assumed to be about 50% of the rate of uptake via inhalation.

Thus, for a person exposed to airborne HTO vapor, about 67% of the uptake occurs via inhalation while 33% occurs via skin absorption. Ingested (oral intakes) HTO is assumed to be almost completely absorbed into the body via the gastrointestinal (GI) tract.

Once inside the body, HTO behaves just like water. It is rapidly and uniformly distributed throughout the entire volume of body water, where it can deliver a radiation dose to the soft tissues of the whole body. Since HTO acts just like water, it is eliminated from the body in the same manner as water via urine, feces, sweat, and exhaled water vapor. Half of the body's water is eliminated and replaced in a period of about 10 days. Thus, HTO in the body is also eliminated with a effective half-life [End Note 2] of 10 days.

For an exposure of an individual via inhalation, ingestion, or skin absorption, a dose conversion factor of 0.064 mrem per microcurie (mrem/ μ Ci) of HTO uptake can be used to calculate the 50-year committed effective dose equivalent (CEDE) [End Note 3]. This factor is found in Federal Guidance Report #11 "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion." The conversion factors in this document are based on the assumptions found in International Commission on Radiological Protection (ICRP) Publication 30 "Limits for Intakes of Radionuclides by Workers."

The biological effects of ionizing radiation on living cells are the result of the energy deposited within the cells. The probability of occurrence of an observed effect is conservatively assumed to be proportional to the amount of energy deposited in the cells. Certain types of radiation are more effective than others in producing such biological effects. An assigned term, the quality factor (Q) [End Note 4], is used in radiation protection in an attempt to account for the effectiveness of different radiations to produce biological effects. Q is the factor by which the radiation absorbed dose (rad) is multiplied to yield the dose equivalent (rem) of the exposure. Q for X-rays, gamma rays, and electrons of all energies is 1.0. In comparison, Q for an alpha particle is 20. This means that for a given radiation absorbed dose, the risk of radiation damage can be 20 times higher for an alpha particle than for an X-ray, gamma ray, or an electron.

Radionuclides that emit only low-energy radiation and that are eliminated relatively quickly from the body have a very low radiotoxicity. The International Atomic Energy Agency (IAEA) has investigated the radiotoxicity of 236 radionuclides, including tritium, and ranked them for toxicity from high (1) to low (236). Tritium was ranked 225 and is classified as having very low radiotoxicity because of the low-energy beta radiation emitted by tritium, the lack of biological concentration of tritium in the body, the uniform distribution of tritium in the body, and the short effective half-life of tritium. No chemical toxicity is displayed by tritium since none arises from hydrogen.

REFERENCES

Eckerman, K.F., et al, Federal Guidance Report #11, "*Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*", USEPA 1988

International Commission on Radiological Protection, *Limits of Intakes of Radionuclides by Workers*, ICRP Publication 30, Part 1, 1979

National Council on Radiation Protection and Measurements, *Tritium in the Environment*, NCRP Report No. 62, 1979

National Council on Radiation Protection and Measurements, *Exposure of the Population of the United States and Canada from Natural Background Radiation*, NCRP Report No. 94, 1987

Title 42 U.S.C., *Safe Drinking Water Act*, Paragraphs 300f-300j-26

APPENDIX A END NOTES

- 1: A_ZX represents all elements/isotopes, where X = chemical symbol, A = atomic weight, and Z = atomic number.
- 2: The effective half-life (T_E) is the time required for a radioactive element deposited in the body to be diminished by 50 percent as the result of the combined action of radioactive decay (T_R) and biological elimination (T_B). It is calculated as:
$$T_E = (T_R \times T_B) / (T_R + T_B).$$
- 3: The 50-year CEDE (H_{50}) represents the total cumulative dose equivalent delivered to an individual for a 50 year time period beginning with the instant of uptake of a radioactive material into the body.
- 4: The quality factor is used in radiation protection in an attempt to account for the effectiveness of different radiations to produce biological effects. Q is the factor by which the radiation absorbed dose (rad) is multiplied to yield the dose equivalent (rem) of the exposure.

APPENDIX B

TRITIUM MEASUREMENT TECHNIQUES

The purpose of this appendix is to demonstrate that the measurement of removable tritium surface contamination and the sampling and analyses for bulk tritium activity are performed in a sound, defensible manner. This requires that these activities be performed using written, approved procedures. These activities are (and will be) part of Pantex Plant's dismantlement quality control (QC) system to assure that HE with removable tritium surface contamination greater than 1,000 disintegrations per minute over an area of 100 square centimeters (dpm/100 cm²) will not be sold to commercial industry.

The Radiation Safety Department (RSD) is responsible for performing wet swipes to determine the level of removable tritium surface contamination on the outer surface of the HE main-charges. The RSD collects the wet swipes using Internal Operating Procedure (IOP)-D5266 "General Survey Techniques." The Applied Technologies Division, formerly called the Explosives Technology Division (ETD), is responsible for collecting samples of HE to determine the bulk tritium activity in the HE main-charges. Depth profile samples from potentially tritium-contaminated HE are dissolved by ATD using Development Instruction (D.I.) 94-122 "Dissolution of Explosive Samples." Samples of "virgin" HE are collected by ATD using D.I. 94-121 "Dissolution of Background Explosive Samples."

To verify the bulk tritium activity in the HE main-charges, the wet HE product will be sampled and analyzed. As main-charge HE is machined down into powder, sufficient quantities will be gathered to constitute a single lot of material. Each lot will be statistically sampled using O&I Standard 7-0706.4 "TATB Receipt and Sampling." Multiple samples, representing each lot, will be assayed for bulk tritium activity using liquid scintillation counting, with "virgin" material constituting the background. The HE analytical results will be compared to the administrative limit of $2 \times 10^3 \mu\text{Ci HTO/g HE}$ for bulk tritium activity.

All wet surface swipes and HE samples are (and will be) prepared and analyzed by RSD using analytical laboratory procedure 3 "Operation of the Packard 2250 Series Liquid Scintillation Analyzer" and/or procedure 4 "Operation of the Packard 2500 Series Liquid Scintillation Analyzer." Both of these procedures may be found in MNL-0028 "Analytical Laboratory Procedure Manual."

LIQUID SCINTILLATION COUNTING

To determine the level of removable tritium surface contamination and the bulk tritium activity on and in the HE main-charges, the liquid scintillation counting technique is used. Liquid scintillation counting is the usual technique for determining the activity of low-energy

beta-emitting radionuclides such as tritium (^3H) and carbon-14 (^{14}C) [End Note 1] in samples.

The liquid scintillation process depends upon the presence of the sample in the liquid scintillator. Liquid scintillators are normally composed of one or more fluorescent solutes in an organic solvent. To prepare the sample for liquid scintillation counting, the radioactive material is introduced into the liquid scintillator. The radioactive material and liquid scintillator are intimately mixed with one another for optimum counting.

The radiation energy emitted from the sample is expended in the ionization and excitation of the solvent. This energy is subsequently transferred to the solute (or fluor) and is re-emitted as photons in the violet and ultraviolet energy spectrum. These photons, having an average wavelength characteristic of the solute, can then be detected at the photocathode of one or more phototubes. In the general application of the method, a "wavelength shifter" may also be used to produce an output pulse of photons having wavelengths compatible with the optimum sensitivity of the photocathode. Following amplification in the photocathode, the signal may be routed to additional amplifiers. The final amplified signal is recorded with a suitable counter.

Each stage of the process of converting energy into an electronic pulse involves some distortion or uncertainty. Thus, an energy input will produce, on average, a pulse height with an uncertainty of one estimated standard deviation (1σ), which can be equal to 50 to 100 percent of the pulse height. In addition, any energy input may not necessarily produce a pulse height that is equal to the energy input because the system is non-linear. Because of these distortions and non-linearities, a background value for the set of samples must be determined. The background count for this application is derived from a sample of "virgin" HE prepared in the same manner as the samples from the HE main-charge. Reagent blanks, consisting of liquid scintillation cocktail only, are counted to determine the background, i.e., noise, of the counting equipment.

Determination of the background count in a liquid scintillation system is more difficult than in many other radiological analytical systems. A number of factors influence the background for this type of analytical system. Certain chemical reactions, e.g., chemiluminescence, may occur in the scintillation liquid that produce light at levels similar to the results from decay of the sample. Excitation of the glass vial, vial cap, and scintillation liquid by sunlight or fluorescent light may lead to delayed light emission. The decay of this phosphorescence can take several hours and can produce a high initial, non-reproducible, background count. In addition, a significant contribution from electronic noise, based on interactions between two phototubes that are facing each other, can cause problems in determining the background count.

Due to the extremely low activity of the tritium in the HE samples for this project, the liquid scintillation detector exhibits a background counting rate (r_b) that is not negligible and has to be subtracted from the gross counting rate (r_g). Typically, a gross count rate is observed by

counting for a total period of time, t_t . A background count rate is measured, in the absence of a sample, over a time t_b . The net count rate (r_{net}), due to the sample alone, is then:

$$r_{net} = r_s - r_b$$

The optimum division of available time between background and sample counting is given by:

$$t_b/t_t = (r_b/r_s)^{1/2}$$

To approximate the optimum division of time, estimates of r_s and r_b have to be obtained from a preliminary run. The RSD currently counts all tritium/HE samples for at least 10 minutes each.

An estimate of the statistical error due to the background count must be included in the final estimate of the activity of the sample. On the basis that the source and background counts are additive, the standard deviation of the net counting rate is:

$$\text{sigma } (\sigma) = (\sigma_s^2 + \sigma_b^2)^{1/2} = (r_s/t_t + r_b/t_b)^{1/2}$$

If the confidence level for the accuracy of the analyses is set at 2σ , there is a 95% probability that the "sample" count rate is within 2σ of the "true" count rate. Alternately, there is a 5% chance that the "sample" count rate is not within 2σ of the "true" count rate.

CRITICAL DECISION LEVEL, L_C , AND DETECTION LIMIT, L_D

For each sample type, a critical decision level, L_C , is established to determine whether or not the sample result is statistically different from what would be expected from a "background" sample. The calculated L_C is used to avoid false conclusions about the presence of tritium in the sample of HE main-charge. The L_C may be calculated as follows:

$$L_C = k \cdot \sqrt{2} \cdot \sqrt{B} = 2.32 \cdot \sqrt{B}$$

where: L_C = critical decision level,
 k = 1.64, the value of the standardized normal deviate that is exceeded with probability $\alpha = 0.05$, and
 B = background counts recorded in time, t .

With the value of L_C thus computed, the probability of a "false" conclusion at a level of L_C is about 5%.

The number of net counts, e.g., net counts = total counts - background counts, resulting from a count time, t , is compared to the calculated L_C . If the number of net counts observed

in the HE sample is greater than L_c , the decision is made that tritium is present in the sample. If the number of net counts is less than or equal to L_c , the decision is made that there is no tritium in the HE sample.

However, the calculated L_c values will not be used to determine whether a lot of HE would be released to the commercial market. The release will be based upon the detection limit, L_D , for the liquid scintillation counting system. The calculated L_D value, commonly referred to as the minimum detectable activity (MDA), is the signal level of the counting system such that a signal at or above this level is likely to be detected. L_D may be calculated as follows:

$$L_D = k^2 + 2L_c = 2.71 + 4.65\sqrt{B}$$

where: L_D = signal level such that a signal at or above this level is likely to be detected,
 L_c = critical decision level,
 k = 1.64, the value of the standardized normal deviate that is exceeded with probability $\alpha = 0.05$, and
 B = background counts recorded in time, t .

With the value of L_D thus computed, the probability of "non-detection" at a level of L_D is about 5%.

The administrative limit of $2 \times 10^{-3} \mu\text{Ci HTO/g HE}$ for bulk tritium activity is well above the L_D for the liquid scintillation counters currently being used by RSD. The calculated L_D value for a "virgin" sample of HE has been as high as 465 counts with a 30-minute count time. This corresponds to a L_D of $2.9 \times 10^{-5} \mu\text{Ci HTO/g HE}$ when corrected for counting efficiency and sample volume. This L_D is about 100 times less than the above administrative limit and is in the range of many of the assayed contamination levels in HE main-charges.

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REFERENCES

Analytical Laboratory Procedure Manual (MNL-0028), Pantex Plant, *Operation of the Packard 2250 Series Liquid Scintillation Analyzer*, June 2, 1994

Analytical Laboratory Procedure Manual (MNL-0028), Pantex Plant, *Operation of the Packard 2500 Series Liquid Scintillation Analyzer*, June 2, 1994

Development Instruction (D.I.) 94-121, Pantex Plant, *Dissolution of Background Explosive Samples*, April 28, 1994

Development Instruction (D.I.) 94-122, Pantex Plant, *Dissolution of Explosive Samples*, April 27, 1994

Internal Operating Procedure (IOP)-D5266, Pantex Plant, *General Survey Techniques*, June 2, 1994

Operations and Inspections Standard (O&I) 7-0706.4, Pantex Plant, *TATB Receipt and Sampling*, June 5, 1990

APPENDIX B END NOTES

- 1: A_ZX represents all elements/isotopes, where X = chemical symbol, A = atomic weight, and Z = atomic number.

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APPENDIX C

PATHWAY ANALYSES

INTRODUCTION

During the process of dismantlement of nuclear weapons at Pantex Plant, high explosive (HE) material is removed from the physics package of the weapons. In some instances, this HE is slightly contaminated with tritium. Pantex Plant is proposing to recycle this HE by selling it to commercial vendors/users. Some of the tritium in the HE may be released to the environment during the handling, storage, and use of the HE. This release of tritium may result in radiation doses to members of the public, e.g., miners and other HE workers.

This appendix details the dose pathway analyses that were performed. These analyses examine possible doses which may be received by members of the public, e.g., commercial users, from the unrestricted release of tritium-contaminated HE. The analyses begin with the premise that the level of volume tritium contamination does not exceed 2×10^3 microcuries of tritium oxide per gram of HE ($\mu\text{Ci HTO/g HE}$).

For each scenario, a set of "worst-case" and "realistic" assumptions were used as input parameters. The assumptions for both of these scenarios are very conservative. The "worst-case" assumptions establish an upper bound on the doses that may result from the unrestricted release of the HE to the public. The "realistic" assumptions reflect anticipated exposure conditions during the "life-cycle" of the recycling of the HE during "normal" work conditions. The scenarios that were examined include:

1. Exposure of workers in a poorly ventilated warehouse in which the HE was stored,
2. Exposure of workers who unpack stored HE in a poorly ventilated area,
3. Exposure to contaminated dust generated by mixing or crushing operations,
4. Exposure to tritium-contaminated water vapor during mixing of a HE water slurry,
5. Inadvertent ingestion of HE slurry due to splashing or hand-to-mouth contact,
6. Skin absorption due to HE slurry splashed on skin, and
7. Exposure to a dust plume from explosive use of HE.

DOSE CALCULATION METHODOLOGY

Seven scenarios by which tritium could be inhaled, ingested, or absorbed through the skin by members of the public were examined. In these analyses, the members of the public most likely to be exposed were assumed to be Pantex Plant workers, explosive company employees who would be working with the HE, and miners who would be using the HE.

The occupational exposures of Pantex Plant workers for storage, unpacking, and blending operations were considered to be exposures to members of the general public since these operations would be similar to the operations in scenarios #1, #2, and #4 that would be performed by members of the public, i.e., industrial users. Therefore, no attempts were made to exclude the occupational exposures of Pantex Plant workers in the evaluation of the HE recycle option.

Conservative assumptions were used to model the possible transfer of tritium activity from the HE to a hypothetical member of the public. In some cases, these assumptions are conservative to the point of being unrealistic. Accordingly, the doses calculated using "worst-case" assumptions should be considered to be upper bounds on the doses that may result from the unrestricted use of the HE. However, the doses calculated using the "realistic" assumptions should be considered to reflect the actual anticipated exposures conditions during the recycling of the HE.

The 50-year CEDE [End Note 1] per unit uptake of tritium activity is independent of the route of uptake. Also, the 10-day effective half-life [End Note 2] of tritium results in the 50-year CEDE being delivered within about 100 days. Thus, there is no significant difference between the first-year EDE (H_E) [End Note 3] and the 50-year CEDE. Therefore, the 50-year CEDE shall be referred to hereafter as the annual EDE. The general sequence of each calculation includes:

1. Determine the fraction of the activity available to be released to the environment or into the transfer media, e.g., air, liquid slurry, etc.;
2. Calculate the rate of uptake that results from the exposure to the contaminated media;
3. Calculate the total uptake of tritium; and
4. Calculate the annual EDE.

ASSUMPTIONS

The following assumptions and conditions, common to most of the analyses, include:

1. The "worst-case" volume tritium contamination is assumed to be $2 \times 10^4 \mu\text{Ci HTO/g HE}$ [NOTE: This value is based on the geometric mean, m_g , of $1.9 \times 10^4 \mu\text{Ci HTO/g HE}$ for all analytical results for bulk tritium activity in the HE sampled];
2. The "realistic-case" volume tritium contamination is assumed to be $1.0 \times 10^4 \mu\text{Ci HTO/g HE}$;
3. All of the tritium released from the HE is assumed to be in the oxidized form (HTO.) [Note: This assumption is probably realistic and maximizes the dose per unit intake of activity.];

4. An uptake via inhalation, ingestion, or skin absorption of 1 μCi of HTO is assumed to result in an annual EDE of 0.064 millirem (mrem). [Note: This value is taken from Federal Guidance Report #11 "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," and is based on the assumptions found in International Commission on Radiological Protection (ICRP) Publication 30 "Limits for Intakes of Radionuclides by Workers," including a Quality Factor (Q) [End Note 4] of 1.0.];
5. The breathing rate for "reference man" is assumed to be 0.02 cubic meters per minute (m^3/min). [Note: This value is specified in ICRP Publication 23 "Task Group Report on Reference Man" and ICRP Publication 30];
6. The total intake rate (inhalation plus absorption through the skin) for an exposure to airborne HTO is assumed to be 1.5 times the inhalation intake rate. [Note: This guidance is specified in ICRP Publication 30];
7. In scenarios involving ingestion, it was assumed that the fractional uptake from the gastrointestinal tract is 1.0, i.e., $f_1 = 1.0$; and
8. In scenarios involving absorption of radioactive material through the skin, it was assumed that no protective clothing was being worn and that normal clothing provided no protection against exposure to the contaminated liquid or vapor.

SCENARIOS

Scenario 1: Warehouse Storage Scenario

This scenario examines possible doses from storage of bulk quantities of HE in a poorly-ventilated warehouse. The assumptions include:

- a. The warehouse is 100-ft x 100-ft x 20-ft, giving a total volume of 2×10^5 cubic feet (ft^3) ($5.7 \times 10^3 \text{ m}^3$);
- b. A specified maximum amount of HE is stored in the warehouse at any one time. The stored amounts are:

● worst case:	40,000 lbs
● realistic case:	2,000 lbs;
- c. Without respect to the manner in which the HE is packaged or contained, the tritium is assumed to diffuse out of the HE at a linear rate of 0.01% per day (d), i.e., 0.01% of the total initial quantity of tritium is assumed to be released into the warehouse each day;
- d. The tritium is assumed to be quickly and uniformly distributed throughout the warehouse;
- e. The air in the warehouse is assumed to have a specified number of air changes per day as indicated:

● worst case:	10/d
● realistic case:	24/d;

- f. A person is assumed to occupy the warehouse for 240 d/yr during his/her work assignment as indicated:
- worst case: 8 hrs/d
 - realistic case: 1/3 hrs/d; and
- g. Uptake of tritium is assumed to take place via inhalation and absorption through the skin.

The interior of the warehouse is modeled as a single compartment with a constant rate of input of tritium. Tritium is removed from the warehouse atmosphere at a rate determined by the number of air changes per day. The concentration of tritium in the warehouse can be calculated by using a single compartment, constant input, and simple dilution model as described by the following equation, (after Till, 1983, pp. 5-5 and 5-6.)

$$C = \frac{Q}{V(\lambda + F/V)} [1 - e^{-(\lambda + F/V)t}] \quad \text{Eq. 1a}$$

where:

- C = tritium concentration in air ($\mu\text{Ci}/\text{m}^3$)
- Q = release rate ($\mu\text{Ci}/\text{d}$)
- V = volume of the warehouse (m^3)
- λ = tritium radioactive decay constant ($1.54 \times 10^{-4}/\text{d}$)
- F = flow rate of air out of the warehouse (m^3/d)
- e = base of the natural logs (2.7183)
- t = number of days (d)

In such a case, the tritium concentration rapidly reaches an equilibrium concentration that is directly proportional to the input rate and inversely proportional to the volume of the warehouse and the number of air changes per day. At equilibrium, the concentration may be expressed as:

$$C_{\text{eq}} = \frac{Q}{V(\lambda + F/V)} \quad \text{Eq. 1b}$$

where:

- C_{eq} = tritium equilibrium concentration in air ($\mu\text{Ci}/\text{m}^3$)

The annual EDE may be calculated as follows:

$$H_E = C_{\text{eq}} \text{ Br (60) (T) D (1.5)} \quad \text{Eq. 1c}$$

where:

H_E	=	annual EDE (mrem)
C_{eq}	=	tritium equilibrium concentration in air ($\mu\text{Ci}/\text{m}^3$)
Br	=	breathing rate ($0.02 \text{ m}^3/\text{min}$)
60	=	conversion: (min/hr)
T	=	number of hours worked per year (hrs/yr) (d/yr x hrs/d)
D	=	dose conversion factor ($0.064 \text{ mrem}/\mu\text{Ci HTO}$)
1.5	=	correction for additional uptake via skin absorption

Using a "worst-case" concentration of $2 \times 10^{-4} \mu\text{Ci HTO}/\text{g HE}$, 10-air changes/d, and an exposure time of 8 hr/d for 240 d/yr, the resulting annual EDE would be 1.4×10^{-3} mrem.

Using the "realistic" concentration of $1 \times 10^{-4} \mu\text{Ci HTO}/\text{g HE}$, 24-air changes/d, and an exposure time of 20 min/d for 40 d/yr, the resulting annual EDE would be 5.5×10^{-7} mrem.

Scenario 2: Exposure During Unpacking of Sealed Containers of HE

This scenario examines the doses that might be received by a worker who is unpacking sealed containers of HE in a poorly ventilated environment. The assumptions include:

- a. Each container holds 50-lbs of contaminated HE;
- b. At equilibrium, 0.1% (0.001) of the total activity in 50-lbs of HE becomes trapped in a "dead air" space inside the container;
- c. Upon opening the container, the activity within this dead air space is dispersed into the volume of air immediately surrounding the worker;
 - worst case: 10 m^3 volume of air
 - realistic case: 20 m^3 volume of air;
- d. Worker is exposed to this concentration of tritium for 1 minute, during which time no further dilution of tritium takes place; and
- e. Worker typically opens the following number of containers/d as indicated:
 - worst case: 10/d
 - realistic case: 1/d; and
- f. Worker performs this task as indicated:
 - worst case: 240 d/yr
 - realistic case: 40 d/yr.

The annual EDE may be calculated as follows:

$$H_E = \frac{(454) M F Br T C_e D (1.5)}{V} \quad \text{Eq. 2}$$

where:

H_E	=	annual EDE (mrem)
454	=	number of grams per pound (g/lb)
M	=	mass of HE in pounds (lbs)
F	=	fraction of activity released (0.001)
Br	=	breathing rate (0.02 m ³ /min),
T	=	number of minutes exposed per year (min/yr) (1 min/container x number opened/d x work d/yr;
C_v	=	volume tritium concentration in HE (μ Ci HTO/g HE)
D	=	dose conversion factor (0.064 mrem/ μ Ci HTO)
1.5	=	correction factor to account for uptake of tritium by absorption through the skin
V	=	volume of air into which the tritium is released (m ³)

In the "worst-case" where the average concentration is assumed to be $2 \times 10^4 \mu$ Ci HTO/g HE, the volume of air available for mixing is assumed to be 10 m³, 10 containers/d are opened, and the worker performs this task for 240 d/yr, the annual EDE is 2.1×10^{-3} mrem.

Using a "realistic" concentration of $1 \times 10^4 \mu$ Ci HTO/g HE and assuming the worker opens 1 container/d for 40 d/yr, the resulting annual EDE would be 8.7×10^{-6} mrem.

Scenario 3: Worker Exposure to Dust-Generating Operations

In this scenario, a worker is assumed to be exposed to a cloud of contaminated HE dust of unspecified origin. Intake is assumed to occur through inhalation of the dust and via absorption of contamination through the skin. The assumptions include:

- a. The worker is exposed to a cloud of contaminated HE dust at a dust loading of:
 - worst case: 10 mg/m³
 - realistic case: 1 mg/m³;
- b. Uptakes of tritium occur via inhalation and via absorption through the skin;
- c. Respiratory protection equipment is used by the worker. The Respiratory protection factor (RPF) of the equipment is as indicated:
 - worst case: 1
 - realistic case: 10; and
- d. The worker performs this task as indicated:
 - worst case: 8 hrs/d for 240 d/yr
 - realistic case: 1 hr/d for 40 d/yr

The resulting annual EDE may be calculated as follows:

$$H_E = L C_v (0.001) Br (60) T D (1.5) \quad \text{Eq. 3}$$

where:

H_E	=	annual EDE (mrem)
L	=	airborne dust loading (mg/m^3)
C_v	=	volume tritium contamination per gram of HE ($\mu\text{Ci HTO}/\text{g HE}$)
0.001	=	conversion from milligrams to grams (g/mg)
Br	=	breathing rate ($0.02 \text{ m}^3/\text{min}$)
60	=	number of minutes per hour (min/hr)
T	=	number of hours exposed per year (hrs/yr) ($\text{d}/\text{yr} \times \text{hrs}/\text{d}$)
D	=	dose conversion factor ($0.064 \text{ mrem}/\mu\text{Ci HTO}$)
1.5	=	correction for additional uptake via skin absorption

In the "worst-case", using a concentration of $2 \times 10^{-4} \mu\text{Ci HTO}/\text{g HE}$, a dust loading of $10 \text{ mg}/\text{m}^3$, and a RPF of 1, the annual EDE would be $4.6 \times 10^{-4} \text{ mrem}$.

In the "realistic" case, using a concentration of $1 \times 10^{-4} \mu\text{Ci HTO}/\text{g HE}$, a dust loading of $1 \text{ mg}/\text{m}^3$, and a RPF of 10, the annual EDE would be $4.6 \times 10^{-5} \text{ mrem}$.

Scenario 4: Adding Wet PBX to HE Slurry

In this scenario, a worker is assumed to be standing over an opening in the vat while he/she manually adds wet PBX to the HE slurry. [NOTE: With the exception of adding wet PBX to the vat, the dissolution/recrystallization process occurs in a closed system.] Intake is assumed to occur through the inhalation of the of tritiated water vapor (HTO) and via absorption of contamination through the skin. The assumptions include:

- a. An operator is exposed to HTO evaporating from HE slurry in a mixing vat;
- b. The air to which the operator is exposed is saturated with water vapor (100% humidity);
- c. The concentration of the HTO water vapor to which the operator is exposed to is:
 - worst case: 1%
 - realistic case: 1%;
- d. There is no further ventilation or dilution;
- e. The operator performs this task as indicated:
 - worst case: 8 hrs/d for 240 d/yr
 - realistic case: 1/3 hrs/d for 100 d/yr; and
- f. Uptakes of tritium occur via inhalation and absorption through the skin.

The resulting annual EDE from a year of operation may be calculated as follows:

$$H_E = F_m C_v L_w F_{wv} Br (60) T D (1.5) \quad \text{Eq. 4}$$

where:

- H_E = annual EDE (mrem)
- F_m = mass fraction of HE in the slurry (0.3 g HE/g slurry)
- C_v = volume tritium contamination ($\mu\text{Ci HTO/g HE}$)
- L_w = airborne water vapor loading (25 g $\text{H}_2\text{O/m}^3$)
- F_{wv} = fraction of water vapor that is contaminated (0.01 or 0.1)
- Br = breathing rate (0.02 m^3/min)
- 60 = conversion for minutes to hour (min/hr)
- T = number of hours exposed per year (hrs/yr)
(hrs/d x d/yr)
- D = dose conversion factor (0.064 mrem/ $\mu\text{Ci HTO}$)
- 1.5 = correction for additional uptake via skin absorption

For the "worst-case" assumptions of $2 \times 10^4 \mu\text{Ci HTO/g HE}$ and 1% HTO contribution to water vapor, the annual EDE would be 3.3×10^{-3} mrem.

For the "realistic" assumptions of $1 \times 10^4 \mu\text{Ci HTO/g HE}$ and 1% HTO contribution to water vapor, the annual EDE would be 2.9×10^{-5} mrem.

Scenario 5: Inadvertent Ingestion of HE Slurry

This scenario examines the doses which would result if small quantities of tritium-contaminated slurry were splashed on the hands, or on or near the mouth and inadvertently ingested. The assumptions include:

- a. The HE slurry is 30% (0.3 g HE/g slurry) tritium-contaminated HE;
- b. The quantity of slurry inadvertently ingested each day that the slurry is handled is:
 - worst case: 0.01 g/d
 - realistic case: 0.001 g/d; and
- c. The worker inadvertently ingests some of the slurry each day that he/she works as follows:
 - worst case: 240 d/yr
 - realistic case: 12 d/yr.

The resulting annual EDE from a year of such exposure may be calculated as follows:

$$H_E = M F_m C_v D Y_d \quad \text{Eq. 5}$$

where:

- H_E = annual EDE (mrem)
- M = mass of slurry ingested each day (g/d)
- F_m = mass fraction of HE in the slurry (0.3 g HE/g slurry)
- C_v = volume tritium contamination (μ Ci HTO/g HE)
- D = dose conversion factor (0.064 mrem/ μ Ci HTO)
- Y_d = number of days worked per year

For the "worst-case" concentration of 2×10^{-4} g HTO/g HE and assuming a daily intake rate of 0.01 g of slurry, the annual EDE would be 9.2×10^{-6} mrem.

For the "realistic" concentration of 1×10^{-4} g HTO/g HE) and assuming a daily intake rate of 0.001 g of slurry, the annual EDE would be 4.6×10^{-7} mrem.

Scenario 6: Skin Absorption due to Splashing of HE Slurry

In this exposure scenario, it is assumed that a worker is splashed with the slurry of contaminated liquid. The assumptions include:

- a. The tritium-contaminated HE comprises 30% (0.3 g HE/g slurry) of the slurry;
- b. The density of the slurry is 1.5 grams per cubic centimeter (g/cm^3);
- c. During handling operations, portions of the worker's body is inadvertently wetted with slurry as follows:
 - worst case: hands and forearms ($3,500 \text{ cm}^2$)
 - realistic case: hands and forearms ($3,500 \text{ cm}^2$)

[Note: The surface area values for the entire body and for the hands and forearms are taken from ICRP 23.];
- d. The slurry is allowed to remain on the skin before it is washed off
 - worst case: 10 min
 - realistic case: 5 min; and
- e. This inadvertent splashing takes place to the same worker one time each day as follows:
 - worst case: 240 d/yr
 - ALARA case: 24 d/yr

The uptake of tritium activity from wetted skin and the corresponding dose is calculated using the following equation (Osborne, 1968):

$$U = MWN + WIPNT \quad \text{Eq. 6a}$$

where:

- U = uptake of tritium into the body per splashing incident ($\mu\text{Ci HTO}$)
- M = intake rate of liquid due to the "blotter effect" (1 g/m^2)
- W = wetted surface area (m^2)
- N = specific activity per unit mass of the contaminated liquid ($\mu\text{Ci HTO/g HE}$)
- I = intake rate for skin ($5.1 \mu\text{Ci}/\text{min}/\text{m}^2/\mu\text{Ci}/\text{L}$) (Osborne, 1968)
- P = humidity at skin temperature (0.04 g/L)
- T = exposure time (min)

Using units of cm^2 for wetted skin area and $\mu\text{Ci}/\text{L}$ for tritium activity concentration in the slurry, Eq. 6a can be re-written as:

$$U = [(1.0 \times 10^{-7}) A C] (1 + 0.204 T) \quad \text{Eq. 6b}$$

where: all of the terms are as defined above, except that:

- A = wetted surface area (cm^2)
- C = concentration of tritium in the liquid slurry ($\mu\text{Ci HTO}/\text{L}$)
- T = exposure time (min)

For example, if the values stated in the above assumptions are used with $2 \times 10^4 \mu\text{Ci HTO}/\text{g HE}$, i.e., concentration of tritium in the liquid, the annual uptake of tritium may be calculated as follows:

$$\begin{aligned} C &= 2 \times 10^4 \mu\text{Ci HTO}/\text{g HE} * 0.3 \text{ g HE}/\text{g slurry} * 1.5 \text{ g}/\text{cm}^3 * 1000 \text{ cm}^3/\text{L} \\ &= 9.0 \times 10^2 \mu\text{Ci HTO}/\text{L} \end{aligned}$$

and

$$\begin{aligned} U &= (1.0 \times 10^{-7} * 3,500 \text{ cm}^2 * 9.0 \times 10^2 \mu\text{Ci HTO}/\text{L}) * [1 + (0.204 * 10)] \\ &= 9.6 \times 10^{-5} \mu\text{Ci HTO}/\text{splashing incident} \end{aligned}$$

For the "worst-case" where the arms and forearms are splashed once each day for the 240 working days in a year, the corresponding annual EDE would be 1.5×10^{-3} mrem.

For the "realistic" case where the arms and forearms are splashed once each day for the 24 working days in a year, the corresponding annual EDE would be 4.9×10^{-5} mrem.

Scenario 7: Exposure to Dust Plume after HE Explosion

This scenario examines possible doses from exposure to the plume of contaminated dust that would be produced when the explosive is detonated. The "General Explosion" module of the

Lawrence Livermore National Laboratory (LLNL) HOTSPOT Health Physics Computer Codes (Homann, 1994) was used to model the airborne release of tritium in an explosion. This program uses a standard gaussian plume model. Both a "worst-case" scenario, using the maximum tritium concentrations and the worst-case meteorological conditions, and a "realistic" scenario were examined.

In the "worst-case," "realistic," and "ALARA" scenarios, the assumptions included:

- a. The mass of slurry involved in the explosion is as follows;
 - worst case: 100,000 lbs
 - ALARA case: 1,000 lbs
- b. The contaminated HE makes up 30% of the total mass of the slurry (0.3 g HE/g slurry);
- c. All of the tritium activity in the slurry was assumed to be available to, and distributed in, the explosion;
- d. The quantity of HE was artificially limited to 10-lbs [Note: This is conservative because it significantly reduces the height of the explosive cloud. Thus, The amount of dispersion initially taking place within the cloud is limited.]; and
- e. The nearest dose receptor, located at the plume centerline during the entire passage of the plume, is assumed to be a mining company employee at a "explosive safety perimeter" of about 300 meters downwind with speeds of:
 - worst case: 2 m/sec
 - realistic case: 4.5 m/sec; and
- f. The mining company employee was assumed to be exposed to such a plume for 50 wks/yr as indicated:
 - worst case: 1/wk
 - realistic case: 7/wk

In the "worst-case" scenario, the HE was assumed to have an average concentration of $2 \times 10^{-4} \mu\text{Ci HTO/g HE}$, the stability conditions were class D (neutral), and the wind speed was 2 m/s. The annual EDE for 50 weeks of such exposure is 9.5×10^{-5} mrem.

In the "realistic" scenario, the HE was assumed to have an average concentration of $1 \times 10^{-4} \mu\text{Ci HTO/g HE}$, the stability conditions were class D (neutral), and the wind speed was 4.5 m/s. The annual EDE for 50 weeks of such exposure is 1.5×10^{-6} mrem.

SUMMARY

Using very conservative assumptions, possible doses from seven credible pathways were examined. The results of the "worst-case" and "realistic" analyses are summarized in Table C1 and Table C2. The "realistic" analysis, which more accurately portrays the anticipated operations in the "life-cycle" of the recycle option during "normal" working conditions, showed that the maximum individual and collective EDEs from the radiological impacts for the recycle option are negligible. It should be noted that for these analyses the occupationally exposed workers at Pantex Plant are considered to be members of the public. Therefore, no attempt was made to exclude their occupational exposures from the recycle effort.

As shown for the "realistic" analysis (see Table C1), the maximum individual EDE for a worker at Pantex Plant, i.e., 3.8×10^{-5} mrem, is postulated to come from the preparation of the HE at Pantex Plant. For a manufacturer's worker, the highest credible annual EDE, i.e., 4.9×10^{-5} mrem, resulted from the HE slurry being splashed on the worker (scenario #6). These highest credible doses are much less than 1% of the applicable primary dose limit for members of the public of 100 millirem per year, contained in DOE 5400.5 and in Title 10 Code of Federal Regulation (CFR) Part 20 "Standards for Protection Against Radiation." They are well below the 30 mrem in a year DOE dose constraint for release of property (RESRAD).

A comparison of the collective EDE received by the members of the public for each scenario is presented in Table C2. It is estimated that about 75 members of the public, including Pantex Plant workers, would be impacted from radiological exposures resulting from the recycle option. The total collective EDE for the recycle option is about 1.5×10^3 person-mrem. For comparison, these 75 workers are collectively exposed to $2 \times 10^{+4}$ person-mrem per year, assuming an "average" exposure to natural background radiation of 360 mrem per year.

Scenario	"Worst-Case" (mrem)	"Realistic-Case" (mrem)
A: Pantex Preparation (See Note)	6.8×10^3	3.8×10^3
1: Warehouse Storage	1.4×10^3	5.5×10^7
2: Unpacking of HE	2.1×10^3	8.7×10^4
3: Mixing/Crushing (Airborne)	4.6×10^4	4.6×10^4
4: Adding Wet PBX to Slurry	3.3×10^3	2.9×10^3
5: Ingestion of Slurry	9.2×10^4	4.6×10^7
6: Skin Splashed with Slurry	1.5×10^3	4.9×10^3
7: Dust Plume from Explosive Use	9.6×10^3	1.5×10^4

NOTE: With the exception of the "worst-case" dose equivalent, the values used for "Pantex Preparation" are simply the sum of the values for scenarios #1, #2, and #4 of Appendix C.

Scenario	Annual Individual EDE (mrem)	Sites	Workers per Site	Annual Collective EDE (person-mrem)
A: Pantex Preparation	3.8×10^3	1	10	3.8×10^4
1: Warehouse Storage	5.5×10^7	2	10	1.1×10^8
2: Unpacking of HE	8.7×10^4	2	10	1.7×10^4
3: Mixing/Crushing (Airborne)	4.6×10^4	2	5	4.6×10^7
4: Adding Wet PBX to Slurry	2.9×10^3	2	5	2.9×10^4
5: Ingestion of Slurry	4.6×10^7	2	5	4.6×10^4
6: Skin Splashed with Slurry	4.9×10^3	2	5	4.9×10^4
7: Dust Plume from Explosive Use	1.5×10^4	5	25	1.9×10^4
Total Collective EDE (person-mrem)				1.5×10^8

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APPENDIX C END NOTES

- 1: The committed effective dose equivalent (CEDE) is the sum of the committed dose equivalents (CDEs) to various tissues in the body ($H_{T,50}$), each multiplied by the appropriate weighting factor (W_T) — that is, $H_{E,50} = \sum W_T H_{T,50}$. The CDE is the dose equivalent calculated to be received by a tissue or organ over a 50-year period after intake of the radionuclide into the body. It does not include any contributions from radiation sources external to the body.
- 2: The effective half-life (T_E) is the time required for a radioactive element deposited in the body to be diminished by 50 percent as the result of the combined action of radioactive decay (T_R) and biological elimination (T_B). It is calculated as:
$$T_E = (T_R \times T_B) / (T_R + T_B).$$
- 3: The effective dose equivalent (H_E) is the quantity that is used to assess the risk to an individual from both uniform whole body and non-uniform partial body exposures. It is the sum of the weighted dose equivalents for irradiated tissues or organs, i.e., $H_E = \sum W_T H_T$, where W_T is the weighting factor for tissue T and H_T is the average dose equivalent received by tissue T.
- 4: The quality factor is used in radiation protection in an attempt to account for the effectiveness of different radiations to produce biological effects. Q is the factor by which the radiation absorbed dose (rad) is multiplied to yield the dose equivalent (rem) of the exposure.

APPENDIX D

COMMITTEE MEMBERS

Chairman:

**David Styers, P.E., CHP - Pantex Plant
Waste Management & Environmental Restoration Division**

Mr. Styers has over 25 years experience in program management that includes test planning, system design, training, research and development, and quality assurance/quality control. Mr. Styers' expertise is in operational and technical radiation health physics. He has conducted field surveys, safety reviews, hazard assessments, and compliance reviews for various private industries and state/federal governments, including DOE/NRC nuclear facilities. He is knowledgeable in gamma spectroscopy analyses and rules, regulations, guidelines, and standards. He is the author of over fifteen technical papers and publications and has made several presentations at DOE technical conferences.

Mr. Styers' managerial positions have included Radiation Protection Program Section Manager, ALARA/Radiological Engineer, Project/Task Manager, and Health Physics Team Leader for various companies in Oak Ridge, TN. Presently, he is the Program Scientist for Battelle-Pantex's Waste Management and Environmental Restoration Division.

He is certified (comprehensive) by the American Board of Health Physics. He is certified as a Professional Civil Engineer in the state of Pennsylvania (PE-023269-E). He has a Bachelor of Sciences in Education degree in Chemistry and Physics from Slippery Rock State College and a Masters of Science degree in Health Physics from Georgia Institute of Technology.

Members:

**Harry Flaugh - Los Alamos National Laboratory
Consultant - Delphi Group, Inc.**

Mr. Flaugh retired from Los Alamos National Laboratory (LANL) in November, 1993 after nearly 32 years in various weapons-related activities. He has operational and managerial experience in explosives development, testing, and product specification; hydrodynamic and nuclear testing; nuclear vulnerability and effects; and systems engineering, testing, and product specification. He served as a Group Leader for an Explosive Chemistry Group and as a Group Leader and Deputy Division Leader for a Weapons System Engineering Group. He also served as a Program Manager for

Weapons Production and Surveillance and (Nuclear Weapons) Dismantlement. He presently is a consultant for Delphi Group, Inc.

He was a major contributor to the development and application of insensitive high explosives (IHE) for nuclear weapons. He participated in a joint DoD/DOE study of IHE sponsored by the National Research Council. He served for many years on the DOE Accident Response Team, most recently as a Senior Scientific Advisor, and participated in exercises in US, UK, and Italy. He also participated in information exchanges with UK, France, and Russia. He served in the group that wrote the DOE Explosives Safety Manual and on the tri-Laboratory/DOE-AL Steering Group for tritium operations during Limited Life Component Exchange.

He has a Bachelor of Science degree and Master of Science degree in Chemical Engineering and Nuclear Engineering from Iowa State University.

**James Humphrey - Lawrence Livermore National Laboratory
Defense Technology Engineering Group**

Mr. Humphrey has served over 35 years at the Lawrence Livermore National Laboratory (LLNL) and LANL. Presently, Mr. Humphrey is a member of the Futures Group in the Defense Technology Engineering Group at LLNL. He is widely published and has specialized in the fields of actinide purification and plutonium chemistry; high explosives formulation, characterization, and application; conventional and nuclear weapon systems materials design applications, analyses, integration, and tritium storage and transfer.

Mr. Humphrey has held managerial positions as an Assistant Deputy Associate Director for Advanced Conventional Weapons, High Explosive Technology Leader and Plutonium Technology Leader, Special Technical Advisor to the Director of the Office of Munitions/Tactical Warfare Program/Office of the Under Secretary of Defense (Acquisition), Chemistry and Materials Science Project Leader for Strategic Nuclear Weapons, Deputy Task Force Leader for the Demilitarization of LLNL nuclear weapons, DOE's Manager of the Energetic Materials Waste Minimization Program and Manager of the Joint DoD/DOE Memorandum of Understanding for Advanced Conventional Munitions. Mr. Humphrey serves, and has served, on several national committees for energetic materials and plutonium.

**Gary Mansfield, CHP - Lawrence Livermore National Laboratory
Hazards Control Group**

Mr. Mansfield has over 16 years experience in operational and technical health physics, primarily in the areas of tritium, plutonium, uranium, and nuclear fuel reprocessing health physics. Mr. Mansfield's specific expertise is in the areas of tritium health physics, internal radiation dosimetry, radiological safety analysis, and

health physics issues related to nuclear weapons. He has reviewed and assisted in the development of several American Nuclear Standards Institute (ANSI) standards related to internal dosimetry.

He is and has been the Health Physicist responsible for the Lawrence Livermore National Laboratory (LLNL) Tritium Facility for the last six years. He provides back-up health physics and policy development support for all LLNL radiological operations. He has also acted as the LLNL Internal Dosimetry Program Coordinator for the last seven years. He was the Health Physicist, Internal Dosimetrist, and Health Physics Manager at the Idaho Chemical (nuclear fuel) Reprocessing Plant at the Idaho National Engineering Laboratory.

He is certified (comprehensive) by the American Board of Health Physics. He has a Bachelor of Arts degree in Physics and a Master of Science degree in Radiological (Health) Physics.

**Benny Richardson, Ph.D - Pantex Plant
Applied Technologies Division**

Mr. Richardson has a Doctor of Philosophy degree in Analytical Chemistry from Texas A&M University. He has been involved for the last 15 years in work with quality control analyses of high explosives and materials characterization. He is presently a Sectional Manager in the Materials Characterization Section of the Applied Technologies Division at the Pantex Plant. He currently serves as Chairman of the Panhandle-Plains Section of the American Chemical Society (ACS) and is an Adjunct Professor in the Math, Physical Sciences, and Engineering Technology Department of West Texas A&M University. He is the author of fourteen technical reports and seven professional publications in refereed journals. He has made numerous presentations at DOE technical conferences, local universities/colleges, and ACS meetings.

**Tony Woltermann, Ph.D - Pantex Plant
Applied Technologies Division**

Mr. Woltermann has a Doctor of Philosophy degree in Chemistry from the University of Cincinnati. He has over 28 years of industrial experience that has involved work with radioactive materials, most notably plutonium and tritium. He was Director of Technology at EG&G Mound Facility. He was the Manager for the Environmental Sciences Center of Monsanto, Co. Currently, he is the Division Manager for the Applied Technologies Division at Pantex Plant. He is the author of over thirty technical publications and presentations.

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Pantex Plant

P. O. BOX 30020

AMARILLO, TEXAS 79177

806-477-3000

JUL 27 1995

Mr. Joe Martillotti
Pantex Special Project Coordinator
Texas Natural Resource Conservation Commission
P.O. Box 13087
Austin, Texas 78711-3087

Thru: *V* Vince Zehrowski, Lead *WAW 8/1/95*
Waste Management/Environmental Protection Team
U.S. Department of Energy
Amarillo Area Office
Amarillo, Texas 79177

Re: Health-Based Release Criteria for Contaminated High Explosives

Dear Mr. Martillotti:

This enclosed documents listed below are for your review and comment.

1. "Health-Based Release Criteria for Bulk Tritium-Contaminated High Explosives," dated July 7, 1995
2. "Analysis to Support an As Low As Reasonably Achievable (ALARA) Determination for Disposition of Tritium-Contaminated High Explosives," dated July 7, 1995

If you have any questions, please contact Gary G. Baker at (806) 477-4440.

Very truly yours,

W. A. Weinreich
W. A. Weinreich
General Manager *WAW*

WAW:sdd
Attachments: As stated

QM:95- 01593-790

BE:95:02086
PS:95:0103

To: Frank Sprague
From: Dave Styers *D. Styers*
Date: July 7, 1995

Subject: Changes to Health-Based Release Criteria, dated May 8, 1995

The following changes have been made to the May 8, 1995, submittal. Changes in the text are indicated in italics.

I. HEALTH-BASED RELEASE CRITERIA

Global Changes:

1. The maximum annual EDE for Pantex Plant worker has been changed to 5.2×10^5 mrem to 3.8×10^5 mrem (math error).
 - a. Page 2, Paragraph 1, Line 2
 - b. Page 16, Paragraph 4, Line 1
 - c. Page 46, Paragraph 2, Line 3
 - d. Page 46, Table C1 - C2, Scenario #1
2. The term "man-mrem" has been changed to "person-mrem."
3. Reference to the "ALARA analysis" has been changed to "analysis to support an ALARA determination."
4. In Appendix C, the original "worst-case" analyses have been deleted in their entirety. Therefore, the original "realistic-case" has been changed to the "worst-case." The original "ALARA-case" is has been changed to the "realistic-case."

Specific Changes:

1. Page 2, Paragraph 2, Line 1: "Based upon the "As Low As Reasonably Achievable analysis, annual" to "Based upon the analysis to support an ALARA determination,annual."
2. Page 16, Paragraph 1, Line 4/Page 47, Table C1 - C2, Scenario #4: "Manual Mixing of HE water slurry" to "Adding wet PBX to HE water slurry."
3. Page 16, Paragraph 2, Line 5: The following has been added: "However, the "life-cycle" of the HE for "normal operational conditions."
4. Page 17, Paragraph 1, Line 2: The following has been added: "....mrem. This anticipated exposure resulted from the skin being splashed with HE slurry (scenario #6)."

5. Page 17, Paragraph 1, Line 4/Page 46, Paragraph 2, Line 8:
The following statement has been added:
"They are well below the 30 mrem in a year DOE dose restraint for release of property (RESRAD)."
6. Page 18, Paragraph 5, Line 5: "The results compared to the removable surface contamination limit of 1000 dpm/100 cm₂ as promulgated in DOE 5400.5." to "The results ... compared to an authorized limit for removable surface contamination consistent with DOE 5400.5 requirements."
7. Page 20/Page 48: Reference for RESRAD was added.
8. Page 35, Paragraph 5, Assumptions: Assumption 1 was deleted. (See global change #4 above.) All other assumptions were renumbered. The following NOTE was added at the end of the new assumption 1:
"[NOTE: This value is based on the geometric mean, m_g , of 1.9×10^{-4} μ Ci HTO/g HE for all analytical results for bulk tritium activity in the HE sampled.]"
9. Page 41, Scenario 4 title: "Manual Mixing of HE water slurry" to "Adding wet PBX to HE water slurry."
10. Page 41, Paragraph 1, Lines 1 - 2: "In this scenario, over an open vat while he/she manually stirs the HE slurry." Intake tritiated" to "In this scenario, ... over an opening in the vat while he/she manually adds wet PBX to the HE slurry. [NOTE: With the exception of adding wet PBX to the vat, the dissolution/recrystallization process occurs in a closed system.]" Intake tritiated."

II. ALARA ANALYSIS

Global Changes:

1. The maximum annual EDE for Pantex Plant worker has been changed to 5.2×10^{-5} mrem to 3.8×10^{-5} mrem (math error).
 - a. Page 3, Paragraph 5, Line 2
 - b. Page 11, Table 2 - 3, Scenario #1
2. The term "man-mrem" has been changed to "person-mrem."
3. Reference to the "ALARA analysis" has been changed to "analysis to support an ALARA determination."
4. Consistent with Appendix C of the main report, the original "worst-case" analyses have been deleted in their entirety. Therefore, the original "realistic-case" has been changed to the "worst-case." The original "ALARA-case" is has been changed to the "realistic-case."

Specific Changes:

1. Title has been changed from "As Low (ALARA) ... High Explosives" to "Analysis to Support an(ALARA) Determination Explosives."
2. Table of Contents has been added.
3. Page 10 -11, Tables 1 - 3, Scenario #4: "Manual Mixing of HE water slurry" to "Adding wet PBX to HE water slurry."
4. Page 14, Paragraph 2, Line 5: The following has been added to the end of Line 5:

"... $\mu\text{Ci HTO/g HE}$ and had removable tritium contamination less than 1,000 disintegrations per minute over an area of 100 square centimeters (dpm/100 cm^2 ."

07-07-95 01:53p
Free: 1,037,824

Directory A:*.*

.	Current	<Dir>	..	Parent	<Dir>
ALARA	.BK	54,998 07-07-95 01:53p	ALARA	.HE	54,998 07-07-95 01:52p
RELEASE	.BK	154,208 07-07-95 01:52p	RELEASE	.HE	154,208 07-07-95 01:52p

Backup

memorandum

DATE: June 26, 1995

REPLY TO
ATTN OF:

Office of Environmental Policy and Technical Assistance: Wallo, EH-412

SUBJECT:

Release Criteria for Volume Tritium Contamination in High Explosives

TO: Frank Sprague, AL

EH-41 has reviewed the proposal contained in the May 30, 1995, Memorandum from N. Dienes (AL) to T. O'Tool (EH-1). In general, the technical analysis is acceptable and supports your proposal. We are prepared to recommend that EH-1 approve the release limits subject to the following:

- o While the attached report includes a section titled "As Low As Reasonably Achievable (ALARA) Analysis for Disposition of Tritium Contaminated High Explosives" the section is not an ALARA analysis. This analysis is really more of an uncertainty analysis of the dose estimates than a true ALARA analysis. It does, however, contain the most of the material necessary to support an ALARA decision but it should be titled "Analysis to Support an ALARA Determination." An ALARA analysis should look at several options and either quantitatively or qualitatively determine the optimum value for the release criteria.
- o To satisfy the ALARA requirement of DOE 5400.5, an AL signed (you or your management as appropriate) memorandum should be included with package. The memorandum should discuss the proposal and the benefits and costs of going selecting a lower limit or a higher limit. Based on the report and dose assessment which shows very low individual and collective doses associated with this action, and the recommendations provided in our March 1991 ALARA implementation guidance, the determination for this action can be qualitative or semi-quantitative. Therefore, the AL memorandum should simply provide a brief statement of the action (to recycle about 50,000 pounds of HE per year using 1000 dpm/100 cm² as a standard for release) and indicated that the recycle effort is generally beneficial to the environment, saves the Department resources through the sale of the explosives and will cause insignificant incremental doses to the public even under conservative assumptions. The memorandum should indicate that doses to the maximally exposed worker or member of the public was estimated to be about 5×10^{-5} mrem from the activity at the proposed standard and that collective doses are estimated to be less than 0.002 person-mrem. The incremental doses associated with the activity are 7 orders of magnitude lower than background and are far below all radiation protection requirements. Because potential doses are already extremely low, lowering the limit will have no impact on public health or welfare but would increase implementation difficulties substantially. Given the supporting data, raising the limit will not substantially increase costs. Therefore, the proposed limit represents the appropriate ALARA-based value for this activity.



- o The memorandum should also indicate that AL has coordinated with the appropriate state regulator and that they are not opposed to the action.
- o Page 17, paragraph 1. Add "It is well below the 30 mrem in a year DOE dose constraint for release of property." to the end of the paragraph.
- o Page 18, paragraph 5. Change the fourth sentence to read "... analysis are compared to an authorized limit for removable surface contamination derived consistent with Order DOE 5400.5 requirements."
- o Appendix and ALARA analysis. There was considerable confusion with the discussion of the various scenarios and the estimated doses. The report discussed "worst case", "realistic" and "ALARA" estimates. The ALARA values should represent the realistic estimates. It appears that the "realistic" case actually represented a realistic upperbound. If that is correct it should be so identified and clarified. If not than please explain the difference between ALARA and realistic cases. Given that this document be an example for other studies it is important that it be clear as possible. While not essential, if possible to complete a global replacement, "man-mrem" should be changed to "person-mrem."
- o Page 46, second paragraph. Change the 7th line from "promulgated in" to "contained in."

Once we receive your memorandum documenting the ALARA decision, we will submit the package to EH-1 for approval. If you have any questions please contact me at 202-586-4996.



Andrew Wallo
Director
Air, Water, and Radiation Division

United States Government

Department of Energy

memorandum

Albuquerque Operations Office

MAY 30 1995

DATE:

REPLY TO

OESH (N. S. Dienes/505-845-6121)

ATTN OF:

SUBJECT:

Proposal to Use a Health-Based Limit for the Release of Volume Tritium Contaminated High Explosives for Recycling**TO: T. J. O'Toole, Assistant Secretary for Environment, Safety and Health, EH-1, HQ**

This is an item requiring action by your office. In accordance with DOE Order 5400.5, "Radiation Protection of the Public and the Environment," Chapter II, Section 5.c.(6), it is requested that you approve the criteria and survey techniques for release of volume tritium contaminated high explosives (HE) for recycling. The criteria and survey techniques are described in the attached report.

This report was first submitted to you for approval on November 7, 1994. Your staff and the staff of the Albuquerque Operations Office (AL) worked together to resolve comments on this original submittal. This report has been modified to address the comments from your staff.

The AL Environmental Protection Division (EPD), the Waste Management Division (WPD), and the Occupational Safety and Health Division (OSHD) concur with the report and recommend the recycle of the HE that meets the release criteria as prescribed in the report.

The criteria, as described in the report, is well within the established parameters stated in the Tritium Surface Contamination Limits Committee Report published by the Battelle Northwest Laboratories, "Recommended Tritium Surface Contamination Release Guides," February 1991. The Committee recommended that "... If the tritium contamination level is below 1/10th the release limit given in Section 8 (1×10^9 disintegrations per minute per 100 centimeter²) (i.e., 100,000 dpm/100 cm²), then the item can be released..." The criteria to be used per the report is 1000 dpm/100 cm² which is 1/100th of the Committee's recommendation of 10,000 dpm/100 cm².

T. J. O'Toole

2

MAY 30 1995

Use of this safety margin and the resulting negligible impact on the environment and public health ensures the release of the HE would meet the purpose and intent of DOE Order 5400.5 for release of materials and equipment having residual radioactive material. This proposal makes recycling of this HE a realistically viable choice.

The Pantex Plant and EPD staff are requesting a meeting with you and your staff, at your convenience, to answer specifics of the proposal or provide needed clarification if it is necessary. A prompt response to this request would be appreciated.

If you have any questions, need further clarification, or wish to arrange a meeting, please contact Constance L. Soden, Director, EPD, at (505) 845-5586, or Frank H. Sprague at (505) 845-4340.



Nicholas S. Dienes
Assistant Manager for
Technical Management and Operations

Attachment

cc w/attachment:

A. Wallo, EH-412, HQ
G. W. Johnson, AAO
G. E. Runkle, OSHD, AL
M. S. Bange, WMD, AL
H. T. Season, Jr., WPD, AL

cc w/o attachment:

D. W. Pearman, Jr., FM-1, HQ

EXAMPLE - Roofing Material to Landfill

memorandum

DATE: November 22, 1995

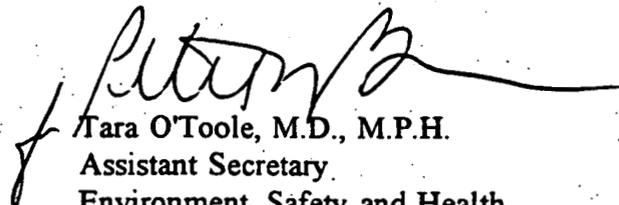
REPLY TO
ATTN OF: Air, Water and Radiation Division:Wallo:586-4996:EH-412

SUBJECT: Approval for release (disposal) of 105 KE Roofing Material

TO: John D. Wagoner, Manager, DOE/RL

This is in response to your November 6, 1995, request for EH approval to dispose of 105 KE and KW roofing material in the Hanford on-site landfill. We have reviewed your supporting material and recommendation and have determined that the data and analysis support your recommendation to dispose of the subject waste in the on-site landfill. EH-41 staff had a number of comments on the analysis attached to your memorandum. However, we have concluded that none of the issues identified would change the conclusion of your analysis. We are attaching the comments for the record.

The disposal of the 105 KE and KW roofing material in the on-site landfill is consistent with the Department's radiological protection requirements. If you have any questions regarding this approval or the attached comments, please contact Mr. Andrew Wallo, EH-412, at 202-586-4996.


Tara O'Toole, M.D., M.P.H.
Assistant Secretary
Environment, Safety and Health

Attachment

memorandum

DATE: NOV 17 1995

REPLY TO
ATTN OF: Air, Water and Radiation Division:Wallo:586-4996:EH-412

SUBJECT: Recommendation to approve release (disposal) of Roofing Material

TO: Tara O'Toole, EH-1

Thru: Raymond Berube, EH-4

On September 15, 1995, the Department's Richland Operations Office (DOE/RL), consistent with the requirements of Order DOE 5400.5, requested EH-1 approval to permit the disposal of some slightly contaminated roofing material in the Hanford site central landfill. EH-41 provided comments on the package on September 25, 1995, and DOE/RL resubmitted the material to EH-1 on November 6, 1995. EH-41 received a copy of the request on November 13, 1995, and confirmed that all previous concerns have been resolved. Based on our review of the data provided, the disposal action recommended by DOE/RL meets all radiation protection requirements and protects the public consistent with the Department's as low as reasonably achievable process.

DOE/RL proposes to dispose of approximately 300,000 kg of roofing material that contains less than 0.6 mCi of Cs-137 and Sr-90 and lesser amounts of Co-60 and Am-241. The most contaminated portions of this material (Area#1 of 105 KE Basin roof) was estimated to have concentrations of less than 17 pCi/g Cs-137.

Maximum individual doses associated with this disposal were conservatively estimated to be less than 0.001 mrem/year for the worst case intruder scenario which is more than four orders of magnitude less than the maximum recommended dose constraint of 25 mrem per year. DOE/RL also considered exposures to the resident-farmer living on the landfill and estimated doses on the order of 0.00001 mrem/year. It is highly unlikely that any individual will ever receive even these low doses as a result of exposure to the subject material.

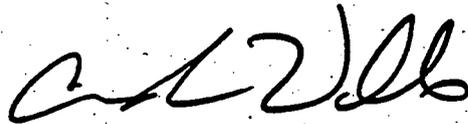
There is no realistic mechanism for collective exposures. However, under worst case situation evaluated by DOE/RL, the collective dose integrated over a 200 year period would be a small fraction of a person-rem (about 0.3 person-mrem).

The only viable alternative to disposal of this material in the central landfill is to dispose of it as radioactive waste. The incremental cost increase for this disposal would be about \$350,000 and the maximum benefit (collective dose reduction) would be less than a person-rem (0.3 person-mrem) which indicates that implementing the alternative disposal

would cost \$1.2 billion per person-rem avoided. We would have used a somewhat different scenario for this assessment which may have produced a slightly lower dollar per person-rem avoided estimate for the alternative low-level waste disposal, however, that estimate would have still exceed a billion dollars per person-rem. Both estimates are well above the \$1,000 to \$10,000 per person-rem avoided range typically used in ALARA-based decision-making.

EH-41 has several comments regarding the DOE/RL submittal that are attached for the record. However, while these concerns would be useful to address in future assessments, for the most part, they relate to overly conservative assumptions and unneeded analyses, and would in no way impact the decision. Therefore, we concur with the RL finding that the recommended approach is consistent with the as low as reasonably achievable process and believe it is protective, and meets the public radiation protection requirements.

We recommend that EH-1 sign the attached memorandum approving the DOE/RL request.



Andrew Wallo
Director
Air, Water and Radiation Division

Attachments

**EH-41 Summary Comments on DOE/Richland Request for
Approval to Dispose of 105 KE Roofing Material, Revision 1 (dated November 6, 1995)**

The analyses and survey data provided in the attachment to the subject request are adequate to demonstrate compliance with DOE requirements for disposal of the KE 105 roofing material in the on-site landfill. The ALARA considerations have been adequately addressed. The following comments resulted from the review of the supporting material.

Although the survey results are adequate to document the acceptability of the material for disposal in the landfill, it would have been desirable to have a more representative distribution of sampling locations. As noted below, the result of this is that the average concentrations of radionuclides in the roofing material may have been overestimated which in turn results in an overestimate of the source term. The survey design could have benefitted from the use of the Data Quality Objectives process.

It appears the estimate of mean concentrations for the radionuclides in the material was based on the arithmetic mean. Our review of the data suggests that the geometric mean may have been more representative. The use of the arithmetic mean in this case would generally overestimate the source term.

In this analysis, DOE/RL included a fairly detailed assessment of the effects using age-specific dose factors. While the analysis is laudable, it is far more detailed than is required for this situation. Given the low doses, the hypothetical nature of the scenarios for exposure and the uncertainties of the scenario ever occurring, an assessment using dose factors for "reference man" is more than sufficient.

The estimate of collective doses assumed ultimate transport of the residual radioactive material to the Columbia River and the exposure of 5 million people. The estimate of collective dose was based on the ratio of individual dose to collective dose from a low-level waste site. This estimate is extremely conservative for several reasons, a major one being that the source term for the low-level waste site is much greater than for the landfill site which would result in a much lower collective dose. The DOE/RL analysis corrected for the difference by ratioing the individual and collective doses. Ratios of the source term to collective dose would have provided a more representative screening tool. Despite this fact, given the small collective doses, the assessment is sufficient for a screening assessment. Because this site is not a DOE low-level waste disposal site, there is no requirement for DOE to continue to control the property. Therefore, another alternative that could have been considered was the loss of institutional control of the site and the use of the landfill area or neighboring areas for residential use. A hypothetical scenario assuming that the area could support one or a few families who obtained their drinking water from the ground water could have been used to estimate collective doses over a few hundred years (e.g., 200 years, the 1000 year period used by DOE/RL is conservative and more than would be warranted). The

incremental collective dose from this scenario, was extremely small and of no consequence to the decision. This scenario would not be appropriate if the location of the landfill is such that the control of the site must be maintained to ensure protection of the public from residual activity in neighboring areas. If site land use plans require continued control of the site, scenarios for estimating doses need not consider loss of institutional control for extended periods of time.

A cost/benefit analysis was completed to demonstrate that the proposed option met the ALARA process requirements. The cost/benefit analysis table in the text provides an estimate of cost per individual dose averted. This is an interesting value but is of little use in evaluating the cost effectiveness of an action. It adds little to the evaluation. The ALARA process uses an assessment of dollars per collective dose averted to assess the incremental benefit of the alternatives. The paragraph following the table contains that information. However, we could not recreate the value indicated in the report. Given an incremental benefit of 1.6 person-mrem over 1000 years and an incremental cost between disposal options of \$345,000, the estimated cost of a person-rem averted was about \$200 million dollars. This clearly would be in excess of ALARA requirements. However, given that the estimate of collective dose was based on data associated with doses from a low-level waste disposal site the incremental doses are not representative. Our assessment of costs per dose averted by disposal in a low-level waste site using the residential scenario discussed above as the basis for the collective dose were well in excess of a hundred million dollars per person-rem. Therefore, DOE/RL's conclusion that the landfill disposal represents the best practicable alternative considering ALARA is justifiable.

The DOE/RL analysis did not specifically consider ground-water protection. However, based on the results of the all pathway analysis, potential ground-water impacts are very low. There is no risk of causing drinking water standards to be exceeded in drinking water systems using the ground water.

Although it would have been desirable to address some of these comments in the final package, given the extremely low doses associated with this action, we do not believe it worthwhile to require Richland to modify the package as our analysis indicates none of the comments would require DOE/RL to reconsider the recommendation. Therefore, we are recommending that these comments be attached to the record with the approval of the request.

United States Government

Department of Energy
Richland Operations Office

memorandum

DATE: NOV 6 1995
REPLY TO:
ATTN OF: QSH:DCW
SUBJECT: REQUEST APPROVAL FOR THE RELEASE OF THE 105 KE ROOFING MATERIAL, REVISION 1

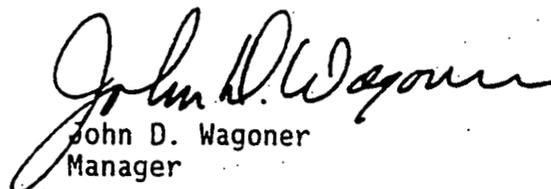
TO: Tara O'Toole
Assistant Secretary for
Environment, Safety, and Health
EH-1, HQ

In my memorandum to you regarding this subject, dated September 15, 1995, RL's Quality, Safety, and Health Programs Division (QSH) requested approval of a dose assessment for the release of the 105 KE and KW roofing material. The dose assessment package was reviewed by HQ and comments on the dose assessment were returned by memorandum dated September 25, 1995. These comments were incorporated into the revised dose assessment, revision 1, which is attached. Revision 1 calculated a dose assessment of less than 0.001 mrem/yr for the disposal of the roofing material in the central landfill. Ground water protection and an As Low As Reasonably Achievable review, which includes a cost-benefit analysis, are also contained in the revised assessment.

In accordance with DOE Order 5400.5, it is requested that the HQ Program Office approve the disposal of the 105 KE basin and transfer area roofing material for unconditional disposal in the Hanford onsite central landfill. The 105 KE and KW office area roofing materials, which were part of the September 15, 1995, request, have been disposed of in the central landfill since no radioactivity could be detected in these materials.

In consideration of the findings of the revised dose assessment, I recommend that EH approve the 105 KE basin and transfer area roofing material for disposal at the onsite central landfill.

If you have any questions concerning this matter, please call me, or your staff may contact Dana C. Ward, QSH, at (509) 372-1261.


John D. Wagoner
Manager

Attachment

cc w/attach:
R. P. Berube, EH-20
F. Cole, EM-37
A. Wallo, EH-232

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**DOSE ASSESSMENT FOR THE VOLUMETRIC RELEASE
OF THE 105 KE ROOFING MATERIAL, REVISION 1****Purpose**

This document analyzes the potential radiological dose that could result from the release of the Westinghouse Hanford Company (WHC) 105 KE Basin and Transfer Area roofing material to the Hanford Central Landfill (CLF). This is a case specific assessment and dose not apply to any other potential releases.

The U.S. Department of Energy (DOE) has the authority to establish criteria and limits for the protection of the public and the environment for disposal in a DOE on-site landfill which is not an authorized low-level waste disposal facility. Disposal of such material must conform to the requirements of DOE Order 5400.5 (DOE 1993) including the ALARA process (As Low As Reasonable Achievable). This criteria should be such that it is not likely that disposal of materials into the landfill will result in a future requirement for remediation of the landfill. To assure that this is achieved, authorized limits for material sent to the CLF must be approved by DOE-HQ.

The criteria must be based on an assessment under the ALARA process to optimize the balance between risks and benefits including costs and collective doses and selected to ensure that individual doses to the public are less than 25 mrem in a year with a goal to reduce potential individual dose to a few millirem in a year. This dose assessment must ensure that the ground water will be protected. There are no known other possible sources of residual radioactive material being sent to the CLF.

Description

The 105 KE Basin and transfer Area roof became contaminated via unfiltered roof vents. Cracks in the roofing material have allowed rain water to wash this radioactivity into the substructure creating low levels of volumetrically contaminated material. Attachment 2 includes maps of the Basin and Transfer roofs with sampling locations indicated with circles. These maps were previously used for other non-radiological testing and have the roof panels numbered. The shaded blocks are from that assessment and should be ignored.

In addition to this dose assessment, all roofing material was surveyed to assure it met DOE Order 5400.5, Chapter IV, Figure IV-1, Surface Contamination Guidelines (DOE 1993), for release to the public prior to being released to the CLF. The U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.86 (NRC 1974) values were used for Transuranics. Any material with radioactivity exceeding either the contact or removable release criteria DOE Order 5400.5 or NRC Regulatory Guide 1.86 was disposed of as radioactive waste. Additionally, all these materials were analyzed using gamma spectroscopy, no radionuclides were identified beyond those of the characterization study. The radioanalysis was performed before this disposal, therefore the concentrations determined are conservative.

Sample collection points concentrated around the roof vents where surface radioactivity had been detected and along the roof edge where rain water pooled. These samples were taken such as to represent the contamination in the substrata of the roofing material. There were 51 core samples taken from the Basin roof and 42 taken from the Transfer Area roof.

These samples were analyzed using gamma spectroscopy for the purpose of detecting the WHC Volumetric Release Criteria. The table of release criteria from WHC procedures is shown in Attachment 2 (WHC 1995). This criteria was approved by DOE-HQ, EM-30 (Memorandum, Lytle 1993), for use in determining if hazardous waste is radiologically contaminated and is therefore mixed waste. This criteria was developed to meet the DOE "no-rad-added" policy (Office of Environmental Management, Performance Objective 1994). The WHC submitted the same criteria to DOE, Richland Operations Office (RL), as interim volumetric release criteria (Memorandum, Dixon 1995) until final volumetric release criteria can be developed and approved.

Attachment 2 includes a summary of the radioanalysis for the roofing material. The roof was divided into three areas so that the ALARA process can be better applied. Please note that any of these three areas may be disposed of as separate units.

The first area consists of the roofing material exceeding the volumetric release criteria. This material, denoted by red dots on the attached map, consists of material whose radiological concentration exceeded that of the volumetric release criteria. This material was found only on the Basin roof and was located around the roof vents (RV 10 & RV 11).

Each of the 9 analysis performed were averaged together to determine a mean concentration. The Cs-137 radioactivity concentration was determined to be 16.7 pCi/g and the Am-241 radioactivity concentration was found to be 0.273 pCi/g. The samples that did not indicate the presence of Am-241 were averaged into the mean as zeros because the criteria for the Minimum Detectable Activity was no-rad-added or zero. The only significant radionuclide in the source term that is not identifiable with gamma spectroscopy is Sr-90. The 105 K Basins Area Sr-90 concentrations are slightly less than the Cs-137 concentrations (WHC 1994); therefore, it was conservatively assumed that Sr-90 would be at the same 16.7 pCi/g level as Cs-137. No Co-60 was detected on the Basin roof; therefore, it was assumed not to be present.

All of the roofing material has been placed in 4X4X8 feet low-level burial boxes. These boxes, when placed in the CLF, will be covered with one foot of soil. At the closure of the CLF an additional three feet of soil will be placed on the boxes for a total of four feet (1.22 m). The roofing material consists mostly of felt insulation, with tar paper, tar and gravel. A density of 1.5 grams per cubic centimeter was assumed. An absorbent was added to solidify any remaining water in the felt.

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This first area of material was placed in four (4) burial boxes. The four burial boxes have a volume of $2.18E4$ kilograms and a burial area of 11.9 meter².

The second area consists of the remaining roofing material from the Basin roof. The analysis of this material indicated a mixture of material with identified radioactivity below the release criteria (red dots with blue centers) and material with undetectable radioactivity (green dots). It is assumed that the distribution of material identified but below the release criteria and material with undetectable contamination was constant throughout the entire roof. Therefore, these samples were averaged together with zeros to determine the mean radioactivity concentration in this area of the roof. Only Cs-137 was detected in this area and at a concentration of 2.59 pCi/g. Sr-90 was assumed to be at the same 2.59 pCi/g level as the Cs-137. As no Am-241 or Co-60 was detected in this area, they are assumed not to be present. This second area of roofing material was placed into 34 burial boxes. The 34 burial boxes have a volume of $1.85E8$ grams and a burial area of 101 meter².

The third area consists of the Transfer Area roof. The analysis of this roofing material indicated no radioactivity exceeding the volumetric release criteria. The analysis indicated a mixture of material with identified radioactivity but below the release criteria (red dots with blue centers) and material with undetectable radioactivity (green dots). Three of the samples indicated the presence of Am-241 and one sample the presence of Co-60. All four of these samples were adjacent to the suspected source of contamination, the roof vents (RV 6 & RV 7). It is assumed that the presence of Am-241 and Co-60 were much less in the remaining area of the roof; therefore, the concentration of these samples was averaged and then divided by 5, the estimated fraction of the roof area adjacent to the roof vents. It is assumed that the distribution of Cs-137 is uniform throughout the entire roof between material identified but below the release criteria and material with undetectable contamination; therefore, these samples were averaged together with the zeros to determine the mean radioactivity concentration. Cs-137 was detected at a concentration of 1.60 pCi/g. Sr-90 was assumed to be at the same 1.60 pCi/g level as the Cs-137. The Am-241 radioactivity concentration was calculated to be 0.074 pCi/g and the Co-60 was calculated to be 0.0814 pCi/g. This third area of roof material was placed into 17 burial boxes. The 17 burial boxes have a volume of $9.25E7$ grams and a burial area of 50.5 meter².

SUMMARY OF RADIOACTIVITY

Radionuclide	Concentration pCi/g	Volume g	Radioactivity uCi	Burial Area m ²
Area #1: 105 KE Basin, Above Release Criteria				
Cs-137	16.7	2.18E7	365	11.9
Sr-90	16.7	2.18E7	365	11.9
Am-241	0.273	2.18E7	5.95	11.9
Area #2: 105 KE Basin, Below Release Criteria				
Cs-137	2.59	1.85E8	47.9	101
Sr-90	2.59	1.85E8	47.9	101
Area #3: 105 KE Transfer Area				
Cs-137	1.60	9.25E7	148	50.5
Sr-90	1.60	9.25E7	148	50.5
Am-241	0.0704	9.25E7	6.51	50.5
Co-60	0.0814	9.25E7	7.53	50.5

RESRAD Doses

The RESRAD dose assessment model was used to calculate individual and collective doses. Both realistic best estimates and worst case estimates were developed. Many possible exposure scenarios for released material have been developed and are available for analyses. The scenarios used here are typically for residential, construction, or recycle activities, including special consideration for potential groundwater transport for those few radionuclides that might have more impact through that transport pathway.

These dose projections for the waste packages to be stored at the CLF are based on two calculations. The first uses RESRAD (Yu 1993) to estimate potential doses with no actual intrusion. An individual lives on top of the buried waste and receives a small amount of external dose from penetrating photon radiation. In addition, he has a well which supplies his household needs. The well enters the unconfined aquifer down-gradient of the waste site. The second method for evaluating potential doses assumes that this individual penetrates the waste boxes in the course of drilling a well. The material exhumed is assumed to be indistinguishable from soil and is spread about in a garden. Post-Drilling dose factors involving a delay of 100 years were taken from the 200 West Area Burial Ground Performance Assessment (Wood 1994). Because of the small area available (Maximum 101 m²), only one

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individual is considered in these scenarios. Multiple individuals will be considered in the collective dose evaluation.

Dose Factors - No Intrusion

The RESRAD program Version 5.60 was used to calculate dose per unit concentration in waste to be stored at the CLF. Geologic data from the Interim Site Characterization Report (Fruiland 1989). In particular, the top 24.4 meters (80 feet) is Hanford Formation sand, while the remaining 13.7 meters (45 feet) down to the unconfined aquifer is Hanford Formation gravel. The aquifer is also gravel. The observed hydraulic gradient is 0.0001.

Hanford Site data prepared for the DOE programmatic Environmental Impact Statement (EIS) was used for the porosity and saturated hydraulic conductivities (Schramke 1994). The effective porosities were assumed to be 0.3 for both soil types. The unsaturated hydraulic conductivities were assumed to be 1 percent of the saturated conductivities. These values are summarized in the table below. The "b" parameter used by RESRAD was assumed to be 4.0 to keep the saturation ratio away from 1.0.

Porosity and Saturated Hydraulic Conductivities

Parameter	Hanford Sand	Hanford Gravel	Aquifer
Soil Density	1.60 g/cc	1.76 g/cc	1.76 g/cc
Hydraulic Conductivity	15.8 m/y	4.8 m/y	480 m/y
Total Porosity	0.42	0.36	0.36
Effective Porosity	0.30	0.30	0.30
Saturation Ratio	0.59	0.66	1.00

Two other important parameters are depth in the aquifer at which the water is extracted and the volume of water pumped from the well each year. The default value of 10 meters below the water table was used for the first parameter. The aquifer is generally more than 20 meters thick, thus the RESRAD default was appropriate and conservative. The volume of water pumped from the well was based on water consumption by a family of four and irrigation needs for a 500 m² area. The average water need per person is 65 gallons/day or 90 m³/y (Miller 1980). The typical irrigation rate in the counties near Hanford is 82 cm/y. This value was used in prior Hanford performance assessments. It is based on the Specific Information on the Terrestrial Environment database referenced by Baes et al (1984) for irrigation in the western states. The Hanford Environmental Dose Reconstruction Project (HEDR) found the irrigation rate in the counties surrounding the Hanford Site ranged from 61 cm/y to 98 cm/y (Snyder et al. 1994). The total needed to water a 500 m² area for a growing season is 410 m³. Thus the assumed annual water need for this family is 770 m³/y. The RESRAD "Non-Dispersive" intake dilution model was used at the well.

Soil distribution coefficients came from the Hanford Site data in (Kaplan 1995). These values are the smallest listed in Table 6.1 of that document.

The annual precipitation for Hanford is 0.16 m/y. In addition, the RESRAD "evapotranspiration" coefficient was set to 95 percent. The RESRAD runoff coefficient was set to 0.1 for these calculations.

Soil Distribution Coefficients

Nuclide	Co-60	Sr-90	Cs-137	Am-241	Np-237	U-233	Th-229
Kd, ml/g	1200	5.0	540	70	2.4	0.08	40

The assumed waste area was 101 m². Parameters for the inhalation and food chain pathways represent negligible contribution to the final results. Most of the dose from Am-241 came 3270 years from now, when the principal progeny, Np-237, reaches the well. For Np-237 the dominant pathway is drinking water. The consumption of 730 L/y was used. Most of the dose from the other nuclides came from direct radiation through the cover in the first year. The external dose was based on being exposed 20 percent of the year, and using the RESRAD area adjustment factors. Dose factors are shown below.

Peak Dose from a Unit Waste Concentration (mrem/y per pCi/g)

Nuclide	Co-60	Sr-90	Cs-137	Am-241
Peak Year	0	0	0	3270
Dose Factor	3.4E-06	1.5E-12	2.8E-08	1.4E-04

RESRAD output graphs and tables are shown in Attachment 2. The dose from Am-241 depends on the land area used by the waste. This is because the larger area represents a larger source entering the groundwater. The dose is proportional to the area. The dose factors for the other nuclides are not affected by the area because the waste is covered with 1.22 meters of soil. Only a small portion of the source actually contributes to the computed dose. The remainder is shielded by the soil.

Dose Factors - Intrusion

The methods used in the performance assessment calculations are described in detail elsewhere (Rittmann 1994). Rittmann gives unit dose factors for the post-intrusion cases which give dose per curie exhumed after a delay of 100 years. The delay represents the time from waste disposal to release to the public so that a well could inadvertently be drilled. These 100 year dose factors are shown below. The second line in the table accounts for the well diameter (30 cm) and waste thickness (1.22 meters) and average waste density

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(1.5 g/cm³). It turns out that 129 kg waste is exhumed in this intrusion scenario.

Intruder Dose from a Unit Waste Concentration

Nuclide	Co-60	Sr-90	Cs-137	Am-241
Exhumed, mrem per Ci	1.78E-2	8.47E+1	2.41E+2	6.37E+2
Dose Factor, mrem/y per pCi/g	2.3 E-9	1.1 E-5	3.1 E-5	8.2 E-5

Note that these dose factors are independent of the land area used by the waste. All that matters is the waste concentration and the amount of waste exhumed and spread over the garden. The area only affects the probability of a well driller encountering the waste packages.

Age Dependence

The internal dose factors for the various age groups are shown below. Also shown are the ratios with adult dose factors. Human food consumption and breathing rates at various ages are also summarized. Values were taken from the U.S. Federal Food and Drug Administration Draft (1991), Yang and Nelson (1986) and the ICRP (ICRP 56). The ingestion dose factors for infants are larger than for adults; however, this is off-set by the reduced intakes. Dose factors for Co-60 were not shown but would be similar to Cs-137 and Sr-90.

Internal Dose Factors, mSv per Bq Intake (to age 70)

Nuclide	Ingestion Effective Dose Equivalent (EDE)					
	0 to 1y	1 - 2 y	2 - 7 y	7 - 12y	12-17 y	Adult
Sr-90+D	1.3E-04	9.1E-05	4.1E-05	4.3E-05	6.7E-05	3.5E-05
Cs-137+D	2.0E-05	1.1E-05	9.0E-06	9.8E-06	1.4E-05	1.3E-05
Np-237+D	5.5E-03	4.9E-04	4.3E-04	4.0E-04	4.7E-04	4.5E-04
Am-241	1.2E-02	1.2E-03	1.0E-03	9.0E-04	9.1E-04	8.9E-04

Nuclide	Inhalation Effective Dose Equivalent (EDE)					
	0 to 1y	1 - 2 y	2 - 7 y	7 - 12y	12-17 y	Adult
Sr-90+D	1.3E-04	9.1E-05	4.1E-05	4.3E-05	6.7E-05	3.5E-05
Cs-137+D	2.0E-05	1.1E-05	9.0E-06	9.8E-06	1.4E-05	1.3E-05
Np-237+D	5.5E-03	4.9E-04	4.3E-04	4.0E-04	4.7E-04	4.5E-04
Am-241	1.2E-02	1.2E-03	1.0E-03	9.0E-04	9.1E-04	8.9E-04

D = Daughters

Ratio with Adult Internal Dose Factors

Nuclide	Ingestion Ratios				
	0-1y	1-2y	2-7y	7-12y	12-17y
Sr-90+D	3.71	2.60	1.17	1.23	1.91
Cs-137+D	1.54	0.85	0.69	0.75	1.08
Np-237+D	12.22	1.09	0.96	0.89	1.04
Am-241	13.48	1.35	1.12	1.01	1.02

Nuclide	Inhalation Ratios				
	0-1y	1-2y	2-7y	7-12y	12-17y
Sr-90+D	2.00	1.67	1.05	1.02	1.48
Cs-137+D	1.51	0.88	0.70	0.76	1.01
Np-237+D	1.55	1.22	1.02	0.95	1.07
Am-241	1.55	1.36	1.18	1.00	1.00

D = Daughters

Average Dietary Intakes, kg/yr, from EPA

	<1 y	1-4 y	5-9 y	10-14y	15-19y	20-24y	25-29y	30-39y	40-59y	60+y
All Dairy	207.6	152.7	180.1	186.1	167.1	112.6	98.2	86.4	80.8	90.6
milk only	99.3	123.1	162.9	166.7	147.8	96.5	79.4	66.8	61.7	70.2
Egg	1.8	7.2	6.2	7.0	9.1	10.3	10.2	11.0	11.4	10.5
Meat	16.5	33.7	46.9	58.4	69.2	71.2	72.4	73.4	70.7	56.3
Fish	0.3	2.5	4.0	4.9	6.1	6.8	7.6	7.1	8.0	6.3
Produce	56.6	59.9	82.3	96.0	97.1	91.4	99.1	101.5	115.3	120.9
Grain	20.4	57.6	79.0	90.6	89.4	77.3	78.4	73.7	70.2	67.1
Beverage	112.1	271.2	314.3	374.1	453.0	541.7	558.8	599.3	632.2	564.7
water only	62.3	158.6	190.2	226.4	242.6	239.6	226.2	232.3	268.4	278.3
Misc	2.0	9.3	13.3	14.8	13.9	10.9	11.9	12.5	13.3	13.0
Totals:	417.3	594.1	726.2	831.9	904.7	922.1	936.6	965.0	1001.9	929.3
Pop Fracs:	0.014	0.0553	0.0689	0.0688	0.0686	0.0681	0.0675	0.1334	0.2489	0.2065

The last row "Pop Fracs" is the fraction of the population which falls into the age group listed. These can be used to combine groups to match the ICRP age categories (ICRP 56). They also indicate the size of the population at risk.

Average Breathing Rates and Ratios for Age Groups

Group	m ³ /d	Ratio
0-1y	0.8	0.035
1-2y	3.8	0.165
2-7y	9.3	0.404
7-12y	15.0	0.652
12-17y	19.5	0.848
>17y	23.0	1.000

The main pathway for the non-intrusion scenario is drinking water consumption. The predominate nuclide of concern is Np-237 with an infant to adult dose ratio of about 12. The adult consumes about 4 times more water than the infant; therefore, the infant drinking water dose would be about a factor of 3 greater than the adult's.

For the intruder case, the inhalation pathway dominates with Am-241 (factor of about 1.5). The adult breathing rate is about 30 times more than the infant; therefore, for this scenario, the infant inhalation dose would be about a factor of 0.05 less than the adult's. The overall result is that the larger infant dose factors are offset by the lower exposure factors. The infant doses are comparable to or considerably lower than the adult doses.

Dose Estimates for Specific Areas

Three areas need to be considered. The first involves 4 boxes, the second 34 boxes, and the third 17 boxes. These three are summarized in the table below. The land area occupied by the waste boxes is shown with each area. In all areas the projected doses are negligible.

Area 1: KE Basin, Above Release Criteria 11.9 m²

Nuclide	pCi/g	Scenario	
		RESRAD mrem/y	Intruder mrem/y
Sr-90	16.7	2.5E-11	1.8E-04
Cs-137	16.7	4.7E-07	5.2E-04
Am-241	0.273	4.5E-06	2.3E-05
Total		5.0E-06	7.3E-04

Area 2: KE Basin, Below Release Criteria 101.0 m²

Nuclide	pCi/g	Scenario	
		RESRAD mrem/y	Intruder mrem/y
Sr-90	2.59	3.9E-12	2.8E-05
Cs-137	2.59	7.2E-08	8.1E-05
Total		7.2E-08	1.1E-04

Area 3: KE Transfer Area 50.5 m²

Nuclide	pCi/g	Scenario	
		RESRAD mrem/y	Intruder mrem/y
Co-60	0.0814	2.8E-07	1.9E-10
Sr-90	1.6	2.4E-12	1.8E-05
Cs-13	1.6	4.5E-08	5.0E-05
Am-241	0.0704	4.9E-06	5.8E-06
Total		5.2E-06	7.3E-05

Collective Dose

There is no significant mechanism for a large population dose to result from the disposal of material in the CLF. Eventually, the contaminated groundwater will enter the Columbia River and give small doses to people using the river. In prior Hanford performance assessments (Wood 1994), the collective dose to 5 million persons drinking 2 L/d from the river as well as eating fish and irrigating with river water were found to be about 70 times greater than the individual dose (110 person-rem/y versus 1.6 rem/y). This factor of 70 can only be applied to Am-241 (worst case, 1.6E-3 person-mrem/y versus 2.3E-5 mrem/y). The other nuclides decay faster than they migrate through the soil. The dose from Am-241 would last over about 1000 years for a total dose

of 1.6 person-mrem. This dose is from a source term comparable to the radioactivity in a single smoke detector (5 uCi)!

Worst Case Dose

The doses are very small, so even unlikely changes to the exposure conditions will not be important. For discussion, it will be assumed that the soil cover is only half the expected amount. With 0.61 m soil over the waste packages, the short-term doses are significantly increased. The table below shows these "worst-case" dose factors. The increase relative to the more realistic case is shown as a ratio of dose factors.

Worst-Case Peak Dose from a Unit Waste Concentration (mrem/y per pCi/g)

Nuclide	Co-60	Sr-90	Cs-137	Am-241
Peak Year	0	0	0	3270
Dose Factor	7.2E-03	1.8E-07	3.0E-04	1.4E-04
Increase	2.1E+03	1.2E+05	1.1E+04	1.0E+00

The Am-241 dose is not affected by the thickness of the soil cover, since the doses are produced after the Np-237 progeny nuclide reaches the groundwater. Since the other nuclides primarily produce dose by direct radiation through the soil cover, the reduction in cover thickness also increases the dose.

Worst-Case Doses (mrem/y) for Specific Areas

Nuclide	Area 1	Area 2	Area 3	Total
Co-60	---	---	5.8E-04	
Sr-90	3.1E-06	4.7E-07	2.9E-07	
Cs-137	5.0E-03	7.7E-04	4.8E-04	
Am-241	4.5E-06	---	4.9E-06	
Totals	5.0E-03	7.7E-04	1.1E-03	6.87E-03

For the areas of interest, the increased dose factor results in larger doses. If all three areas are combined the resulting dose is 6.87E-03. This is 7.5 times higher than the highest realistic scenario (intruder) as summed in the following table.

Dose Assessment Summary

The doses from both of the scenarios for all the areas were summed resulting in a total effective dose equivalent (TEDE) of less than 0.001 mrem per year (9.1E-04 mrem/y).

DOSE ASSESSMENT

Area	Scenario	
	RESRAD	Intruder
Area #1; 105 KE Basin, Above Release Criteria	5.0E-6	7.3E-4
Area #2; 105 KE Basin, Below Release Criteria	7.2E-8	1.1E-4
Area #3; 105 KE Transfer Area	5.2E-6	7.3E-5
TEDE	1.0E-5	9.1E-4

Cost Benefit Analysis

The cost for burial at the Hanford Central Landfill is \$21/yd³ (\$0.78/ft³). The cost at the Hanford Low-level Radioactivity Burial Ground is \$50.00/ft³. This is a ratio of 64 times increase to bury as low-level radioactive waste. To bury each box in the CLF is \$100. To bury each box in the radioactive burial ground is \$6,400.00. The following table summarizes the costs for each area. The Intruder Scenario is used because it gives the highest dose.

COST/BENEFIT ANALYSIS

	Landfill Burial	Low-Level Radioactivity Burial	Cost (\$) per mrem/y Averted
Area #1; 105 KE Basin, Above Release Criteria	\$400	\$25,600	3.5E7
Area #2; 105 KE Basin, Below Release Criteria	\$3,400	\$217,600	1.9E9
Area #3; 105 KE Transfer Area	\$1,700	\$108,800	1.5E9
Total	\$5,500	\$352,000	3.8E8

The most dose averted would come from burying Area #1 in the Low-Level Radioactive Burial Ground. But even this option is not considered cost/beneficial from an ALARA standpoint. The collective dose of 1.6 person-mrem results in a cost/benefit analysis of 5.6E7 \$/mrem averted or \$56 Billion per rem averted. This demonstrates that it is definitely not ALARA to dispose of the 105 KE roofing material in a low-Level radioactivity burial ground.

Conclusions

A TEDE of less than 0.001 mrem/y is significantly less than 25 mrem/year, and within the guidelines acceptable to the State of Washington. The cost/benefit analysis clearly indicates that it is ALARA to bury all three roofing areas in the CLF at a savings of \$350,000. Accordingly, WHC requests that RL submit this dose assessment to DOE-HQ, for approval of the release of the 105 KE Basin and Transfer Area roofing material to the CLF.

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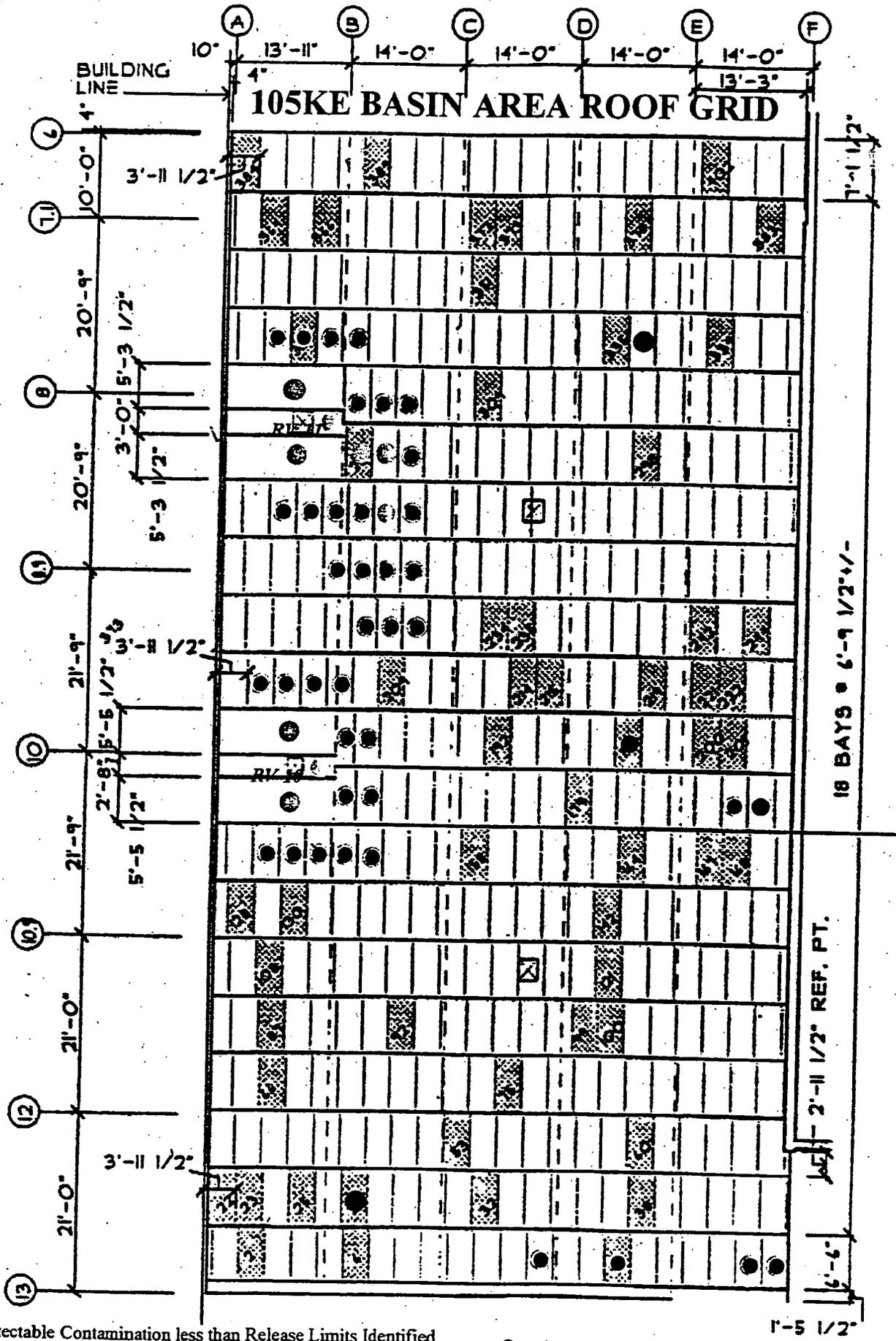
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Attachment 2

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13 Pages



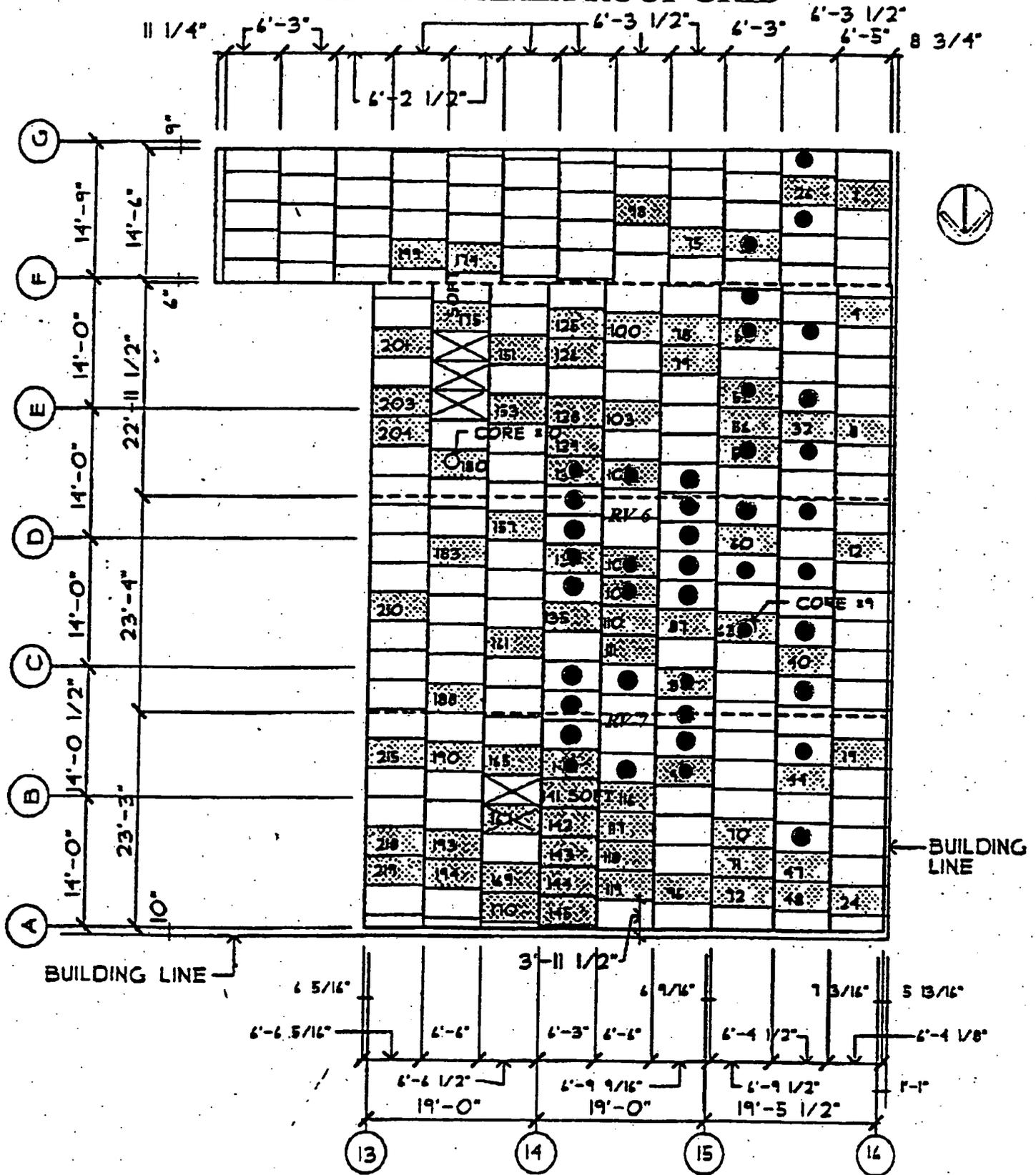
● Detectable Contamination less than Release Limits Identified in WHC Hazardous Waste Radiation Release Protocol

⊙ Detectable Contamination In Excess Of Release Limits Identified in WHC Hazardous Waste Radiation Release Protocol

○ Less Than Minimum Detectable Concentration Guidelines Identified in WHC Hazardous Waste Radiation Release Protocol

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105KE TRANSFER AREA ROOF GRID



- Less Than Minimum Detectable Concentration Guidelines Identified in WHC Hazardous Waste Radiation Release Protocol
- Detectable Contamination less than Release Limits Identified in WHC Hazardous Waste Radiation Release Protocol
- Detectable Contamination In Excess Of Release Limits Identified in WHC Hazardous Waste Radiation Release Protocol

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Gamma Spectroscopy Data

105KE Basin Roofing Material

Reviewed By: Thomas Bratvold Date: 08-16-95

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Laboratory Sample #	Sample Location	Measured Activity
K95-2806	Composite From S. of RV-11	2.268E+01 +/- 5.768E-01 pCi/g Cs-137 1.644E+00 +/- 6.821E-01 pCi/g Am-241
K95-2817	Composite From S. of RV-10	2.283E+01 +/- 6.729E-01 pCi/g Cs-137
K95-2706	Composite From E. of RV-10	7.470E+00 +/- 3.839E-01 pCi/g Cs-137 3.122E-01 +/- 1.289E-01 pCi/g Am-241
K95-2709	Composite From W. of RV-11	1.749E+01 +/- 5.245E-01 pCi/g Cs-137
K95-2707	Composite From W. of RV-10	1.010E+01 +/- 5.098E-01 pCi/g Cs-137 4.966E-01 +/- 1.644E-01 pCi/g Am-241
K95-2708	Composite From E. of RV-11	1.764E+01 +/- 6.901E-01 pCi/g Cs-137
K95-2703	105KE Basin Panel # 12	4.935E-01 +/- 1.173E-01 pCi/g Cs-137
K95-2702	105KE Basin Panel # 15	3.512E+00 +/- 3.067E-01 pCi/g Cs-137
K95-2701	105KE Basin Panel # 20	6.220E-01 +/- 1.180E-01 pCi/g Cs-137
K95-2700	105KE Basin Panel # 21	2.731E-01 +/- 1.339E-01 pCi/g Cs-137
K95-2836	105KE Basin Panel # 27	< MDC
K95-2816	105KE Basin Panel # 150	1.894E+00 +/- 1.623E-01 pCi/g Cs-137
K95-2813	105KE Basin Panel # 151	2.715E+00 +/- 2.224E-01 pCi/g Cs-137
K95-2807	105KE Basin Panel # 152	2.782E+00 +/- 3.550E-01 pCi/g Cs-137
K95-2811	105KE Basin Panel # 153	3.005E+00 +/- 2.065E-01 pCi/g Cs-137
K95-3025	105KE Basin Panel # 154	2.376E+00 +/- 2.562E-01 pCi/g Cs-137
K95-3027	105KE Basin Panel # 170	4.753E+00 +/- 4.618E-01 pCi/g Cs-137

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183KE Radiological Control Laboratory
Gamma Spectroscopy Data

105KE Basin Roofing Material

Reviewed By: ~~_____~~ Date: 08-16-95
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K95-2809	105KE Basin Panel # 171	4.340E+00 +/- 3.510E-01 pCi/g Cs-137
K95-2698	105KE Basin Panel # 185	6.163E-01 +/- 1.423E-01 pCi/g Cs-137
K95-2699	105KE Basin Panel # 186 (Near Reactor Wall)	< MDC
K95-2815	105KE Basin Panel # 186 (South Of RV-10)	4.608E+00 +/- 4.216E-01 pCi/g Cs-137
K95-3026	105KE Basin Panel # 187	2.708E+00 +/- 3.823E-01 pCi/g Cs-137
K95-2835	105KE Basin Panel # 197	< MDC
K95-2812	105KE Basin Panel # 204	1.783E+00 +/- 2.375E-01 pCi/g Cs-137
K95-2814	105KE Basin Panel # 205	1.734E+00 +/- 2.778E-01 pCi/g Cs-137
K95-2808	105KE Basin Panel # 206	2.343E+00 +/- 2.660E-01 pCi/g Cs-137
K95-2810	105KE Basin Panel # 207	1.267E+00 +/- 1.843E-01 pCi/g Cs-137
K95-3115	105KE Basin Panel # 229	1.180E+00 +/- 2.119E-01 pCi/g Cs-137
K95-3116	105KE Basin Panel # 230	6.299E-01 +/- 1.925E-01 pCi/g Cs-137
K95-3117	105KE Basin Panel # 231	9.923E-01 +/- 1.980E-01 pCi/g Cs-137
K95-3118	105KE Basin Panel # 250	2.018E+00 +/- 1.926E-01 pCi/g Cs-137
K95-3119	105KE Basin Panel # 251	1.057E+00 +/- 2.320E-01 pCi/g Cs-137
K95-3120	105KE Basin Panel # 252	1.534E+00 +/- 2.020E-01 pCi/g Cs-137
K95-3121	105KE Basin Panel # 253	1.160E+00 +/- 2.115E-01 pCi/g Cs-137
K95-2800	105KE Basin Panel # 270	2.192E+00 +/- 3.066E-01 pCi/g Cs-137
K95-2798	105KE Basin Panel # 271	1.912E+00 +/- 2.087E-01 pCi/g Cs-137
K95-2803	105KE Basin Panel # 272	6.965E+00 +/- 4.320E-01 pCi/g Cs-137
K95-2804	105KE Basin Panel # 273	3.421E+00 +/- 4.399E-01 pCi/g Cs-137

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K95-3028	105KE Basin Panel # 274	1.673E+01 +/- 5.375E-01 pCi/g Cs-137
K95-3122	105KE Basin Panel # 275	7.224E-01 +/- 2.106E-01 pCi/g Cs-137
K95-2802	105KE Basin Panel # 287	1.641E+01 +/- 4.906E-01 pCi/g Cs-137
K95-3029	105KE Basin Panel # 288	1.924E+01 +/- 5.754E-01 pCi/g Cs-137
K95-3123	105KE Basin Panel # 289	5.576E+00 +/- 3.312E-01 pCi/g Cs-137
K95-2796	105KE Basin Panel # 304	6.329E+00 +/- 4.621E-01 pCi/g Cs-137
K95-3030	105KE Basin Panel # 305	6.768E+00 +/- 4.625E-01 pCi/g Cs-137
K95-3124	105KE Basin Panel # 306	3.697E+00 +/- 3.192E-01 pCi/g Cs-137
K95-2805	105KE Basin Panel # 322	6.093E+00 +/- 3.846E-01 pCi/g Cs-137
K95-2797	105KE Basin Panel # 323	3.914E+00 +/- 2.786E-01 pCi/g Cs-137
K95-2801	105KE Basin Panel # 324	3.582E+00 +/- 2.503E-01 pCi/g Cs-137
K95-2799	105KE Basin Panel # 325	7.364E+00 +/- 3.697E-01 pCi/g Cs-137
K95-2834	105KE Basin Panel # 336	< MDC

< MDC (Minimum Detectable Concentration) as defined in WHC-EP-0063-4, Appendix J, Table J-1b.

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Laboratory Sample #	Sample Location	Measured Activity
K94-3231	105KE T/A Panel # 25	< MDC
K94-3242	105KE T/A Panel # 27	< MDC
K94-3240	105KE T/A Panel # 29	< MDC
K94-3243	105KE T/A Panel # 31	2.844E-01 +/- 1.762E-01 pCi/g Cs-137
K94-3228	105KE T/A Panel # 33	< MDC
K94-3229	105KE T/A Panel # 35	< MDC
K94-3238	105KE T/A Panel # 37	< MDC
K94-3230	105KE T/A Panel # 39	5.920E-01 +/- 1.616E-01 pCi/g Cs-137
K94-3241	105KE T/A Panel # 41	6.170E-01 +/- 2.381E-01 pCi/g Cs-137
K94-3236	105KE T/A Panel # 43	< MDC
K94-3244	105KE T/A Panel # 46	2.236E-01 +/- 9.597E-02 pCi/g Cs-137
K94-3232	105KE T/A Panel # 50	4.432E-01 +/- 1.266E-01 pCi/g Cs-137
K94-3233	105KE T/A Panel # 52	< MDC
K94-3239	105KE T/A Panel # 53	7.204E-01 +/- 3.548E-01 pCi/g Cs-137
K94-3235	105KE T/A Panel # 55	4.707E-01 +/- 1.627E-01 pCi/g Cs-137
K94-3234	105KE T/A Panel # 57	< MDC
K94-3237	105KE T/A Panel # 59	4.788E-01 +/- 1.939E-01 pCi/g Cs-137
K94-3226	105KE T/A Panel # 61	< MDC
K94-3227	105KE T/A Panel # 63	4.208E-01 +/- 1.768E-01 pCi/g Cs-137
K95-3023	105KE T/A Panel # 82	1.623E+00 +/- 1.900E-01 pCi/g Cs-137
K95-2825	105KE T/A Panel # 83	2.973E+00 +/- 2.788E-01 pCi/g Cs-137

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K95-2833	105KE T/A Panel # 84	3.700E+00 +/- 2.609E-01 pCi/g Cs-137 3.273E-01 +/- 1.208E-01 pCi/g Am-241
K95-2823	105KE T/A Panel # 85	1.969E+00 +/- 3.511E-01 pCi/g Cs-137
K95-2829	105KE T/A Panel # 86	1.228E+00 +/- 1.959E-01 pCi/g Cs-137
K95-2826	105KE T/A Panel # 89	5.868E-01 +/- 1.515E-01 pCi/g Cs-137
K95-2822	105KE T/A Panel # 90	3.667E+00 +/- 2.664E-01 pCi/g Cs-137 3.092E-01 +/- 1.236E-01 pCi/g Am-241
K95-2827	105KE T/A Panel # 91	1.713E+00 +/- 3.057E-01 pCi/g Cs-137
K95-2819	105KE T/A Panel # 92	1.257E+00 +/- 2.127E-01 pCi/g Cs-137
K95-2710	105KE T/A Panel # 105	3.000E+00 +/- 3.232E-01 pCi/g Cs-137
K95-2711	105KE T/A Panel # 108	6.072E+00 +/- 3.819E-01 pCi/g Cs-137
K95-2824	105KE T/A Panel # 109	1.273E+00 +/- 2.795E-01 pCi/g Cs-137
K95-2712	105KE T/A Panel # 112	1.604E+00 +/- 2.137E-01 pCi/g Cs-137
K95-2713	105KE T/A Panel # 115	5.064E-01 +/- 2.542E-01 pCi/g Cs-137
K95-3022	105KE T/A Panel # 130	6.009E+00 +/- 5.450E-01 pCi/g Cs-137
K95-2831	105KE T/A Panel # 131	7.321E+00 +/- 3.127E-01 pCi/g Cs-137 4.068E-01 +/- 1.228E-01 pCi/g Co-60
K95-2828	105KE T/A Panel # 132	3.590E+00 +/- 2.423E-01 pCi/g Cs-137
K95-2830	105KE T/A Panel # 133	1.071E+00 +/- 2.378E-01 pCi/g Cs-137
K95-2832	105KE T/A Panel # 134	1.820E+00 +/- 2.163E-01 pCi/g Cs-137
K95-3024	105KE T/A Panel # 137	6.054E+00 +/- 3.693E-01 pCi/g Cs-137 4.201E-01 +/- 1.797E-01 pCi/g Am-241
K95-2821	105KE T/A Panel # 138	4.117E+00 +/- 2.589E-01 pCi/g Cs-137
K95-2818	105KE T/A Panel # 139	9.031E-01 +/- 1.596E-01 pCi/g Cs-137

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K95-2820	105KE T/A Panel # 140	1.086E+00 +/- 4.579E-01 pCi/g Cs-137
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< MDC (Minimum Detectable Concentration) as defined in WHC-EP-0063-4, Appendix J, Table J-1b.

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Volumetric Release Criteria (WHC-EP-0063-4, Table J-1b and WHC-IP-0718, Section 3.1.1, Attachment 7.4)

Required Lower Limits of Detection for Radionuclides

ANALYSIS	LIMIT	UNITS	LIMIT	UNITS
	WATER		SOIL, OTHER	
Gross Alpha	5 [*]	pCi/l	5 [*]	pCi/g
Gross Beta	4 ^{**}	pCi/l	10 ^{**}	pCi/g
Gamma Emitters				
Co-60	50	pCi/l	10	pCi/g
Cs-137	50		10	
Eu-152	50		10	
Eu-154	50		10	
Eu-155	NA		2	
Ra-226	NA		2	
Ra-228	50		10	
Na-22	50		10	
Na-24	50		10	
Report Any Detectable Isotope				
Beta Specific Emitters				
Tritium	400	pCi/l	400	pCi/g
C-14	200	pCi/l	50	pCi/g
Ni-59	30	pCi/l	30	pCi/g
Ni-63	30	pCi/l	30	pCi/g
Se-79	***		***	pCi/g
Sr-89	5	pCi/l	10	pCi/g
Sr-90	2	pCi/l	10	pCi/g
Tc-99	30	pCi/l	30	pCi/g
I-129	25	pCi/l	25	pCi/g
Ra-228 (via Ac-228 daughter)	3	pCi/l	by GEA****	pCi/g
Radium Alpha Emitters				
Ra-226 (via Rn-222 daughter)	2	pCi/l	by GEA	pCi/g
Gross Radium	2	pCi/l	5	pCi/g
Specific Actinide Emitters				
Isotopic Thorium (Th-228, 230, 232)	2****	pCi/l	2****	pCi/g
Isotopic Uranium (U-234, 235, 238)	2****	pCi/l	2****	pCi/g
Total Uranium (Chemical Analysis)	0.2	ug/l	2	ug/g
Np-237	2	pCi/l	2	pCi/g
Pu-238	2	pCi/l	2	pCi/g
Pu-239/240 (sum)	2		2	
Pu-241	20		20	
Am-241	2	pCi/l	2	pCi/g
Cm-244	2	pCi/l	2	pCi/g

*Applies only if the absence of alpha emitting radionuclides with lower LLDs is known.

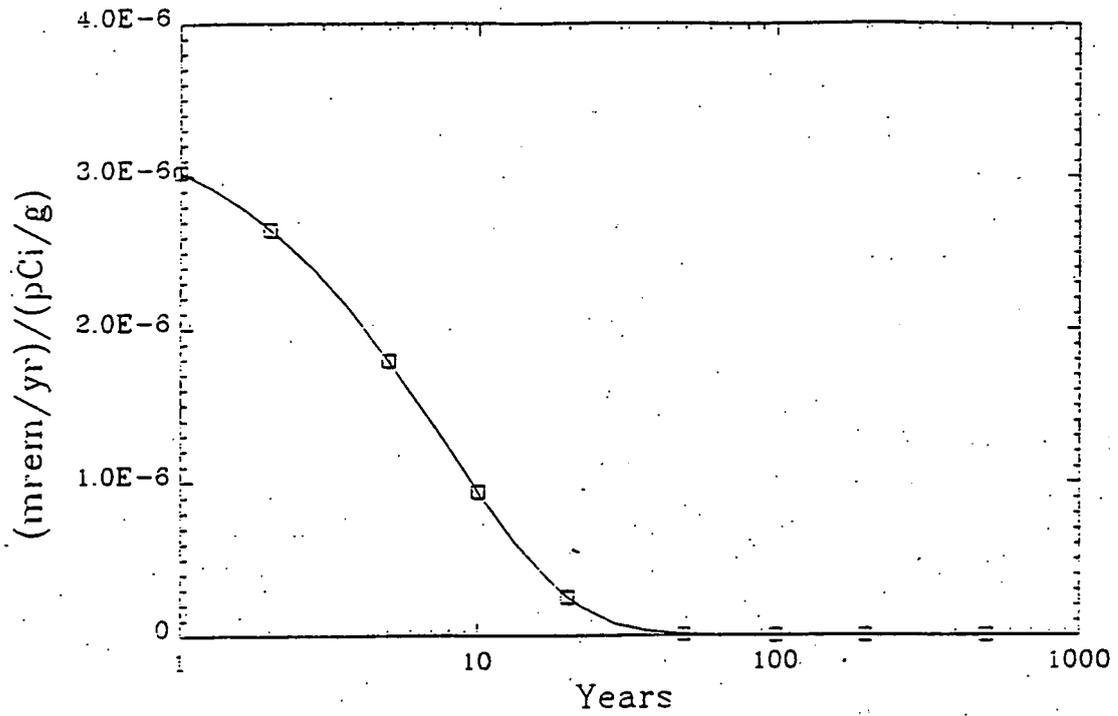
**Applies only if the absence of beta emitting radionuclides with lower LLDs is known.

***If fission products (e.g., Cs-137 and Sr-90) are below their LLD values the Se-79 will also be below detectable limits.

****Gamma Energy Analysis

*****Signifies 2 pCi/l (or 2 pCi/g respectively) for each isotope. It should be noted that the analysis will not differentiate between some isotopes (i.e., analysis will not differentiate between U-235 and U-236).

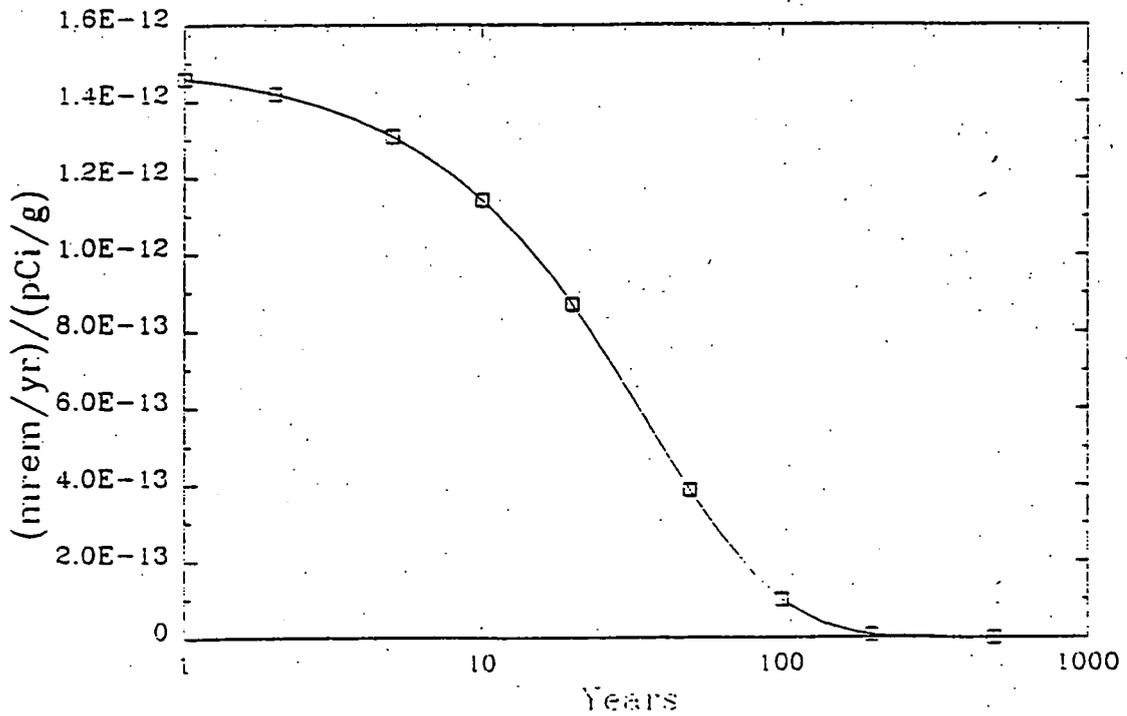
DOSE/SOURCE RATIO: All Pathways Summed, Co-60



SWL-3.DAT

10/09/95 13:04

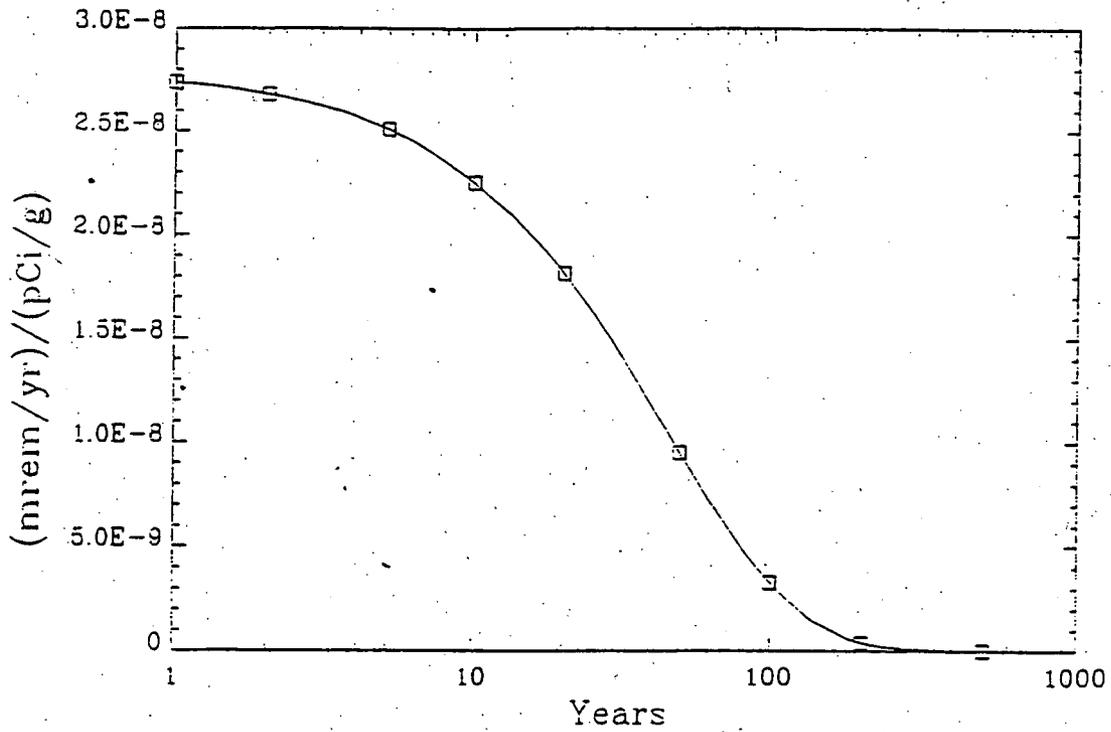
DOSE/SOURCE RATIO: All Pathways Summed, Sr-90



SWL-3.DAT

10/09/95 13:04

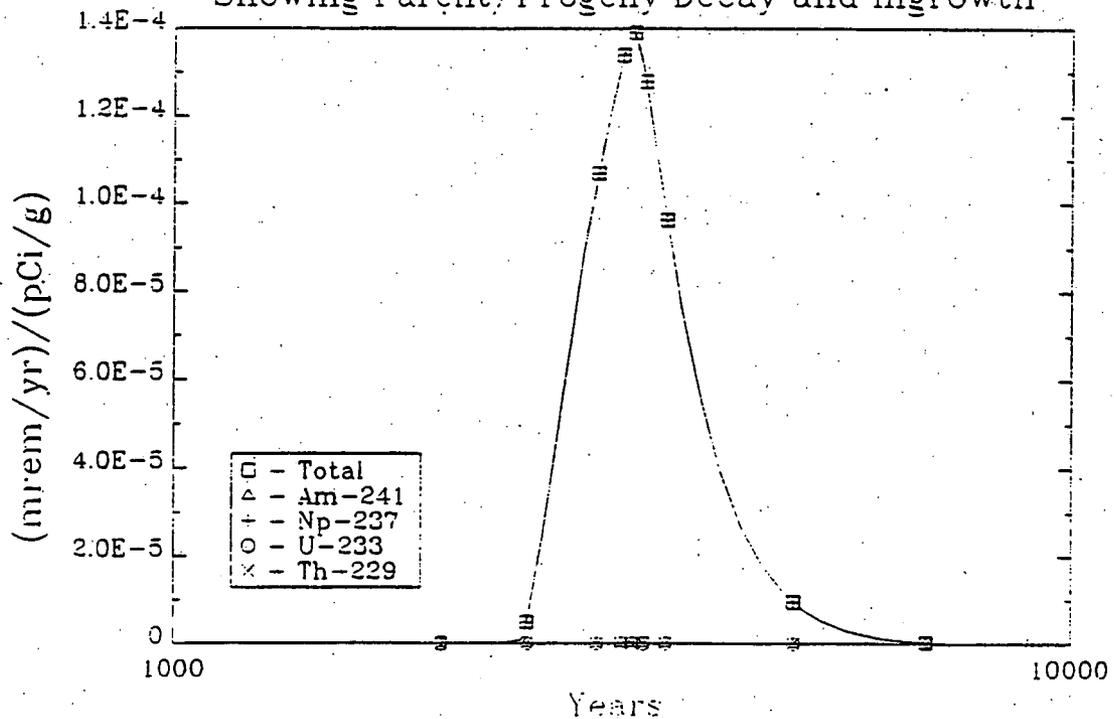
DOSE/SOURCE RATIO: All Pathways Summed, Cs-137



SWL-3.DAT

10/09/95 13:04

DOSE/SOURCE RATIO: All Pathways Summed, Am-241
Showing Parent/Progeny Decay and Ingrowth



SWL-3.DAT

10/09/95 13:24

RESRAD Doses at Times up to 500 Years

DOSE/SOURCE RATIO: All Pathways Summed, mrem/y per pCi/g
 SWL-2.DAT 10/09/95 13:05

Years	Co-60	Sr-90	Cs-137
0.000E+00	3.446E-06	1.498E-12	2.797E-08
1.000E+00	3.025E-06	1.458E-12	2.738E-08
1.296E+00	2.911E-06	1.447E-12	2.720E-08
1.678E+00	2.769E-06	1.432E-12	2.698E-08
2.000E+00	2.656E-06	1.419E-12	2.680E-08
2.175E+00	2.596E-06	1.413E-12	2.670E-08
2.817E+00	2.388E-06	1.388E-12	2.633E-08
3.650E+00	2.142E-06	1.357E-12	2.586E-08
4.729E+00	1.861E-06	1.318E-12	2.527E-08
5.000E+00	1.797E-06	1.308E-12	2.512E-08
6.126E+00	1.552E-06	1.269E-12	2.452E-08
7.937E+00	1.226E-06	1.208E-12	2.358E-08
1.000E+01	9.367E-07	1.142E-12	2.256E-08
1.028E+01	9.028E-07	1.134E-12	2.242E-08
1.332E+01	6.077E-07	1.044E-12	2.101E-08
1.726E+01	3.638E-07	9.382E-13	1.930E-08
2.000E+01	2.546E-07	8.710E-13	1.820E-08
2.236E+01	1.872E-07	8.169E-13	1.730E-08
2.897E+01	7.914E-08	6.829E-13	1.500E-08
3.753E+01	2.594E-08	5.413E-13	1.248E-08
4.862E+01	6.115E-09	4.007E-13	9.833E-09
5.000E+01	5.112E-09	3.860E-13	9.546E-09
6.300E+01	9.404E-10	2.713E-13	7.219E-09
8.162E+01	8.316E-11	1.637E-13	4.837E-09
1.000E+02	7.582E-12	9.943E-14	3.258E-09
1.057E+02	3.591E-12	8.510E-14	2.880E-09
1.370E+02	6.125E-14	3.645E-14	1.471E-09
1.775E+02	3.137E-16	1.216E-14	6.157E-10
2.000E+02	1.668E-17	6.598E-15	3.794E-10
2.299E+02	3.379E-19	2.929E-15	1.993E-10
2.979E+02	4.830E-23	4.636E-16	4.623E-11
3.859E+02	5.045E-28	4.255E-17	6.962E-12
5.000E+02	0.000E+00	1.928E-18	5.992E-13

RESRAD Doses for Am-241:

DOSE/SOURCE RATIO: All Pathways Summed, Am-241, mrem/y per pCi/g
 Showing Parent/Progeny Decay and Ingrowth

SWL-3.DAT 10/09/95 13:18

Years	Total	Am-241	Np-237	U-233	Th-229
2.000E+03	4.888E-08	4.662E-26	1.535E-15	4.837E-08	5.042E-10
2.107E+03	4.905E-08	6.457E-26	1.535E-15	4.853E-08	5.232E-10
2.220E+03	4.915E-08	9.101E-26	1.535E-15	4.861E-08	5.394E-10
2.339E+03	4.921E-08	1.307E-25	1.535E-15	4.866E-08	5.574E-10
2.464E+03	9.759E-07	1.913E-25	9.266E-07	4.865E-08	5.758E-10
2.500E+03	4.917E-06	2.131E-25	4.868E-06	4.864E-08	5.816E-10
2.596E+03	2.378E-05	2.857E-25	2.373E-05	4.847E-08	5.948E-10
2.736E+03	5.610E-05	4.361E-25	5.605E-05	4.608E-08	6.097E-10
2.882E+03	8.662E-05	6.809E-25	8.658E-05	4.258E-08	6.265E-10
3.000E+03	1.070E-04	9.743E-25	1.070E-04	3.890E-08	6.366E-10
3.037E+03	1.126E-04	1.089E-24	1.126E-04	3.748E-08	6.393E-10
3.199E+03	1.339E-04	1.786E-24	1.339E-04	3.085E-08	6.455E-10
3.200E+03	1.340E-04	1.789E-24	1.340E-04	3.082E-08	6.465E-10
3.300E+03	1.391E-04	2.425E-24	1.390E-04	2.627E-08	6.480E-10
3.371E+03	1.320E-04	3.007E-24	1.320E-04	2.316E-08	6.485E-10
3.400E+03	1.280E-04	3.286E-24	1.279E-04	2.194E-08	6.460E-10
3.551E+03	1.039E-04	5.206E-24	1.039E-04	1.648E-08	6.441E-10
3.600E+03	9.644E-05	6.036E-24	9.643E-05	1.501E-08	6.416E-10
3.742E+03	7.676E-05	9.285E-24	7.674E-05	1.142E-08	6.337E-10
3.942E+03	5.522E-05	1.708E-23	5.521E-05	7.737E-09	6.216E-10
4.153E+03	3.892E-05	3.246E-23	3.892E-05	5.134E-09	6.086E-10
4.376E+03	2.691E-05	6.384E-23	2.691E-05	3.335E-09	5.919E-10
4.610E+03	1.823E-05	1.302E-22	1.823E-05	2.117E-09	5.700E-10
4.857E+03	1.211E-05	2.759E-22	1.211E-05	1.313E-09	5.534E-10
5.000E+03	9.548E-06	4.255E-22	9.546E-06	9.965E-10	5.401E-10
5.118E+03	7.868E-06	6.087E-22	7.867E-06	7.938E-10	5.296E-10
5.392E+03	4.986E-06	1.401E-21	4.985E-06	4.677E-10	5.089E-10
5.681E+03	3.088E-06	3.372E-21	3.087E-06	2.681E-10	4.895E-10
5.985E+03	1.863E-06	8.505E-21	1.862E-06	1.494E-10	4.695E-10
6.306E+03	1.095E-06	2.255E-20	1.095E-06	8.078E-11	4.468E-10
6.644E+03	6.269E-07	6.297E-20	6.264E-07	4.234E-11	4.522E-10
7.000E+03	3.468E-07	1.858E-19	3.463E-07	2.148E-11	4.348E-10

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