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Hazards Assessment Document

ETR Facility Hazard Categorization

Idaho National Engineering and Environmental Laboratory

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REVISION NOTES

Revision 1 of this facility hazard categorization report for the Engineering Test Reactor addressed the comments made by DOE-ID during the request for approval for the change in facility categorization. DOE-ID review comments were transmitted in the R. V. Fustenau letter to D. M. Lucoff, Director, Reactor Programs, BBWI, "Comments of Hazard Categorization Documents for the Engineering Test Reactor and TRA Effluent Processing Facility," TPO-TRA-00-049, August 7, 2000. Approval signatures for this revision are on Document Action Request form, number 67000.

Revision 2 of this facility hazard categorization report for the Engineering Test Reactor removed the discussions of the Filling, Storage, and Remelt Facility and its hazards since the system was physically removed from the facility in January 2002.

ABSTRACT

This document evaluates the radiological and hazardous materials in the Engineering Test Reactor (ETR) and determines the facility characterization based on DOE requirements. The hazard category for the ETR facility was not specified in the 1982 version of the Safety Analysis Report prepared for the facility being placed in the Inactive Status. Afterwards a preliminary estimate based on the potential material present in the ETR, concluded that it should be a Category 3 facility. This preliminary estimate for ETR was transmitted to DOE in the Test Reactor Area Implementation Plan For Nuclear Facility Safety Analysis Report.

The radiological and hazardous material source terms were evaluated to determine the categorization of the ETR facility. The evaluation of the facility hazards results in a categorization of "radiological facility" based on the criteria in DOE-STD-1027 and DOE-EM-STD-5502. Radiological facilities are those facilities that do not meet or exceed the hazard Category 3 threshold quantity values published in DOE-STD-1027, but still contain quantities of radioactive material above those listed in Appendix B to Table 302.4 of 40 CFR 302. Also DOE-ID Notice, ID O 420.D, specifies requirements for a hazard categorization process for all DOE-ID facilities and non-facility operations which are not categorized by DOE-STD-1027 as Hazard Category 1, 2, or 3 nuclear facilities or activities. The ETR facility meets the low-hazard criteria listed in the notice.

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TABLES

1. INTRODUCTION

The radiological and hazardous material source terms were evaluated to determine the categorization of the Engineering Test Reactor (ETR) facility. This categorization is required by DOE Order 5480.23, Nuclear Safety Analysis Reports, December 1992 to determine if a need exists to develop a Safety Analysis Report (SAR) for the facility. DOE Order 5480.23 recommends that a graded approach be used that directs the effort proportional to the complexity of the facility and the safety systems relied on to maintain an acceptable level of risk. In applying the graded approach, this categorization used guidance and criteria provided in DOE requirements, DOE-STD-1027[1] and DOE-EM-STD-5502[2] to evaluate the hazards and classify the facility. Criteria from these DOE standards specify reviewing basic facility information and using estimates of material quantities to achieve an acceptable assessment, and an acceptable level of detail as allowed in the graded approach. DOE-STD-1027 provides the threshold values for each radionuclide used to classify facilities as Category 1, 2, or 3. Facilities with source inventories greater than the Category 3 threshold level are referred to as nuclear facilities. DOE-EM-STD-5502 provides guidelines for facilities that have inventories less than the Category 3 level but above the reportable quantity established in 40 Code of Federal Regulation (CFR) 302.[3] Radiological facilities are those facilities that do not meet or exceed the hazard Category 3 threshold quantity values published in DOE-STD-1027, but still contain quantities of radioactive material above those listed in Appendix B to Table 302.4 of 40 CFR 302. DOE-ID Notice, ID O 420.D[4] applies to non-nuclear facilities that are not classified as Category 1, 2, or 3 yet have hazardous material inventories exceeding Title 29 Code of Federal Regulations Part 1910.119, Process Safety Management[5] thresholds or the reportable quantities in Title 40 Code of Federal Regulations Part 355, Emergency Planning and Notification.[6]

The hazard category for the ETR facility was not specified in the 1982 Safety Analysis Report prepared for the facility being placed in the Inactive Status. Afterwards a preliminary estimate based on the potential inventory present in the ETR, concluded that it should be a Category 3 facility. This preliminary estimate for ETR was transmitted to DOE in the Test Reactor Area implementation plan for nuclear facility safety analysis reports.

The facility's enveloping radiological inventory is compared against the threshold quantities identified in requirement documents. The hazardous materials contained in the ETR facility have been evaluated and identified to determine their effect on the facility categorization.

The hazard categorization of the ETR is used to establish the criteria for the safety basis and the approval authority. Based on the radiological and hazardous material evaluation, the ETR is categorized as a low-hazard radiological facility.

2. FACILITY DESCRIPTION

The inactivation of ETR was initiated in December 1981, including defueling the reactor, draining all the liquid systems, and preparing all major equipment for long term storage. An inactive status SAR was written which provides the current controls for the facility and new SAR will be written when the inactive status is changed. The inactivated ETR facility consists of the reactor building and a number of attached or nearby supporting buildings or structures. The hazard evaluation consisted of reviewing the facilities that comprise the ETR facility complex listed in the ETR SAR. These buildings are noted below.

NUMBER	DESCRIPTION
TRA-642➢	Reactor Building
TRA-643➢	Compressor Building
TRA-644➢	Heat Exchanger Building
TRA-645☼	Secondary Pump House
TRA-647*	Administrative Building
TRA-648➢*	Electrical Building
TRA-655	Air Intake Building
TRA-663*	Standby Power Building
TRA-704	Underground Exhaust Pipe, Storage Pits, and Tunnel
TRA-705	Underground Exhaust Pipe, Storage Pits, and Tunnel
TRA-706	Underground Exhaust Pipe, Storage Pits, and Tunnel
TRA-751☼	Cooling Tower
TRA-752*	Transformer Station
TRA-753*	Waste Gas Stack
TRA-755➢	Filter Pit Building

➢ These buildings are also listed as nuclear facilities in company procedure, MCP-2446 Controlling Lists of Nuclear Facilities and Nuclear Facility Managers, Revision 6, January 4, 2000.

☼ These buildings have since been dismantled.

* These buildings are further declassified in a TRA hazard assessment report, EDF-TRA-1554, Facility Hazard Evaluation and Facility Categorization.[14]

The reactor building houses the defueled ETR reactor. The reactor vessel is a multi-diameter pressure vessel approximately 35 feet in height. The beryllium and aluminum reflector pieces are supported and located on the grid plate. The vessel is constructed of stainless clad carbon steel or stainless steel and the wall thickness varies from 1 to 3 inches. The top is covered with an ellipsoidal head that is bolted to the vessel. The vessel bottom head is a flat circular stainless steel plate 8.5 inches thick. The reactor vessel is fully enclosed and supported by a high density concrete biological shield. At the reactor mid-plane, the shield's thickness is 8 feet. This thickness is uniform around the reactor and extends from the first floor to the basement ceiling. The top head shielding consists of one to three foot thick high density concrete. The 25 foot outside diameter of the biological shielding is covered with a ¾ inch steel plate.

Ventilation air flows for the ETR are now provided only by the Cubicle Exhaust System. Deenergizing the motor control, closing dampers, and disconnecting lines deactivated all the other ventilation systems. The Experiment Effluent system includes the primary and secondary filter pits, and the delay tanks. The primary and secondary filter pits are underground and each houses two filters. The filters are surrounded with four inches of lead and encased with carbon steel and high density concrete. The delay tanks are eight feet underground in a concrete pit. The ETR Filter Pit filter building has the filter pit housings underground containing the three loop filters. These three filters that enclosed in steel canisters surrounded by high density concrete.

The liquid effluent system consists of three liquid waste tanks for hot, warm, and cold liquid waste tanks, are located in the waste pits beneath hatches in the north floor of the reactor building. All the liquid wastes had been removed from the liquid waste storage tanks before placing the facility in the inactive status. Of the three only the warm waste tank is still in service.

The 5,000-gal Hot Waste Tanker is a horizontal axis cylindrical shaped steel tank. It is normally stored empty in the reactor building. Its purpose is to transport waste for processing from TRA to another site area.

The CO_2 fire protection system in the electrical equipment building in TRA-648 has been removed from service.[7] The remaining fire protection systems used for the remaining buildings associated with the ETR facility are the standard water systems that were not identified as new safety hazards.

3. HAZARD EVALUATION

The hazard evaluation is based on the unmitigated release of the radiological and hazardous chemical material. The unmitigated release of hazardous material does not take credit for any active safety features, which could prevent or mitigate the release of the material. Evaluation of the material at risk (MAR) for release can consider the location, or segregation, and dispersiblity of the material, the interaction with available energy sources, and/or the possible initiating events, or release mechanisms. The MAR for the ETR is based on an estimate of an enveloping inventory of the radiological and hazardous chemical material that would be present in the facility. Following the guidance in ID O 420.D, the threshold quantities of DOE-STD-1027 were modified using more appropriate airborne release fractions (ARF) for the particular release mechanisms for the location of the radiological inventories in the facility. When appropriate, dispersiblity is accounted for by following the guidance in ID O 420.D.

To determine the potential of radiological hazardous material in the facility, a review was made of the ETR Safety Analysis Report in the Inactive Status[8], the Characterization of the Engineering Test Reactor Facility[9], and Characterization and Decision Analysis for Engineering Test Reactor Facilities.[10] The ETR characterization report provided radiological data and surveys used to develop an enveloping facility inventory.

To determine the potential of non-radiological hazardous material in the facility, a review was made of the ETR Safety Analysis Report in the Inactive Status[8], the Characterization of the Engineering Test Reactor Facility[9], Characterization and Decision Analysis for Engineering Test Reactor Facilities[10], and the Characterization Hazards Assessments for Facilities Located at the Test Reactor Area.[11] A facility inspection and walk down and a review of the chemical inventory of the INEEL Chemical Management System (ICMS), supplemented this document review. A sampling program to determine the hazardous conditions associated with the ETR facilities was performed in support of the Decontamination and Decommissioning (D&D) decision making process and documented in the Characterization and Decision Analysis for Engineering Test Reactor Facilities report. The facilities included in the sampling program were the following areas: TRA-642 – Engineering Test Reactor, TRA-643 – ETR Compressor Building, TRA-644 – Heat Exchanger Building, TRA-647 – ETR Office Building, TRA-648 – ETR Electrical Building, TRA-663 – ETR Superior Diesel Building, TRA-645 – Secondary Coolant Pumphouse (removed), TRA-611 – Plug Storage Building.

The evaluation consisted of identifying the radiological and hazardous material inventories and then comparing the results against the guidelines established for categorizing the facilities.

3.1 Radiological Source Term

The ETR characterization report[9] provided the majority of the radiological data and surveys used to develop an enveloping facility inventory. Radiological areas that had values for general radiation fields below 0.1 rem/hr and counts on smears below 200 dpm were considered radiologically clean. Some buildings and structures had been evaluated and did not contain significant hazardous materials to warrant further evaluation. The Air Intake building, Administration building, Electrical building, the Transformer building, and the ETR Standby Power building had values for general radiation fields and counts on smears at background levels. Because of these low levels, these buildings are not discussed in the radiological evaluation. A fire would be one feasible accident that could have the potential to release the contamination in the ventilation systems. The areas serviced by the only operating ventilation system such as the Heat Exchange Building, the Waste Gas Stack, the Cubicle Exhaust Room, and reactor vessel are largely constructed of non-combustible material and contain equipment, piping or tanks that would neither initiate nor sustain a fire. It is assumed that the interior of these systems is contaminated, based on the representative internal smear results in the characterization report. However, general background radiation readings of ductwork in most cases were low. Some were less than 0.1 mrem/hour, which is considered non-contaminated by the characterization report.

The ETR Contamination smear activity identified in the characterization report[9] has been divided into three areas: the activity in the ETR reactor building, the activity associated with the rest of the ETR support facilities, and the activity associated with the empty Hot Waste Tanker. The smear activity for each area has been averaged. The assumptions used to derive the amount of contamination activity are in Appendix A. The ETR reactor building has been categorized into the following areas to evaluate the smear activity: Reactor Nozzle Trench, Canal, Main Reactor Floor Area, Reactor Console Floor and Balcony, Reactor Building Basement Header, Warm and Hot Waste Pits, Annulus Gas, M-7 and P-7 Cubicle, J-10/L-10 Cubicle, C-7/M-13/N-14 Cubicle, F-10/H-10 Cubicle, L-12/M-7 Cubicle, C-13/G-16 Cubicle, Helium System Cubicle, Control Access Room, and Subpile Room. The results for the averaging of the smear activity in each area are in Table A1.B of Appendix A.

The second area for the "ETR support facilities" consisted of the following: General Electric Experimental Loop (GEEL) Tunnel, Compressor Building Primary Pump Tunnel, Compressor Building Storage Areas, Heat Exchanger Building Demineralizer Bypass Valve Room, Heat Exchanger Building Demineralizer Tank, Heat Exchanger Building Main Floor, Heat Exchanger Building Basement, Heat Exchanger Building Degas Tank, Primary Pump Pits, and Emergency Shutdown and Degas Pump Pits. The results for the averaging of the smear activity in each area are in Table A1.C of Appendix A.

The third area is the empty Hot Waste Tanker stored on the reactor main floor. Measurements of the radiation fields associated with this tanker have indicated that a hot spot exists at the front of the tanker. Measurements of the radiation fields associated with the empty Hot Waste Tanker were low except the hot spot. This hot spot on the front bulkhead has been measured to read about 70 mrem/hr at 0.5 in. The activity for the empty Hot Waste Tanker is extrapolated from the gamma scan measurement and the results are in Table A1.D in Appendix A.

A later radiological survey[12] was completed for the cubicles in the reactor building as part of a project to characterize the low level waste of the facility. It is a limited survey and not comprehensive enough to replace the characterization report data that was used. However, a comparison of the data between the two reports indicates that the level of conservatism applied to the assumptions and calculations of this report is adequate.

The activity is calculated for each area described above in Appendix A to this report. Table 1 combines the results listed in the Appendix Tables A1.B through A1.D.

Table 1. ETR Contamination Source Term.

Activity Source	Curies	Radionuclides Identified
Nozzle trench	6.8E-07	Cs-137, Co-60
Canal	8.1E-04	Cs-137, Co-60, Sr-90, U-235, Pu-239, Pu-240, Am-241
Main floor	5.8E-05	Cs-137, Co-60
Reactor console floor and balcony	1.8E-04	Cs-137, Co-60, Sr-90
Reactor basement vent header	1.6E-05	Cs-137, Co-60
Warm and hot waste pits	5.0E-05	Cs-137, Co-60
Annulus Gas	2.4E-05	Cs-137, Co-60
M-3 and P-7 cubicles	6.3E-05	Cs-137, Co-60
J-10 and L-10 cubicles	1.1E-06	Cs-137, Co-60
C-7/M-13/N-14 cubicles	2.6E-04	Cs-137, Co-60
F-10/H-10 cubicles	1.5E-04	Cs-137, Co-60
L-12/M-7 cubicles	1.8E-03	Cs-137, Co-60

Table 1. (continued).

Activity Source	Curies	Radionuclides Identified
C-13/G-16 cubicles	1.1E-05	Cs-137, Co-60
Helium system cubicle	3.9E-05	Cs-137, Co-60
Control access room	1.6E-06	Cs-137, Co-60
Subpile room	3.6E-06	Cs-137, Co-60, Sr-90
GEEL tunnel	6.3E-06	Cs-137, Co-60
Comp Bldg PCS pipe tunnel	3.5E-05	Cs-137, Co-60
Comp Bldg Storage area	7.2E-06	Cs-137, Co-60
HX Bldg bypass valve	6.8E-07	Cs-137, Co-60
HX Bldg tank room	3.8E-06	Cs-137, Co-60
HX Bldg main room	2.0E-05	Cs-137, Co-60
HX Bldg basement	7.6E-05	Cs-137, Co-60
HX Bldg Degassing Tank	2.9E-06	Cs-137, Co-60
HX Bldg primary pump pits	9.4E-06	Cs-137, Co-60, Hf-181
HX Bldg emergency shutdown pump pits	7.6E-07	Cs-137, Co-60, Hf-181
HX Bldg degasifier pump pits	6.3E-07	Cs-137, Co-60, Hf-181
Empty Hot Waste Tanker hot spot	6.9E-05	Co-60, Cs-137, Eu-152, Eu-154
Total	3.7E-03	-

Table 2 compares the contamination source term to the guidelines of DOE-STD-1027. For conservatism, each radionuclide identified is assumed to have the total curie content of the entire releasable contamination activity from Table 1. The reason for this assumption is that the distribution of radionuclides is not provided in the reference material describing the existing contamination at ETR. Since the activity levels are low, this assumption will allow the radionuclides present to be evaluated without making isotopic distribution assumptions that would be difficult to justify. The resulting facility categorization should be conservative, but representative of the nature of the facility. The methodology prescribed in DOE-STD-1027 categorizes facilities having combinations of radioactive materials by calculating the ratio of the quantity of each material to the Category 2 or 3 thresholds and determining if the sum exceeds one (e.g., [inventory of radionuclide A/threshold of radionuclide A] + [inventory of radionuclide

B/threshold of radionuclide B] + [inventory of radionuclide n/threshold of radionuclide n] >1).

Table 2. Comparison of the ETR Contamination Source Term to the DOE-STD-1027 Guidelines.

Radionuclide	Curies	DOE-STD-1027 Guideline values for Category 3	Ratio of curies/guideline value
Am-241	3.7E-03	5.2E-1	7.10E-03
Co-60	3.7E-03	2.8E+2	1.30E-05
Cs-137	3.7E-03	6.0E+1	6.00E-05
Pu-239	3.7E-03	5.2E-1	7.10E-03
Pu-240	3.7E-03	5.2E-1	7.10E-03
Sr-90	3.7E-03	1.6E+1	2.30E-04
U-235	3.7E-03	4.2E+0	8.80E-04
Hf-181	3.7E-03	7.6E+2	4.90E-06
Eu-152	3.7E-03	2.0E+2	1.85E-05
Eu-154	3.7E-03	2.0E+2	1.85E-05
Total	-	-	2.26E-02

Comparing the contamination source term values to the DOE-STD-1027 guidelines indicates that the sum total of the ratios is at least 40 times less than the minimum value that would designate a Category 3 classification. The total surface contamination area value can be doubled to account the potential activity on the ceiling that was not calculated because of lack of data (see assumptions in appendix A). This would reduce the total to a difference between the sum of the ratios and the threshold value to a factor 20 less. Thus, based the surface contamination activity and the empty tanker contribution, the source term would result in the facility being classified as a less than Category 3.

The airborne release fractions (ARFs) used in generating the threshold values for DOE-STD-1027 are intended to be generally conservative for a broad range of possible situations. It is stated in the standard that other ARFs may be used provided there is a reasonably conservative analysis for using these defensible realistic values based on the characteristics of the release. DOE-STD-1027 and ID O 420.D allow modifying the threshold quantity values (TQVs) by choosing an airborne release fraction that is more representative of the dispersiblity of the material. The sum of the ratios for the reactor vessel, the liquid waste tanks, and the filters was calculated in Appendix A using modified ARFs.

The reactor vessel was defueled and drained but some irradiated components remain stored within the vessel. The radionuclide cobalt 60 in the stainless steel grid plate and four inpile tubes is the dominant source. The characterization report estimated that a total of 30,000 curies remained in the enclosed reactor vessel with the major radionuclides identified as cobalt and cesium. Tritium is a decay product from the neutron activation of the beryllium reflector. Tritium as the activation product of the beryllium reflector has been conservatively calculated and added to the estimated radiological inventory in the reactor vessel. The enveloping reactor vessel inventory was used to calculate the sum of the ratios and results, reported in Table 3 below, indicate that the reactor vessel inventory is below the allowable guidelines.

All the filters are underground and isolated in pits and typically encapsulated layers of lead, stainless steel and high concrete. Radiological information from the characterization report was limited but an inventory was estimated. The Experiment Effluent system includes the primary and secondary filter pits, and the delay tanks. Since the primary filters are the first removal mechanism in the ventilation system, their inventory would be significantly higher then downstream components. Therefore, their inventory was conservatively applied to the downstream secondary filters. The delay tanks are empty and the radiation readings from the characterization report were all less than 0.1 mR/hr. The delays tanks are considered radiologically clean. The characterization report provided radiation readings for one of the loop filters and the data was applied to the other two loop filters. A radiological engineer used the data from the characterization report to develop a conservative enveloping inventory for the seven filters in the systems. The inventory is estimated to be a total of 615 curies remaining in the enclosed filters with the major radionuclides identified as cobalt and cesium. The enveloping filter inventory was used to calculate the sum of the ratios and the results are presented in Appendix A. The results, reported in Table 3 below, indicate that the filter inventory is below the allowable guidelines.

The liquid waste storage tanks have been sampled to determine the radioactive contamination.[10] The results for the cold and the warm waste samples indicate that both tanks have activity concentrations in the nanocurie range. The total activity would be less than one millicurie for either of the identified radioactive contaminants, Cs-137 and Co-60. The hot waste tank has not been sampled because there is no sampling provision provided on the tank, but it had been completely drained and now only contains overflow from the warm waste tank. Since the hot waste tank only contains the overflow from the warm waste tank, it should not contain levels of radioactivity different from those in the warm waste tank. An enveloping radiological inventory was calculated in Appendix A, and used to calculate the sum of the ratios. The result, reported in Table 3 below, indicates that the liquid waste tank inventory is below the allowable guidelines.

When these sum of ratios is added to the total from Table 2 and compared to the threshold the result is still below the guideline limit of one. Thus, the inventory calculated for the reactor vessel, filters, and liquid waste tanks when added to the contamination source term does not change the facility hazard characterization of this report.

Table 3. Comparison of the ETR Total Inventory to the DOE-STD-1027 Guidelines

	Sum of Ratios
Table 2 Total doubled	4.52E-02
Reactor Vessel	1.14E-02
Liquid Waste Tanks	4.99E-09
Filters	5.81E-04
Total	5.72E-02

The reportable quantities (RQ)from 40 CFR 302 Appendix B[3] are used to establish the dividing line between radiological and non-nuclear facilities. The same methodology as described in DOE-STD-1027 is used for this determination. In accordance with the facility classification methodology in DOE-EM-STD-5502, the radionuclides from Table 1 are compared to the threshold values in 40 CFR 302. Table 4 presents the results of the comparison.

Table 4. Comparison of Radionuclides with 40 CFR 302 Threshold Values

Radionuclide	**40 CFR 302[a] RQ Threshold (Ci)**	**Curies (Ci)**	**Ratio of Ci/Threshold**
Am-241	0.01	3.7E-03	3.7E-01
Co-60	10	3.7E-03	3.7E-04
Cs-137	1	3.7E-03	3.7E-03
Pu-239	0.01	3.7E-03	3.7E-01
Pu-240	0.01	3.7E-03	3.7E-01
Sr-90	0.1	3.7E-03	3.7E-02
U-235	0.1	3.7E-03	3.7E-02

Table 4. (continued).

Radionuclide	40 CFR 302[a] RQ Threshold (Ci)	Curies (Ci)	Ratio of Ci/Threshold
Hf-181	10	3.7E-03	3.7E-04
Eu-152	10	3.7E-03	3.7E-04
Eu-154	10	3.7E-03	3.7E-04
Total	-	-	1.19

a. The Reportable Quantity threshold values are taken from 40 Code of Federal Regulation 302 Appendix B to paragraph 302.4 which cites the final RQ for radionuclides.

The cumulative sum of the ratios of the facility contamination for all radionuclides exceeds the value of 1.0 by about 19 percent. Based on the conservative assumptions that each radionuclide has all of the activity calculated for surface contamination in a particular facility area, none of the individual radionuclides exceed the threshold values established in 40 CFR 302, Appendix B. The principal radionuclides responsible for exceeding the threshold value are attributed to the transuranics, however assumptions to reduce the transuranic radionuclides would be difficult to justify. Further the addition of the inventories from the reactor vessel, the filters, liquid waste tanks, and the filled waste tanker would continue to increase the amount by which the threshold is exceeded.

Based on comparing the sum of the ratios with the guideline thresholds, the facility would be classified a "radiological facility" since the inventory of radiological materials evaluated is below the levels as defined in DOE-STD-1027, but above the reportable quantity (RQ) values listed in Appendix B to Table 302.4 per 40 CFR 302.

The 5,000-gal Hot Waste Tanker is used to transport waste for processing from TRA to another site area. The approval and use of the Hot Waste Tanker for this purpose is delineated in the Hot Waste Tanker Transport Plan.[13] Although normally stored empty in the facility, an evaluation has been performed for the hypothetical case where the Hot Waste Tanker is filled with such waste and cannot be shipped, but must be stored indoors because of low outside temperature conditions. The source term assumed in the waste is the maximum isotopic concentrations permitted by the transportation plan. DOE-STD-1027 and ID O 420.D allow modifying the threshold quantity values (TQVs) by choosing an airborne release fraction that is more representative of the dispersiblity of the material. The new TQVs are calculated in Appendix A and a new ratio for each radionuclide is determined. The sum of the ratios is below the Category 3 threshold. If this ratio value of 2.31E-01 is added to the total from Table 3, the result is 2.66E-01. Even thought it is an order of magnitude larger, it is still below the guideline limit, and the hypothetical situation of the Hot Waste Tanker, filled according to the limits specified in the transport plan, does not change the facility hazard characterization.

3.2 Hazardous Material Source Term

A TRA hazard report[11] was completed to establish an emergency-planning zone for TRA. All radiological and non-radiological hazardous materials, stored, used or produced at the different facilities were identified and screened. The ETR facilities were evaluated for chemical and toxic hazards by screening against threshold values in 40 CFR Part 355[6] Appendix A and 40 CFR Part 302.[3] The ETR facilities did not contain material above the threshold screening criteria limits, which would cause a classifiable emergency. Review of the facility chemical inventories in the ICMS, did not identify any materials listed in either of the 29 CFR 1910.119[5] highly hazardous materials lists, nor were they listed on the appendix to part 355 of 40 CFR 355.

A sampling program to determine the hazardous conditions discussed in the Characterization and Decision Analysis for ETR Facilities[10] report documented that a total of 48 samples were collected and analyzed. The items analyzed included Toxicity Chemical Leaching Procedure (TCLP) metals, total metals, TCLP volatile organic compounds (VOC), and polychlorinated biphenyls (PCBs). Of the materials that were present, based on the sample results including those that exceeded RCRA limits, none were found on the 29 CFR 1910.119[5] highly hazardous materials list, nor were they listed on the appendix to part 355 of 40 CFR 355 except for chloroform. The liquid waste storage tanks were sampled for VOC and the amount of chloroform derived from the sample result for filled tanks is calculated in Appendix A. Chloroform is listed on the appendix to part 355, but the tank amount calculated is well below the threshold limit that would require the emergency planning and notification per 40 CFR 355. Identifying the hazardous materials and comparing the identified hazardous material to the guideline values of 29 CFR 1910.119 or 40 CFR 355 results in no identified hazardous material that exceeds low hazard thresholds.

4. CONCLUSIONS

Based on the unmitigated release of the radiological and hazardous chemical inventory of the ETR it is classified as a low hazard radiological facility for the purpose of establishing the required safety baseline documentation. Radiological facilities are those with an inventory of radiological materials below the levels as defined in DOE-STD-1027, but above the reportable quantity (RQ) value listed in Appendix B to Table 302.4 (per 40 CFR 302). The ETR facility fits into that category since its inventory is below the levels is DOE-STD-1027, but above the RQ value listed in Appendix B to Table 302.4.

Further per DOE-EM-STD-5502-94, radiological facilities with inventories of hazardous materials at or above 29 CFR 1910.119 thresholds or the levels specified in 40 CFR 355, require additional documentation to demonstrate compliance with the principles of the Process Safety Management and Emergency Planning and Notification. No ETR hazardous materials that were identified, such as the sodium system, and compared to the guideline values of 29 CFR 1910.119 or 40 CFR 355, resulted in material that exceeded thresholds.

Also DOE-ID Notice, ID O 420.D, specifies requirements for a hazard categorization process for all DOE-ID facilities and non-facility operations which are not categorized by DOE-STD-1027 as Hazard Category 1, 2, or 3 nuclear facilities or activities. Facilities are to be classified as low, moderate, or high, to determine the level of review and approval required for the safety documentation. The ETR facility meets the low-hazard criteria listed in the notice since there is no sealed radioactive sources nor radiation-producing devices in the facility. In addition, potential radiation exposure from non-releasable radioactive material will not be in excess of 2 rem from a single event because implementation of the institutional radiation protection program. There are no existing systems that would have the potential to injure more than 5 on site people or to increase risk to off site public. Finally, there are no MAR quantities of hazardous material that meet or exceed guideline values of 29 CFR 1910.119 or 40 CFR 355.

Based on this evaluation, the ETR facility is categorized per DOE-EM-STD-5502-94 as a radiological facility and thus will need an auditable safety analysis. Since there were no hazardous materials that exceeded the CFR thresholds no additional compliance documents, are indicated per DOE-EM-STD-5502-94. Review and approval of the auditable safety analysis documentation will be per DOE-ID Notice, ID O 420.D, as required for a low hazard facility.

A TRA hazard assessment[14] has been completed as part of the implementation of the Integrated Safety Management System (ISMS). The ETR areas that have had their status downgraded to Not Requiring Additional Safety Analysis (NRASA) since the first issue of this report are listed below. The information used for this further declassification of those particular buildings would not conflict with the information in this report.

TRA-647	Administrative Building
TRA-648	Electrical Building
TRA-655	Air Intake Building
TRA-663	Standby Power Building
TRA-752	Transformer Building
TRA-753	Waste Gas Stack

5. REFERENCES

1. Department of Energy, DOE Standard, Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports, DOE-STD-1027-92, September 1997

2. Department of Energy, DOE Standard, Hazard Baseline Documentation, DOE-EM-STD-5502-94, August 1994

3. Title 40 Code of Federal Regulations Part 302, Designation, Reportable Quantities, and Notification, Appendix B to Paragraph 302.4, Radionuclides, July 1999

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6. APPENDICES

Appendix A, Calculations

APPENDIX A

Calculations

A1. SMEAR ACTIVITY CALCULATION

ASSUMPTIONS

The smear activity in the ETR facilities has been used to determine the amount of radioactivity that could be released during a postulated accident in ETR. The smear activities identified in the Characterization Report[A1] have been averaged for all the smear samples taken in each area. The assumptions used during the evaluation, the grouping of the areas to be considered, the average activity present in each area, and the estimate of the total activity present in each area are presented in this Appendix.

The assumptions used to derive the amount of activity that is included in the contamination is provided below:

- The source term used in the previous safety analysis was developed 16 years ago. The decay of the short lived activation/fission products has changed the composition of the radionuclides remaining in the source term. For example, the Mn-54 has a 312-day half-life and would have undergone greater than 16 half lives since the original report. All activities calculated were less than nanocurie amounts after a 16-year time period. For this evaluation, any radionuclide with less than 3-year half-life is considered to have decayed away and is not included in the source term. Those radionuclides affected by this would include the following:

Table A-1.A. Decayed Radionuclides.

Radionuclide	Half-life (years)	Initial Activity	Activity after 16 years	Progeny
Mn-54	8.55E-01	3.62E-04	8.41E-10	Cr-54 (stable)
Cs-134	5.64E-03	2.32E-03	0.00E+00	Ba-134 (stable)
Nb-95	9.56E-02	1.80E-04	7.82E-55	Mo-95 (stable)

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Table A-1.A. (continued).

Radionuclide	Half-life (years)	Initial Activity	Activity after 16 years	Progeny
Zr-95	1.70E-01	1.80E-04	8.06E-33	NB-95 (62 days)
Ag-110m	6.84E-01	4.38E-02	4.03E-09	Cd-110 (stable)
Sb-125	2.74	3.60E-06	6.29E-08	Te-125 (57 days)

- The smear activities are taken from the information provided in the Characterization of the Engineering Test Reactor Facility Report.[A1] There has been no reduction assumed for the activity levels cited in the document for the 16 years of radioactive decay of long lived radionuclides, with half lives greater than 3 years.

- The smear samples are obtained from a 100 cm^2 sample of the area. Some of the samples resulted in maximum values for the activity present. The other samples for the area, however, had a wide range of activity levels for the contamination levels. The assumption is made that the smear samples are taken uniformly over the area of the facility and represents the distribution of activity in the area proportional to the number of smears taken. Thus, an average of the smear activity from all of the samples in the area would represent the activity in the entire area. Samples taken inside highly contaminated piping and other openings into contaminated areas are included in the determination of the average for the activity, since it is not known if these areas have been subsequently closed to prevent the release of the activity inside the piping/areas.

- Because the distribution of the isotopic activity of the samples is not known, it is conservatively assumed that the total activity that is calculated on the smear sample is the activity for each of the major radionuclides identified in the sample. Even though this overestimates the activity present, it provides a bounding estimate for determining the facility classification. Since the total activity levels are low, this assumption should not cause the classification of the facility to be classed in a significantly higher category than is warranted by the nature of the facility.

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- The surface area to be used in calculating the source term, resulting from contamination, assumes that the activity is uniform over the floor and wall surface where applicable. Conservative estimates have been used in determining the floor area. The walls surrounding the highly contaminated areas are assumed to be eight feet high unless otherwise noted in the detailed calculations. The ceiling surface was not included in the calculations as information about the activity on the ceilings is limited. The values calculated for the floor and walls was doubled to account conservatively for the remaining walls and the ceilings.

- It is assumed that the surface contamination is available for release during postulated accident conditions. One of the postulated accidents that would release the surface contamination would be a fire.

- Conservative estimates have been made for the contaminated surface areas. Dimensioned detailed drawings were not always available so that some of the dimensions were estimated by scaling from other information presented.

SMEAR ACTIVITY AVERAGES

The smear activity has been divided into areas to separate the activity in the ETR reactor building from the activity associated with the rest of the ETR support facilities. The results for the averaging of the smear activity in each area are presented in the Table A1.B. The ETR reactor building has been categorized into the following areas to evaluate the smear activity.

Table A-1.B. Smear Activity Average for ETR Building 642.

ETR Facility Area	Average dpm/ cm^2
Reactor nozzle trench	236
Canal	35,209
Main reactor floor area	742
Reactor console floor and balcony	1,513
Reactor building basement header	2,113
Warm and hot waste pits	4,183
Annulus gas	13,500
M-3 and P-7 cubicle	18,620
J-10/L-10 cubicle	313

APPENDIX A

Table A-1.B. (continued)

ETR Facility Area	Average dpm/ cm^2
C-7/M-13/N-14 cubicle	61,755
F-10/H-10 cubicle	28,586
L-12/M-7 cubicle	367,819
C-13/G-16 cubicle	3,963
Helium system cubicle	7,064
Control access room	938
Subpile room	1,543

The second area was the "Support ETR facilities" and results for the averaging of the smear activity in each section are presented in Table A1.C.

Table A-1.C. Smear Activity Average for Support ETR Facilities

Support ETR Facilities	Average dpm/ cm^2
GEEL Tunnel	282
Compressor Building PCS pipe tunnel	1,612
Compressor Building Storage area	1,354
Heat Exchanger Building Demineralizer Bypass Valve Room	388
Heat Exchanger Building Demineralizer Tank	2,574
Heat Exchanger Building Main Floor	2,013
Heat Exchanger Building Basement	3,291
Heat Exchanger Building Degas Tank	2,750
Primary Pump Pits	1,528
Emergency Shutdown Pump Pits	743
Degas Pump Pits	602

APPENDIX A

ACTIVITY OF HOT SPOT ON THE EMPTY TANKER

The Hot Waste Tanker hot spot at 0.5 in. surveyed at 0.07 rem/h. This radiation reading and the percent distribution from the gamma scan analysis[A2] were then converted to curies using the computer software program, Microshield V5.03. The results are listed in Table A1.D below.

Table A-1.D Contamination Source Term Empty Tanker

Radionuclide	Percent Distribution [A2]	Curies
Cesium 137	71	4.88E-05
Cobalt 60	20	1.37E-05
Europium 152	0.6	3.09E-06
Europium 154	0.3	3.09E-06
Total	--	6.87E-05

The source term associated with the empty Hot Waste Tanker was added to the activities calculated for the surface contamination (See Table 1 in the body of the report). Its total activity is several orders of magnitude less than the contributions from the other source term in the facility.

ACTIVITY CALCULATIONS BY AREA

Smear sample is for 100 cm^2

Average dpm = dpm/100 $cm^2 \times (2.54\ cm/in)^2\ (12\ in/ft)^2$ = reading/100 $cm^2 \times 929\ cm^2/\ ft^2$

Curies = 1.56 E6 dpm × dps/60 dpm × Ci/3.7 E10 dps

Reactor Nozzle Trench

Inside diameter = 11.5 ft
Outside diameter = 17.5 ft
Height = 6 ft
Floor area: $\pi/4\ (d^2{}_o - d^2{}_I) = 136.6\ ft^2$
Wall area: $h \times \pi\ (d_o + d_I) = 547\ ft^2$
Total area is 684 ft^2
Average dpm = 236 dpm/100 $cm^2 \times 929\ cm^2/\ ft^2 \times 684\ ft^2$ = 1.5 E6 dpm

= 1.5 E6 dpm × 4.5 E-13 Ci/dpm = 6.8 E-7 Ci

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Canal

Shape is "T" configuration
Width = 8 ft
Length = 35 ft
Width = 12 ft
Length =60 ft
Height = 21 ft
Floor area is (8 ×35) +(12 × 60) = 1000 ft^2
Wall area is 21 × (8+35+35+60+12+60-8+12) = 4494 ft^2
Total area is 5494 ft^2

Average dpm = 35209 dpm/100 cm2 × 929 cm2/ ft^2 × 5494 ft^2= 1.8 E9 dpm

Curies = 1.8 E9 dpm × 4.5 E-13 Ci/dpm = 8.1 E-4 Ci

Main Reactor Floor Area

The area of this floor is a rectangle of 112 ft × 136 ft minus the canal and reactor footprint

The canal footprint is noted in the canal evaluation. The footprint of the reactor is assumed to be 30 ft in diameter.

Floor area is (112 × 136) – 1000 - π/4 × 302 = 13525 ft^2
Wall area is 8 × [2(112 + 136) + π 30] = 4722 ft^2
Total area = 18247 ft^2

Average dpm = 742 dpm/100 cm^2 × 929 cm^2/ ft^2 × 18247 ft^2 = 1.3 E8 dpm

Curies = 1.3 E8 dpm × 4.5 E-13 Ci/dpm = 5.8 E-5 Ci

Reactor Console Floor and Balcony

Console floor and wall area is the same as the Main Reactor Floor = 18247 ft^2
Balcony is a rectangle with dimensions of 136 ft by 50 ft = 6800 ft^2
Balcony wall area is 8 ft high by 2(136 +50) ft = 2960 ft^2
Total area is 28007 ft^2

Average dpm = 1513 dpm/100 cm^2 × 929 cm^2/ ft^2 × 28007 ft^2= 3.9 E8 dpm

Curies = 3.9 E8 dpm x 4.5 E-13 Ci/dpm = 1.8 E-4 Ci

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Reactor Building Basement Header

This is a 10-in. pipe that goes around the basement. The lengths are runs assumed to be (80 + 90 + 80 + 90) ft. Both the inside and outside surfaces of the pipe are assumed to be contaminated with radioactivity.

The total area is = π (10/12) ft $\times$ 2 $\times$ 340 ft = 1780.2 ft^2

Average dpm = 2113 dpm/100 cm^2 $\times$ 929 cm^2/ ft^2 $\times$ 1780.2 ft^2= 3.5 E7 dpm

Curies = 3.5 E7 dpm $\times$ 4.5 E-13 Ci/dpm = 1.6 E-5 Ci

Warm and Hot Waste Pits

Two room with dimensions of 15 $\times$ 20 ft
Wall height is assumed to be 10 ft.
The surface area of the three tanks is based on the tanks having diameters of 4, 6, and 8 ft and lengths of 8, 8, and 18.5 ft.

The tank area is = 2 times the end area + the circumferential area.

Tank diameter (ft)	End area (ft^2)	Surface area (ft^2)	
4	'25.1	100.5	
6	56.5	150.8	
8	100.5	465.0	
Totals	182.1	716.3	898.4 ft^2 tank area

Room area is 2 $\times$ 15 $\times$ 20 = 600 ft^2
Wall area is 2 $\times$ 10 $\times$ 2(15 +20) = 1400 ft^2
Total area of tanks, floors, and walls is 2898.4 ft^2

Average dpm = 4183 dpm/100 cm2 x 929 cm2/ ft2 x 2898.4 ft2= 1.1 E8 dpm

Curies = 1.1 E8 dpm x 4.5 E-13 Ci/dpm = 5.0 E-5 Ci

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Annulus Gas

The Annulus Gas Room is assumed to be 10 by 10 ft. The walls are assumed to be 8 ft high.
The floor area is 100 ft^2. The wall area is $4 \times 10 \times 8 = 320$ ft^2
Total area of floors and walls is 420 ft^2

Average dpm = 13500 dpm/100 cm^2 × 929 cm2/ ft^2 × 420 ft^2= 5.3 E7 dpm

Curies = 5.3 E7 dpm × 4.5 E-13 Ci/dpm = 2.4 E-5 Ci

M-3 and P-7 Cubicle

The cubicle is 25 ft wide by 12 ft deep. The corners are triangles starting at about 6 ft along the walls on both sides. The walls are 8 ft high.
The area of the floor is 25 ft × 12 ft – 2(1/2 × 6 × 12.5) ft^2 = (300 – 75) ft^2 = 225 ft^2
The wall surface area is 8 ft × (25 + 6 + 17.5 +17.5 + 6) ft = 576 ft^2
Total area of floors and walls is 801 ft^2

Average dpm = 18620 dpm/100 cm^2 × 929 cm2/ ft^2 × 801 ft^2= 1.4 E8 dpm

Curies = 1.4 E8 dpm × 4.5 E-13 Ci/dpm = 6.3 E-5 Ci

J-10/L-10 Cubicle

The cubicle is conservatively assumed to be a 12-ft by 25-ft room to account for its double pie shaped side and end. The walls are assumed to be 8 ft high.
The floor area is 300 ft^2
The wall surface area is 8 ft × (25 + 6 + 35 + 6) ft = 576 ft^2
Total area of floors and walls is 876 ft^2

Average dpm = 313 dpm/100 cm2 × 929 cm^2/ ft^2 × 876 ft^2= 2.5 E6 dpm

Curies = 2.5 E6 dpm × 4.5 E-13 Ci/dpm = 1.1 E-6 Ci

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C-7/M-13/N-14 Cubicle

The cubicle is approximated by a 10-ft by 30-ft room with an 8-ft long wall separating the two portions of the room. One of the 10-ft walls is open
The floor area is 300 ft^2
The wall surface area is 8 ft × [2(30 + 8) + 10] ft = 688 ft^2
Total area of floors and walls is 988 ft^2

Average dpm = 61755 dpm/100 cm^2 × 929 cm^2/ ft^2 × 988 ft^2 = 5.7 E8 dpm

Curies = 5.7 E8 dpm × 4.5 E-13 Ci/dpm = 2.6 E-4 Ci

F-10/H-10 Cubicle

This cubicle is 10-ft by 30-ft footprint with a triangular shape missing near the reactor vessel. This triangular shape is 5 ft by 12 ft. An added rectangle of 10 ft by 11 ft deep is added on to one of the sides of the major area. In addition, there is a 6-ft wall extending from one side of the cubicle partially separating the major cubicle.
The floor area is 10 ft by 30 ft - 0.5 × 5 ft × 12 ft + 10 ft by 11 ft = 380 ft^2
The wall surface area is 8 ft × [30 + 5 + 16.8 + 11 + 10 + 11 + 2 x 6 +8 +10] ft = 910.4 ft^2
Total area of floors and walls is 1290.4 ft^2

Average dpm = 28586 dpm/100 cm^2 × 929 cm^2/ ft^2 × 1290.4 ft^2 = 3.4 E8 dpm

Curies = 3.4 E8 dpm × 4.5 E-13 Ci/dpm = 1.5 E-4 Ci

L-12/M-7 Cubicle

This cubicle is 20-ft by 20-ft footprint with a triangular shape missing. This triangular shape is 12 ft by 8 ft. An added rectangle of 12 ft by 8 ft deep is added on to one of the sides of the rectangle.
The floor area is 20 ft × 20 ft – (0.5 × 12 ft × 8 ft) + 12 ft × 8 ft = 448 ft^2
The wall surface area is 8 ft × [20 + 8 + 16.8 + 8 + 12 + 8 + 20] ft = 742.4 ft^2
Total area of floors and walls is 1190 ft^2

Average dpm = 367819 dpm/100 cm^2 × 929 cm^2/ ft^2 × 1190 ft^2 = 4.0 E9 dpm

Curies = 4.0 E9 dpm × 4.5 E-13 Ci/dpm = 1.8 E-3 Ci

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C-13/G-16 Cubicle

This cubicle is 15-ft by 15-ft footprint with a triangular shape missing and an added triangle added to one end. The missing triangular shape is 10 ft by 15 ft. An added end triangle is 5 ft by 5 ft high.
The floor area is 15 ft $\times$ 15 ft – (0.5 $\times$ 10 ft $\times$ 15 ft) + 0.5 $\times$ 5 ft $\times$5 ft = 162.5 ft^2
The wall surface area is 8 ft $\times$ [15 + 15 + 21 + 5 + 5] ft = 488 ft^2
Total area of floors and walls is 650.5 ft^2

Average dpm = 3963 dpm/100 cm^2 $\times$ 929 cm^2/ ft^2 $\times$ 650.5 ft^2 = 2.4 E7 dpm

Curies = 2.4 E7 dpm $\times$ 4.5 E-13 Ci/dpm = 1.1 E-5 Ci

Helium System Cubicle

This cubicle is 20 ft by 20 ft footprint with two triangular shapes missing. The missing triangular shapes are 13 ft by 15 ft and 5 ft by 5 ft. There are two added walls in the cubicle that form a room 5 ft by 10 ft.
The floor area is 20 ft $\times$ 20 ft – (0.5 $\times$ 13 ft $\times$ 15 ft)–(0.5 $\times$ 5 ft $\times$ 5 ft) = 290 ft^2
The wall surface area is 8 ft $\times$ [20 + 7 + 21 + 7 + 15 + 8 + 20 + 2(5 +10)] ft = 1024 ft^2
Total area of floors and walls is 1314 ft^2

Average dpm = 7064 dpm/100 cm^2 $\times$ 929 cm^2/ ft^2 $\times$ 1314 ft^2 = 8.6 E7 dpm

Curies = 8.6 E7 dpm $\times$ 4.5 E-13 Ci/dpm = 3.9 E-5 Ci

Control Access Room

This control access room is a circular structure with a diameter of about 12 feet. The walls are assumed to be 8 ft high.
The floor area is $\pi/4 \times (12\text{ ft})^2$ = 113.1 ft^2
The wall surface area is 8 ft $\times$ π $\times$ 12 ft = 301 ft^2
Total area of floors and walls is 414.1 ft^2

Average dpm = 938 dpm/100 cm^2 $\times$ 929 cm^2/ ft^2 $\times$ 414.1 ft^2 = 3.6 E6 dpm

Curies = 3.6 E6 dpm $\times$ 4.5 E-13 Ci/dpm = 1.6E-6 Ci

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Subpile Room

This subpile room is a circular structure with a diameter of about 15 feet. The walls are assumed to be 8 ft high.
The floor area is $\pi/4 \times (15\ \text{ft})^2 = 176.7\ \text{ft}^2$
The wall surface area is $8\ \text{ft} \times \pi \times 15\text{ft} = 377.0\ \text{ft}^2$
Total area of floors and walls is $553.7\ \text{ft}^2$

Average dpm = $1543\ \text{dpm}/100\ \text{cm}^2 \times 929\ \text{cm}^2/\ \text{ft}^2 \times 553.7\ \text{ft}^2$ = 8 E6 dpm

Curies = 8 E6 dpm × 4.5 E-13 Ci/dpm = 3.6 E-6 Ci

The areas in the ETR Support Facilities for calculating the total smear activity are the same areas identified in the Characterization Report grouping of the smear activity as described above.

GEEL Tunnel

The GEEL tunnel runs beneath the reactor building and ends at the delay tanks located north and east of the reactor building.
The tunnel floor area 8×30 ft beneath the reactor building is $240\ \text{ft}^2$
The tunnel floor area outside the building is $8 \times 200\ \text{ft} = 1600\ \text{ft}^2$
The walls would be 230 ft long and 8 ft high on both sides = $3680\ \text{ft}^2$
Total area of floors and walls is $5520\ \text{ft}^2$

Average dpm = $282\ \text{dpm}/100\ \text{cm}^2 \times 929\ \text{cm}^2/\ \text{ft}^2 \times 5520\ \text{ft}^2$ = 1.4 E7 dpm

Curies = 1.4 E7 dpm × 4.5 E-13 Ci/dpm = 6.3 E-6 Ci

The radionuclides detected in the smears for this area were Cs-137 and Co-60.

Compressor Building PCS pipe tunnel

The pipe tunnel provides the path for the primary coolant pipe to enter the heat exchangers and passes under the compressor building.
The tunnel floor area is two rectangles 21×36 ft and 30×72 ft. = $2916\ \text{ft}^2$
The walls are assumed to be 8 ft high × $\{(2 \times 21) + 36 + 20 + (2 \times 30) + 72 + 56\}$ ft = $2288\ \text{ft}^2$
Total area of floors and walls is $5220\ \text{ft}^2$

Average dpm = 1612 dpm/100 cm2 x 929 cm2/ ft2 x 5220 ft2 = 7.8 E7 dpm

Curies = 7.8 E7 dpm x 4.5 E-13 Ci/dpm = 3.5 E-5 Ci

The radionuclides detected in the smears for this area were Cs-137 and Co-60.

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Compressor Building Storage Area

The only contaminated storage area in the Compressor Building was the storage area.
The storage floor area is two rectangles 14.4 × 25.2 ft and 12 × 23 ft. = 639 ft^2
The contaminated walls are assumed only for the larger of the storage areas. The walls are assumed to be 8 ft high × {(2 × 25.2) + (2 × 14.4)} ft = 633.6 ft2
Total area of floors and walls is 1273 ft2

Average dpm = 1354 dpm/100 cm^2 × 929 cm^2/ ft^2 × 1273 ft^2 = 1.6 E7 dpm

Curies = 1.6 E7 dpm × 4.5 E-13 Ci/dpm = 7.2 E-6 Ci

The radionuclides detected in the smears for this area were Cs-137 and Co-60.

Heat Exchanger Building Demineralizer Bypass Valve Room

The Heat Exchanger Demineralizer Bypass Valve Room has a floor area of 7.2 ft × 13.2 ft.
The floor area is a rectangle of 7.2 ft × 13.2 ft = 95 ft^2
The walls are assumed to be 8 ft high × {(2 × 7.2) + (2 × 13.2)} ft = 326 ft^2
Total area of floors and walls is 421 ft^2

Average dpm = 388 dpm/100 cm^2 × 929 cm^2/ ft^2 × 421 ft^2 = 1.5 E6 dpm

Curies = 1.5 E6 dpm × 4.5 E-13 Ci/dpm = 6.8 E-7 Ci

The radionuclides detected in the smears for this area were Cs-137 and Co-60.

Heat Exchanger Building Demineralizer Tank Room

The Heat Exchanger Demineralizer Tank Room has a floor area of 12.6 ft × 5.4 ft.
The floor area is a rectangle of 12.6 ft × 5.4 ft = 68 ft^2
The walls are assumed to be 8 ft high × {(2 × 12.6) + (2 × 5.4} ft = 288 ft^2
Total area of floors and walls is 356 ft^2

Average dpm = 2574 dpm/100 cm^2 × 929 cm2/ ft^2 × 356 ft2 = 8.5 E6 dpm

Curies = 8.5 E6 dpm × 4.5 E-13 Ci/dpm = 3.8 E-6 Ci

The radionuclides detected in the smears for this area were Cs-137 and Co-60.

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Heat Exchanger Building Main Room

The floor area is a rectangle of 24 ft × 42.6 ft = 1022.4 ft^2
The walls are assumed to be 10 ft high × {(2 × 24) + (2 × 42.6} ft = 1332 ft^2
Total area of floors and walls is 2354.4 ft^2

Average dpm = 2013 dpm/100 cm^2 × 929 cm^2/ ft^2 × 2354.4 ft^2 = 4.4 E6 dpm

Curies = 4.4 E6 dpm × 4.5 E-13 Ci/dpm = 2.0 E-5 Ci

The radionuclides detected in the smears for this area were Cs-137 and Co-60.

Heat Exchanger Building Basement

The floor area is a rectangle of 24 ft × 42.6 ft = 1022.4 ft2
The walls are assumed to be 20 ft high × {(2 × 24) +3(2 × 15) + (2 × 42.6} ft = 4464 ft^2
Total area of floors and walls is 5486.4 ft^2

Average dpm = 3291 dpm/100 cm^2 × 929 cm^2/ ft^2 × 5464 ft^2 = 1.7 E8 dpm

Curies = 1.7 E8 dpm × 4.5 E-13 Ci/dpm = 7.6 E-5 Ci

The radionuclides detected in the smears for this area were Cs-137 and Co-60.

Heat Exchanger Building Degassing Tank

The floor area is a rectangle of 6 ft × 7.2 ft = 43.2 ft^2
The walls are assumed to be 8 high × {(2 × 6) + (2 × 7.2} ft = 211.2 ft^2
Total area of floors and walls is 254.4 ft^2

Average dpm = 2750 dpm/100 cm2 × 929 cm^2/ ft^2 × 254.4 ft^2 = 6.5 E6 dpm

Curies = 6.5 E6 dpm × 4.5 E-13 Ci/dpm = 2.9 E-6 Ci

The radionuclides detected in the smears for this area were Cs-137 and Co-60.

APPENDIX A

Heat Exchanger Building Secondary Pipe Trench

This has less than 200-dpm contamination – impact not calculated.
The radionuclides detected in the smears for this area were Cs-137 and Co-60.
Heat Exchanger Building Primary Pump Pits
The floor area is 4 pits each 9 ft × 9 ft = 324 ft^2
The walls are assumed to be 8 ft high × 4{(2 × 9) + (2 × 9} ft = 1152 ft^2
Total area of floors and walls is 1476ft^2

Average dpm = 1528 dpm/100 cm^2 × 929 cm^2/ ft^2 × 1476 ft^2 = 2.1 E7 dpm

Curies = 2.1 E7 dpm × 4.5 E-13 Ci/dpm = 9.4 E-6 Ci

The radionuclides detected in the smears for this area were Cs-137, Co-60, and Hf-181.

Heat Exchanger Building Emergency Shutdown Pump Pits

The floor area is a rectangle of 7.3 ft × 5.7 ft = 41.6 ft^2
The walls are assumed to be 8 ft high × {(2 × 7.3) + (2 × 5.7} ft = 208 ft^2
Total area of floors and walls is 250 ft^2

Average dpm = 743 dpm/100 cm^2 × 929 cm^2/ ft^2 × 250 ft^2 = 1.7 E6 dpm

Curies = 1.7 E6 dpm × 4.5 E-13 Ci/dpm = 7.6 E-7 Ci

The radionuclides detected in the smears for this area were Cs-137, Co-60, and Hf-181.

Heat Exchanger Building Degasifier Pumps

The floor area is a rectangle of 7 ft × 5.7 ft = 40 ft^2
The walls are assumed to be 8 ft high × {(2 × 7) + (2 × 5.7} ft = 203 ft^2
Total area of floors and walls is 243 ft^2

Average dpm = 602 dpm/100 cm^2 × 929 cm2/ ft^2 × 243 ft^2 = 1.4 E6 dpm

Curies = 1.4 E6 dpm × 4.5 E-13 Ci/dpm = 6.3 E-7 Ci

The radionuclides detected in the smears for this area were Cs-137, Co-60, and Hf-181.

APPENDIX A

A2. ACTIVITY OF FILLED TANKER

The airborne release fractions (ARFs) used in generating the threshold values for DOE-STD-1027 are intended to be generally conservative for a broad range of possible situations. It is stated in the standard that other ARFs may be used provided there is a reasonably conservative analysis for using these defensible realistic values based on the characteristics of the release. ARFs listed in the DOE HandbookA3are provided for various materials based on experimental data. For choosing an applicable ARF, the accident considered is an event that fails the transport Hot Waste Tanker, resulting in the waste dropping through air. The bounding ARF for a free-fall spill of aqueous solutions at a 3-meter fall distance was used. As shown in the equation below, following the guidance in ID O 420.D, the threshold quantities of DOE-STD-1027 were modified using more appropriate ARF for the particular release mechanisms for the location of the radiological inventories in the facility.

$TQV_{new} = TQV_{STD\text{-}1027} / (ARF/ARF_{STD\text{-}1027})$

Table A-2. Contamination Source Term Filled Tanker.

Transport Plan Limits Radionuclide	Concentration (microCi/mL)	MAR* (curies)	DOE-STD-1027 TQV (curies)	Handbook ARF	DOE-STD-1027 ARF	TQV_{new} (curies)	MAR/TQV_{new}
Ag-108	0.002	4.38E-02	260	2. E-04	0.01	1.30E+04	3.37E-06
Ag-110m	0.002	4.38E-02	260	2. E-04	0.01	1.30E+04	3.37E-06
Am-241	0.002	4.38E-02	0.52	2. E-04	0.001	2.60E-00	1.69E-02
Ba-140	0.16	3.51E+00	600	2. E-04	0.01	3.00E+04	1.17E-04
Ce-141	0.11	2.41E+00	1000	2. E-04	0.01	5.00E+04	4.82E-05
Ce-143	0.002	4.38E-02	3800	2. E-04	0.01	1.90E+07	2.31E-07
Ce-144	0.023	5.04E-01	100	2. E-04	0.01	5.00E+03	1.01E-04
Co-58	0.01	2.19E-01	900	2. E-04	0.001	4.50E+03	4.87E-05
Co-60	0.05	1.10E+00	280	2. E-04	0.001	1.40E+03	7.83E-04
Cr-51	2.4	5.26E+01	22000	2. E-04	0.01	1.10E+06	4.78E-05
Cs-134	0.2	4.38E+00	42	2. E-04	0.01	2.10E+03	2.09E-03
Cs-137	0.28	6.14E+00	60	2. E-04	0.01	3.00E+03	2.05E-03
Eu-152	0.044	9.64E-01	200	2. E-04	0.01	1.00E+04	9.64E-05

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Table A-2. (continued).

Transport Plan Limits Radionuclide	Concentration (microCi/mL)	MAR⁺ (curies)	DOE-STD-1027 TQV (curies)	Handbook ARF	DOE-STD-1027 ARF	TQV_{new} (curies)	MAR/TQV_{new}
Eu-154	0.019	4.16E-01	200	2. E-04	0.01	1.00E+04	4.16E-05
Eu-155	0.01	2.19E-01	940	2. E-04	0.01	4.70E+04	4.66E-06
Fe-59	0.002	4.38E-02	600	2. E-04	0.01	3.00E+04	1.46E-06
Hf-181	0.002	4.38E-02	760	2. E-04	0.01	3.80E+04	1.15E-06
I-131	0.063	1.38E+00	0.92	2. E-04	0.5	2.30E+03	6.00E-04
I-132	0.002	4.38E-02	1660	2. E-04	0.5	4.25E+06	1.06E-08
I-133	0.002	4.38E-02	19.4	2. E-04	0.5	4.75E+04	9.04E-07
Ir-192	0.002	4.38E-02	940	2. E-04	0.001	4.70E+03	9.33E-06
La-140	0.18	3.94E+00	400	2. E-04	0.01	2.00E+04	1.97E-04
Mn-54	0.0021	4.60E-02	880	2. E-04	0.01	4.40E+04	1.05E-06
Mo-99	0.002	4.38E-02	3400	2. E-04	0.01	1.70E+05	2.58E-07
Na-24	0.002	4.38E-02	300	2. E-04	0.01	1.50E+04	2.92E-06
Nb-95	0.012	2.63E-01	960	2. E-04	0.01	4.80E+04	5.48E-06
Nd-147	0.0063	1.38E-01	1280	2.E-04	0.01	6.50E+04	2.16E-06
Np-239	0.002	4.38E-02	7800	2. E-04	0.001	3.90E+04	1.12E-06
Ru-103	0.011	2.41E-01	1560	2. E-04	0.01	8.00E+04	3.09E-06
Ru-106	0.002	4.38E-02	100	2. E-04	0.01	5.00E+03	8.77E-06
Sb-122	0.002	4.38E-02	1860	2. E-04	0.01	9.50E+04	4.71E-07
Sb-124	0.002	4.38E-02	360	2. E-04	0.01	1.80E+04	2.44E-06
Sb-125	0.002	4.38E-02	1200	2. E-04	0.01	6.00E+04	7.31E-07
Sc-46	0.002	4.38E-02	360	2. E-04	0.01	1.80E+04	2.44E-06
Sm-153	2	4.38E+01	9200	2. E-04	0.01	4.60E+05	9.53E-05
Sr-90	0.032	7.01E-01	16	2. E-04	0.01	8.00E+02	8.77E-04
Ta-182	0.002	4.38E-02	620	2. E-04	0.001	3.10E+03	1.41E-05

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Table A-2. (continued).

Transport Plan Limits Radionuclide	Concentration (microCi/mL)	MAR✦ (curies)	DOE-STD-1027 TQV (curies)	Handbook ARF	DOE-STD-1027 ARF	TQV_{new} (curies)	MAR/TQV_{new}
Te-132	0.038	8.33E-01	600	2. E-04	0.01	3.00E+04	2.78E-05
Tm-170	0.002	4.38E-02	520	2. E-04	0.01	2.60E+04	1.69E-06
Zn-65	0.011	2.41E-01	240	2. E-04	0.01	1.20E+04	2.01E-05
Zr-95	0.012	2.63E-01	700	2. E-04	0.01	3.50E+04	7.51E-06
U	2.9E-05	6.40E-04	4.2	2. E-04	0.001	2.10E+01	3.05E-05
Pu*	2.45E-02	5.37E-01	0.52	2. E-04	0.001	2.10E+01	2.07E-01
Total							2.31E-01

✦ A filled tanker volume was conservatively assumed to be 5,790 gal.

* Of the radionuclides that could be present, the TVQ value chosen is the most conservative.

The sum of the ratios indicates that the full tanker MAR is below the allowable guidelines. It does not change the hazard categorization when added to the source term calculated in Table 3 of this report.

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A3. WASTE TANKS

CHLOROFORM IN WASTE TANKS

Using the sample result from the D&D report[A4] a quantity of chloroform was calculated by assuming a full volume in both the hot and warm waste tanks for a total of 5,500 gal.

Table A-3.A. Chloroform in Waste Tanks.

VOC	CAS No.	uG/L	MAR lbs	40 CFR 355 (Reportable Quantity/Planning Threshold)
Chloroform	67-66-3	25	1.14E-03	10 lbs /10000 lbs

The chloroform result is about four orders of magnitude below the reportable quantity guideline.

CURIES IN WASTE TANKS

Using the sample results from the D&D report[A4] a undecayed quantity of radiation (Table A3.B) is calculated by assuming a full volume in all three waste tanks. A ratio of sums (Table A3.C)was calculated from the inventory total using the ARF from DOE Handbook[A3] for a liquid spill as was used for the tanker calculation above. The sum of the ratios indicates that the tank inventory is below the allowable guidelines and does not change the hazard categorization when added to the source term calculated in Table 3 of this report.

Table A-3.B. Radiological Inventory in Waste Tanks.

Waste Tanks	Volume Gallons	Cs137 Picocuries/L	Co60 Picocuries/L	Cs137 Curies	Co60 Curies
Cold	1000	18000	104	6.80E-05	4.09E-08
Warm	5000	7570	790	1.43E-04	2.36E-06
Hot	500	7570	790	1.43E-05	2.36E-06
Total				2.25E-04	4.76E-06

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Table A-3.C. Liquid Waste Tank Source Term

Radionuclide	Curies Inventory	Curies DOE-STD-1027 TQV	DOE Handbook ARF	DOE-STD-1027 ARF	Curies New TQV	Inventory/New TQV Ratio
Co-60	4.76E-06	280	2.00E-04	0.001	1.40E+03	3.40E-09
Cs-137	4.76E-06	60	2.00E-04	0.01	3.00E+03	1.59E-09
Total						4.99E-09

APPENDIX A

A4. REACTOR VESSEL

The characterization report[A1] estimated that a total of 30,000 curies remained in the enclosed reactor vessel with the major radionuclides identified as cobalt and cesium. Tritium is a decay product from the neutron activation of the beryllium reflector. Its inventory was calculated using the reported maximum neutron flux of 2.5E15 neutrons/cm^2-sec, a cross section based on the average neutron energies, an activation time that totaled 1.8 years calculated from reported megawatt days of intermittent operation from when the new reflector was installed in 1970 to the final reactor shutdown in 1981, and a decay time of 18 years.[A1] Following the guidance in ID O 420.D, the threshold quantities of DOE-STD-1027 were modified using more appropriate ARF for the particular release mechanism the radiological inventories in the reactor vessel. The chosen ARF listed in attachment II of ID O 420.D was one that coincided with the material form of a fixed matrix in a sound and closed container. For the reactor radiological inventory, threshold quantities of DOE-STD-1027 were modified using the more appropriate ARF and ratios calculating quantity of each radionuclide to the Category 3 thresholds are listed below. The sum of the ratios indicates that the reactor vessel is below the allowable guidelines and does not change the hazard categorization when added to the source term calculated in Table 3 of this report.

Table A-4. Reactor Vessel Source Term.

Radionuclide	Curies Inventory	Curies DOE-STD-1027 TQV	ID O 420.D ARF	DOE-STD-1027 ARF	Curies New TQV	Inventory/New TQV ratio
Tritium	7.20E+06	1.60E+04	1.00E-06	5.00E-01	8.00E+09	9.00E-04
Cobalt	2.74E+03	280	1.00E-06	1.00E-03	2.80E+05	9.80E-03
Cesium	3.96E+02	60	1.00E-06	1.00E-02	6.00E+05	6.60E-04
Total						1.14E-02

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A5. FILTER CANISTERS

The filter inventory listed Table A5 below was calculated by a radiological engineer.[A5] Following the guidance in ID O 420.D, the threshold quantities of DOE-STD-1027 were modified using a more appropriate ARF for the particular release mechanism of the radiological inventories in the filters. The chosen ARF listed in attachment II of ID O 420.D was one specified for a fixed matrix material in a sound and closed container. The modified threshold quantities of DOE-STD-1027 using the more appropriate ARF and the ratios of each radionuclide to the Category 3 thresholds are listed below. The sum of the ratios indicates that the filter inventory is below the allowable guidelines and does not change the hazard categorization when added to the source term calculated in Table 3 of this report.

Table A-5. Filter Source Term

	Curies Inventory	TQV STD-1027	ID O 420.D ARF	1027 ARF	New TQV	Ratio inventory/ new TQV
Primary Filters						
Cobalt-60	8.40E+00	2.80E+02	1.00E-06	1.00E-03	2.80E+05	3.00E-05
Cesium-137	1.68E+02	6.00E+01	1.00E-06	1.00E-02	6.00E+05	2.80E-04
Strontium-90	1.29E+00	1.60E+01	1.00E-06	1.00E-02	1.60E+05	8.06E-06
Barium-137m	1.60E+02	2.20E+04	1.00E-06	1.00E-02	2.20E+08	7.25E-07
Total	3.37E+02					3.19E-04
Loop Filters						
Cobalt-60	6.90E+00	2.80E+02	1.00E-06	1.00E-03	2.80E+05	2.46E-05
Cesium-137	1.38E+02	6.00E+01	1.00E-06	1.00E-02	6.00E+05	2.30E-04
Strontium-90	1.06E+00	1.60E+01	1.00E-06	1.00E-02	1.60E+05	6.63E-06
Barium-137m	1.32E+02	2.20E+04	1.00E-06	1.00E-02	2.20E+08	6.00E-07
Total	2.78E+02					2.62E-04
All Filters	6.15E+02					5.81E-04

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A6. REFERENCE

A1. L.L. Kaiser, et. al., Characterization of the Engineering Test Reactor Facility, EGG-PR-5784, September 1982

A2. R.K. Murray, RML Gamma-Ray Analysis of Tanker Trailer (Hot Spot) Stages in ETR, RKM-01-99, July 28, 1999

A3. DOE Handbook, Airborne Release Fractions/Rates and Respirable Fractions for Non-reactor Nuclear Facilities, DOE-HDBK-3010, October 1994

A4. S. A. LaBuy and D. J. Kenoyer, Characterization and Decision Analysis for Engineering Test Reactor Facilities, INEL-96/0167, July 1996

A5. D. E. Hovis, TRA-704, -705, -706, -755,and TRA-642 Reactor Vessel Radiological Source Term Determinations, EDF-TRA-2000-004, June 2000