**Greater-Than-Class C
Low-Level Radioactive Waste
Characterization:
Estimated Volumes, Radionuclide
Activities, and Other
Characteristics**

*Greater-Than-Class C Low-Level Waste
Management Program*

August 1991

Greater-Than-Class C
Low-Level Radioactive Waste Characterization:
Estimated Volumes, Radionuclide Activities,
and Other Characteristics

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ABSTRACT

Planning for storage or disposal of greater-than-Class C low-level radioactive waste (GTCC LLW) requires characterization of that waste to estimate volumes, radionuclide activities, and waste forms. Data from existing literature, disposal records, and original research were used to estimate the characteristics and project volumes and radionuclide activities to the year 2035.

GTCC LLW is categorized as:

- Nuclear Utilities Waste
- Sealed Sources Waste
- DOE-Held Potential GTCC LLW
- Other Generator Waste

It has been determined that the largest volume of those wastes, approximately 57%, is generated by nuclear power plants. The Other Generator waste category contributes approximately 10% of the total GTCC LLW volume projected to the year 2035. Waste held by the Department of Energy, which is potential GTCC LLW, accounts for nearly 33% of all waste projected to the year 2035; however, no disposal determination has been made for that waste. Sealed sources are less than 0.2% of the total projected volume of GTCC LLW.

SUMMARY

The Greater-Than-Class C Low-Level Radioactive Waste (GTCC LLW) Program was initiated by the U.S. Department of Energy (DOE) in response to the U.S. Congress' enactment of Public Law 99-240 (National Low-Level Waste Policy Amendments Act of 1985). That law made DOE responsible for disposal of GTCC LLW (low-level wastes that exceed Class C limits). Characterization of those wastes is necessary in order to establish an estimate of the volumes, radionuclide activities, and physical forms associated with each waste stream. Characterization data are required to help support storage and disposal planning for GTCC LLW by DOE.

Initial estimates of the characteristics of GTCC LLW were made through a 1986 Energy Information Administration (EIA) survey of 1275 Nuclear Regulatory Commission (NRC) and Agreement State licensees. Inconsistencies and incompleteness of the information collected through that survey suggested a potential for uncertainty in any data correlation or analysis associated with such data. Uncertainties identified include regulatory, institutional, and technical issues that affect the projections of GTCC LLW stream characteristics. Those issues include disposition of DOE-held potential GTCC LLW, waste generated by fuel rod consolidation, and nonfuel-bearing components; concentration averaging; appropriate packaging factors for disposal; and timing of decommissioning waste disposal. A team of reviewers, including DOE, NRC, and private industry representatives, was assembled to participate in a technical review of those issues identified as having a major impact on GTCC LLW characteristic projections and to discuss what, if any, assumptions would be used in the waste characterization. The results of the technical review process provided direction into each of those issues, including suggestions concerning regulatory and institutional assumptions, and recommended experienced individuals for further research into technical issues where no assumptions could be developed in the technical review process.

Research into the issues of nonfuel-bearing components, packaging factors, concentration averaging, and further research on light water nuclear reactor GTCC LLW component production rates was conducted by independent researchers to support this study. Reports of this research are included in the appendices. Information from those studies includes waste characteristics such as volumes, radionuclide activities, and waste forms.

Those data were categorized by generator type (for the projections model classification) as

- Nuclear Utility Waste
 - Operations
 - Decommissioning
- Sealed Source Waste
- DOE-Held Potential GTCC LLW
- Other Generators Waste.

Information for nuclear utility waste, listed in two sub-categories, is based on several data sources. The primary sources of data for operations waste comes from characterization data of pre packaging analysis of commercial nuclear power reactor generated LLW. Those data are used for disposal packaging analysis based on radionuclide activity and volumes. Those data are also specific to individual reactor types (BWR and PWR) and manufacturer type (General Electric, Westinghouse,

Combustion Engineering, or Babcock & Wilcox). Data from each of those reactor types are included in this study.

Decommissioning waste volumes are based on specific reactor design characteristics (final safety analysis documents and reactor design specifications) while the radionuclide activities are based on computer modeling of reactor neutron flux data and activation data from surveillance capsules located near the reactor vessel wall. Calculations used for the decommissioning waste were carried out beyond the biological shield and normalized to the activities measured at the surveillance capsules.

Using the waste generation rates per fuel cycle and time between fuel cycles, an average annual waste production rate was estimated for nuclear utility operations waste. Decommissioning waste generation time tables are based on license closure dates and assume the material will be available for disposal at least three years after closure.

Sealed source data used in this study were taken from an NRC study which lists sealed source types, estimated numbers and volumes, and radionuclide activities. However, no timing or waste generation rates were given for those wastes.

DOE-held potential GTCC LLW information comes from a DOE data base on wastes with characteristics similar to GTCC LLW which were received by DOE because of the inability of the generator to safely store the waste. At the present time, it is unclear if the waste held by DOE will require disposal in an NRC licensed facility. To ensure those wastes are accounted for, all volumes are included in the final projections total. However, a final disposal determination will be made which may substantially reduce the volume of that waste which will be considered as GTCC LLW.

The "Other Generators Waste" category information came from data received from the EIA survey. Followup telephone contacts with those generators, whose business activities placed them in a specific category, were used to clarify inconsistencies and/or incompleteness in the 1986 EIA survey results. The data included in the projected GTCC LLW totals detail all waste forms, volumes, and radionuclide activities available.

The data were compiled and entered into a computer data base which was used in a modeling effort to project future volumes and radionuclide activities. Those data were projected using three cases, low, base, and high; the base case data represents the standard for the characterization described in this report, and low and high cases depict the limits of the base case. The base case represents results of research and the use of actual data from current waste generation practices. Those practices are specific to the various generator categories recognized in this study. The low and high cases are based on limits to the base case as recognized in the data used in this report.

Three scenarios for data projection are developed using: (a) unpackaged volumes, (b) packaged volumes based on the application of packaging factors to the unpackaged volumes, and (c) concentration averaging values applied to the packaged volumes (reflecting the change in GTCC LLW volumes from applying that scenario). These scenarios were used to represent the effects of packaging and concentration averaging on each specific waste stream. Waste must be packaged for shipment, storage, and ultimately disposal. Existing technology yields several packaging options for GTCC LLW which could be used for disposal. Variations in those options produces different packaging factors (the ratio of the volume of the disposal package to the volume of waste material

in the package). Optimization of the space in the liner is important to reduce disposal volumes and disposal cost.

Concentration averaging uses the concept of blending similar components or materials together and averaging the radionuclide activity of the waste in the container over the volume of the waste in the container or, in some cases, over the volume of the container.

These three scenarios are each represented in the report with low, base, and high cases, resulting in nine data sets for projected volumes. The projected data is calculated using annual waste generation rates, rather than fuel cycle intervals. The waste generations rates were normalized to annual rates to provide a consistent time scale for all waste generation. Waste volumes are accumulated annually, along with their respective radionuclide activities. Before a new annual volume is added to the old volume at the end of each projected year, the radionuclide activity of the old waste is decayed for one year. The model accounts for the annual decay of each annual volume of waste before the new volume is added.

The three different scenarios listed above are applied to the waste volumes, and the radionuclide activity is assumed the same for each component whether packaged or unpackaged.

Results of the projected data, as they apply to the base case volumes, are as follows:

<u>Scenario</u>	<u>Nuclear Utility Waste (m³)</u>	<u>Sealed Sources (m³)</u>	<u>DOE-Held Potential GTCC LLW (m³)</u>	<u>Other Generators Waste (m³)</u>
Unpackaged	8.41E+02	1.6E-02	a	a
Packaged	1.85E+03	6.08	1.08E+03	3.07E+02
Concentration Averaged	1.21E+03	b	b	b

a. Much of these wastes are already packaged. No information was available for packaging factors of those volumes.

b. Limited information was available on the affects of concentration averaging on these wastes, therefore no conclusion was made.

Radionuclide activity associated with each generator category is:

<u>Activity</u>	<u>Nuclear Utility Waste (curies)</u>	<u>Sealed Sources (curies)</u>	<u>DOE-Held Potential GTCC-LLW (curies)</u>	<u>Other Generators Waste (curies)</u>
Total	6.50E+07	3.03E+05	5.38E+05	2.92E+03

The total volume and radionuclide activity for all base case packaged GTCC LLW is:

Volume	3.25E+03 m ³
Radionuclide activity	6.58E+07 Ci

The projected volume data show a substantial increase between the unpackaged and packaged waste scenarios. The increase represents void space in the disposal liner package. Concentration averaging, as applied to nuclear utility waste, reduces the total disposal volume by classifying packages as Class C waste through blending of components and their radionuclide activities.

Data suggest the largest percentage of GTCC LLW volume and radionuclide activity will be generated by the nuclear utility industry. The majority of that waste is activated metals; filters and ion-exchange resins make up the remainder of the waste. Another large volume of potential GTCC LLW comes from DOE-held potential GTCC LLW. No determination has been made concerning ultimate disposal requirements for DOE-held potential GTCC LLW. Portions of those wastes may not require disposal in an NRC licensed facility, which may reduce the projected volumes and radionuclide activity of that waste category substantially. Sealed source information is based on data from a 1989 NRC survey of NRC and Agreement State licensees. Although the volumes of waste associated with sealed sources are small, the radionuclide activity is the same order of magnitude as the DOE-held potential GTCC LLW. Waste generated by the "Other Generators" category represents a relatively small volume and limited radionuclide activity.

Packaging of GTCC LLW has the effect of more than doubling the volume of waste which will be available for disposal. If those wastes are concentration averaged, a substantial reduction in disposal volumes are recognized.

Data trends between low, base, and high cases for packaged waste show an overall three fold increase. The low case total (including DOE-held potential GTCC LLW) is approximately 2220 m³ while the high case (to 2055) total is approximately 6500 m³. The increases (in the high case) are from nuclear power reactor life extension (additional operations waste) and less packaging efficiency.

Previous estimates of GTCC LLW volumes have been as high as 19,000 m³. This report concludes that there is substantially less GTCC LLW than previously estimated. The majority of the waste is activated metals with large radionuclide activities. However, there is still a level of uncertainty associated with the GTCC LLW projections. Concentration averaging, if allowed by all future LLW disposal facilities, could reduce the GTCC LLW disposal volume. It is unclear how each of those facilities will deal with the concentration averaging issue. A clearly defined, all-inclusive list

of components for the Standard Contract for disposal of high-level waste and spent fuel may include more components than are mentioned in the current version. That definition may also reduce the total disposal volume of GTCC LLW. Further research, including research into decommissioning waste materials, is proposed which will reduce the uncertainties of those volumes and radionuclide activities.

FOREWORD

This report details findings of research into the issue of characterization of greater-than-Class C low-level radioactive waste (GTCC LLW). Much of the data required for this study were generated through research into issues where little information was available. The greatest research effort was centered on LLW generated by nuclear utilities. The largest volumes and radionuclide activities associated with GTCC LLW are generated by this industry, and the potential for documenting waste characteristics for those generators is greatest due to the level of detail available on waste generation practices.

To ensure a sufficient level of detail, the data were divided into individual waste streams and components such as filters or control rod blades. This allows for analysis of the waste at a greater level of detail than if it were generalized into basic categories such as metals and nonmetals. There are no presumptions that these data are without potential error. However, the research provided in this study has endeavored to identify the primary sources of error and furnish suggestions for additional research to reduce those errors.

Data for activated metals from operations and decommissioning waste from reactor internals are based on current knowledge of elemental concentrations of stainless steel, inconel, and zircaloy. Measuring the activation levels of ^{60}Co , the dominant radionuclide in those activated metals, and scaling those data to predict the niobium and nickel activation in metal, is the common practice for determining the nuclides listed in 10 CFR Part 61.

Stainless steel is a common alloy used in metallic hardware for the nuclear industry. This material contains elemental nickel, is suspected of having significant niobium elemental concentrations, and is commonly activated to above Class C limits when subjected to a neutron flux comparable to that in a light water reactor core (LWR). A wide variation in measured and reported elemental concentrations of niobium in zircaloy suggest uncertainty as to whether those components will be activated to GTCC LLW levels.

Limited data are available to clarify the niobium content in zircaloy. Therefore, the assumptions used in the modeling process to estimate future waste generation are based on concentration data gathered from existing disposal records of various LLW components. The primary zircaloy component that could be considered GTCC LLW is boiling water reactor (BWR) fuel channels. However, historical records for fuel channel disposal suggests that those components will not exceed Class C limits or will be disposed of with the fuel bundles.

Cost and radiation worker safety concerns make it difficult to gather detailed information on all potential GTCC LLW components. Nuclear utilities waste estimates, listed in the projections data, were developed using available disposal data. Those data provide the estimate for projected waste volumes and radionuclide activities for that waste category.

Information gathered from generators, other than nuclear utilities, is based on known disposal volumes and radionuclide concentrations and is also based, in part, on subjective reasoning of the waste generator as reported in the EIA survey results. Characterizing potential GTCC LLW before disposal options are available is costly and subjects radiation workers to unnecessary radiation exposure. Because of "as low as reasonably achievable" (ALARA) considerations, it is unlikely that additional data will be available in the near future for those waste types.

Sources for GTCC LLW projections data include:

1. Initial volume data (1986 inventories) for nuclear utilities and other generator categories are based on followup information to the initial Energy Information Administration (EIA) survey responses.
2. Incremental operations waste volumes and radionuclide concentrations used in projected production rate calculations for nuclear utilities are based on average waste disposal practices for individual reactors as reported in Appendix F. Those reactors are categorized by type (pressurized water reactor or boiling water reactor), manufacturer, and reactor model (each manufacturer produces more than one model of reactor). Those data are established from disposal records of representative reactor types.
3. The "Other Generators Waste" category initial and incremental data are from the EIA survey information. No other data were available for this study.
4. Sealed source initial and incremental data are from a 1989 Nuclear Regulatory Commission (NRC) study on estimated sealed source types and number of sealed sources.
4. Decommissioning waste components and radionuclide concentrations are based on current research data listed in this report and Appendix F.
5. DOE-Held Potential GTCC LLW data are from existing storage records of those waste forms.

Three cases were developed to give a range in the projections of GTCC LLW. The base case is intended to represent the most probable waste generation projections. That case is based on current generation rates and disposal practices using packaging and concentration averaging which are in use at the present time. The low case is the lower limits of the base case, assuming different practices in packaging and concentration averaging techniques. The high case is presented in two parts:

High Case (2035)

1. The base case unpackaged data (projected to 2035) are modified for use in the high case by including more material (core barrel) from decommissioning of nuclear utilities and using higher packaging factors and more stringent concentration averaging practices. These more stringent concentration averaging practices assume no averaging between components. However, it is assumed that the radionuclide activity is homogeneously distributed throughout that component.

High Case (2055)

2. The second high case is similar to the high case above, except that life extension is assumed for 70% of all operating or licensed reactors. It is assumed that all life extension will be for 20 years. Only operational waste volumes will increase, and decommissioning waste volumes will not be affected.

The "Standard Contract" for disposal of high level waste and/or spent nuclear fuel (10 CFR 961) lists several nonfuel-bearing components that may be included as high level waste and that are subject

to disposal at a high level waste repository. However, there are questions as to which components will comprise an all-inclusive list for high level waste disposal. Several components listed in this study will, or may, fall into this category. Several of those components of questions were listed as GTCC LLW on the EIA Survey or are routinely changed out of commercial nuclear reactors as operations waste and included in this report. It is not the intention of this study to try to clarify the issue of nonfuel-bearing components versus high level waste; hence, all components included in the projections will be listed individually, with volumes and radionuclide concentrations, allowing for changes in the final data as the status of those components is determined. Those components are marked in the tables in Section 10.

Concentration averaging (mixing or blending of similar materials with differing radionuclide concentrations) assumptions reflect the interpretation of that process by the existing disposal facilities. The base case is established through practices at Barnwell, South Carolina; low case is from the Richland, Washington facility; and the high case assumes homogeneous nuclide concentration over an individual component only--thus, no averaging or mixing between components.

There is no definite approach to predict what LLW practices will be used in the future. The best approach is to use existing practices and data obtained from those techniques. Accordingly, a basic assumption used in this study is that waste generation practices will not change in the future to produce more waste than do current practices.

Information presented in this report is based on the data available. Assumptions are applied to the data projections only after research and consultation with a technical review team.

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GREATER-THAN-CLASS C LOW-LEVEL RADIOACTIVE WASTE CHARACTERIZATION: ESTIMATED VOLUMES, RADIONUCLIDE ACTIVITIES, AND OTHER CHARACTERISTICS

1. INTRODUCTION

This report documents the characterization of volumes and radionuclide concentrations associated with greater-than-Class C low-level waste (GTCC LLW). This characterization is necessary as a support to planning and managing the GTCC LLW Program. The purpose of this study was to provide information about estimates of present and future GTCC LLW volumes and radionuclide activities, waste generators, and waste forms. Research and investigation estimates of the characteristics of GTCC LLW are furnished here in order to clarify several areas of uncertainty. The U.S. Department of Energy (DOE) began the GTCC LLW Program in response to a congressionally mandated responsibility, Public Law 99-240 [the Low-Level Radioactive Waste Policy Amendments Act of 1985 (LLRWPA)]. That law gave DOE responsibility for disposal of GTCC LLW. This study evaluates the characterization of GTCC LLW, that waste which exceeds the Class C limits defined in 10 CFR 61¹ and is generated by U.S. Nuclear Regulatory Commission (NRC) and Agreement State licensed radioactive waste generators.

Initially, data for this study were collected from an Energy Information Administration (EIA) survey² of potential GTCC LLW generators to provide the basis for establishing an inventory of GTCC LLW in 1985 and future generation rates. Several topics of uncertainty, along with data deficiencies, were recognized during the survey. Clarification of the recognized uncertainties was developed through technical research, investigation into current literature, and a technical review process.

Results of research into areas of uncertainty are also included in this study, using recognized authorities in the fields of radioactive waste management, packaging, and nuclear physics. The technical review process was used to help refine the scope of the project by determining the significant issues related to GTCC LLW characterization and by eliminating further consideration into those issues beyond the scope of the study.

2. BACKGROUND

Low-level radioactive waste is defined in the LLRWPA as radioactive waste that is neither high-level radioactive waste, spent fuel, nor by-product material (mill tailings). In the absence of a concentration-based definition for high-level waste, there is currently no upper limit for the concentration of radionuclides in low-level waste.

In 1980, Federal law mandated that states provide the availability of disposal capacity for all low-level radioactive waste generated within their borders, except for certain waste generated by the Federal government. Some waste, however, was potentially unsuitable for disposal in near-surface facilities because of relatively high concentrations of specified radionuclides. This situation resulted in a category of waste that states were initially responsible for but could not dispose of using available technology.

2.1 10 CFR Part 61

In 1983, 10 CFR Part 61 codified disposal requirements for three classes of low-level waste considered generally suitable for near-surface disposal: A, B, and C, with Class C waste requiring the most rigorous disposal specifications. Waste with concentrations above Class C limits for certain short- and long-lived radionuclides was identified as greater-than-Class C low-level waste. GTCC LLW is not suitable for near-surface disposal, except on a case-by-case evaluation of the waste and approval of the proposed disposal method by the Nuclear Regulatory Commission. Therefore, Part 61 defined the categories of LLW and better delineated the category of waste not generally acceptable for near-surface disposal, but it did not relieve the states of their statutory requirement to dispose of such waste.

In 1985, the LLRWPA corrected the situation by assigning the states responsibility for Classes A, B, and C low-level waste and by making the Federal government (Department of Energy) responsible for the disposal of GTCC LLW. The law also required that GTCC LLW be disposed of in a facility licensed by the NRC. In May 1989, the NRC amended its regulations in 10 CFR 61 to require disposal of GTCC LLW in a facility other than a near surface site unless disposal elsewhere is approved by the Commission.

2.2 Greater-Than-Class C Low-Level Radioactive Waste

Table 2-1 lists the 10 CFR 61 long-lived radionuclide concentration limits for Class C low-level waste, and Table 2-2 lists the 10 CFR 61 short-lived radionuclide concentration limits. GTCC LLW is defined as waste that contains radionuclide concentrations greater than the NRC limits for Class C low-level waste, as listed in Tables 2-1 and 2-2.

For wastes containing more than one radionuclide in either Table 2-1 or Table 2-2, GTCC LLW limits are obtained by applying the sum-of-fractions rule separately to long-lived and to short-lived radionuclides. The sum of fractions for each nuclide contained in each component is determined by dividing each nuclide's concentration by its Class C limit. These fractions (one for each nuclide in the waste) are added independently between Table 2-1 (long-lived radionuclides) and Table 2-2 (short-lived radionuclides). If the sum of fractions exceeds 1.0 for either short- or long-lived radionuclides, the waste is GTCC LLW. This defines the lower limit of GTCC LLW; there is no upper limit.

Characterization of GTCC LLW is necessary in order to establish an estimate of the volumes, radionuclide activities, and physical form associated with each waste stream. In this study, waste streams are developed and categorized to include specific components, including the waste in inventory in 1986 and the waste generated after 1986. Each component is characterized for production rate of: (a) volume, (b) radionuclide activity, and (b) physical form. Three cases for waste projection are developed in the report. The base case was developed using data developed from current technology, operating practices, and disposal procedures. Future waste production rates for individual components are based on information from average generation and disposal rates of the various generators listed in the study. High and low case scenarios are considered as upper and lower limits of the base case data scenario. Using upper and lower limits for the base case reflects results based on operating history to date.

Table 2-1. NRC Class C limits for long-lived radionuclides^a

<u>Long-lived Radionuclides</u>	<u>Concentration Limits (Ci/m³)</u>
¹⁴ C	8
¹⁴ C in activated metal	80
⁵⁹ Ni in activated metal	220
⁹⁴ Nb in activated metal	0.2
⁹⁹ Tc	3.0
¹²⁹ I	0.08
	<u>Concentration Limits (nCi/g)</u>
Alpha emitting transuranic nuclides	
> 5 year half-life	100
²⁴¹ Pu	3,500
²⁴² Cm	20,000

a. From 10 CFR 61, Table 1.

Table 2-2. NRC Class C limits for short-lived radionuclides^a

<u>Short-lived Radionuclides</u>	<u>Concentration Limits (Ci/m³)</u>
⁶³ Ni	700
⁶³ Ni in activated metal	7,000
⁹⁰ Sr	7,000
¹³⁷ Cs	4,600

a. From 10 CFR 61, Table 2.

2.3 Characterization Development

The GTCC LLW generation activities can be generalized into two categories:

- Operations waste from normal operations of facilities
- Decommissioning waste generated on a one-time basis as a result of closure procedures at a facility generating nuclear waste. This does not include final decontamination processes (those wastes are considered in the operations category).

Operations waste is generated continuously, while decommissioning waste is a one-time occurrence, which comes at the end of the operating life of the facility. Each may contribute to the estimated volume of GTCC LLW available for disposal. Transuranic (TRU) waste is, without regard to source or form, waste contaminated with alpha-emitting transuranium radionuclides (nuclides with atomic numbers greater than 92) that have half-lives greater than five years^a and concentrations greater than 100 nCi/g at the time of assay. Transuranic GTCC LLW nuclides are listed in Table 2-1 and are included in the waste projections inventory.

Mixed waste is another concern of GTCC LLW characterization, and is defined in DOE Order 5820.2A as waste containing both radioactive and hazardous components as defined by the Atomic Energy Act and the Resource Conservation and Recovery Act, respectively.

The LLRWPA is directed toward low-level waste (LLW), the disposal of which is the responsibility of the states. The Act (Section 3 (a)(1)) excludes from the state responsibility LLW that is "... (a) owned or generated by the DOE (b) owned or generated by the U.S. Navy as a result of decommissioning vessels of the U.S. Navy, (c) owned or generated as a result of any research, development, testing, or production of any atomic weapon, (d) any other low-level radioactive waste with concentrations of radionuclides that exceed the limits established by the [Nuclear Regulatory] Commission for Class C Radioactive Waste [greater-than-class C low-level waste], as defined by Section 61.55 of Title 10, code of Federal Regulations, as in effect on January 26, 1983."

a. This definition differs from the DOE definition which uses half lives greater than 20 years. The 5-year half life comes from 10 CFR 61, Table 1.

Previously available information on the characterization, volume inventory, and projected volumes of GTCC LLW was limited. The primary source of information for GTCC operations waste inventories to 1986 was an EIA survey of the major commercial generators of GTCC LLW (see Reference 2). Limited information on the projected decommissioning waste was provided by the respondents in the EIA survey. Two documents were referenced in earlier studies and the EIA survey for decommissioning waste volumes and activities. Those documents are NUREG/CR-0130³ and NUREG/CR-O672⁴ (*Technology, Safety, and Cost of Decommissioning a Reference Boiling Water/Pressurized Water Reactor*). The documents were reviewed for data on volumes, activities, and radionuclide concentrations of nuclear utility decommissioning waste. The volume and activity data in those reports are estimated from projections made on a limited number of nuclear reactors.

Further research into volumes and radionuclide concentrations of nuclear utility decommissioning waste (Appendix G) has suggested that those NUREG estimates are greater than observed data from operating commercial nuclear reactors. Packaging of waste volumes reported in those documents uses conservative packaging factors, allowing for large void volumes inside the disposal liner and incorporating less waste volume per disposal liner than are used in this study (Appendix H). Current research data developed for this study are used to project nuclear utility GTCC decommissioning waste volumes as well as other GTCC LLW.

A computer projection model^b was developed to estimate future waste volumes, and radionuclide activities for GTCC LLW. The projection model generates three cases: base, high, and low generation rates. The low, base and high cases project GTCC LLW through 2035; an additional high case also projects GTCC LLW through 2055. The additional 20 years between 2035 and 2055 include waste from proposed life extension of 70% of all operating commercial nuclear reactors. An extension of 20 years per reactor is assumed; however, no determination has been made for life extension at this date.

2.4 Survey Form

The 1986 EIA survey form, which was used to collect data originally intended for use in this study, is the DOE Energy Information Administration's (EIA) "NE-869 Greater-Than-Class C Low-Level Radioactive Waste and Radium Waste Data Form" (see Reference 2). The survey form (included in Appendix A of this report) provides a definition of GTCC LLW and specifies the wastes included in, and excluded from, the survey. The form requested information about GTCC LLW generators, waste-generating activities, current waste inventories, future waste generation (including decommissioning waste), and capabilities of storing the waste.

Although the survey form requested information about radium-226 waste, that information is not included in this study, since such wastes are not regulated under 10 CFR 61.

The survey was funded by DOE and was approved by the U.S. Office of Management and Budget.

b. Computer projection model developed for the Greater-Than-Class C Low-Level Radioactive Waste Program at the Idaho National Engineering Laboratory by EG&G Idaho, Inc., Idaho Falls, Idaho.

2.4.1 Distribution of Survey Forms

EIA survey forms were distributed and collected by the official information-gathering arm of DOE. Under the authority of the Federal Energy Administration Act of 1974, it is mandatory that the recipient respond to the survey.

A total of 1275 NRC and Agreement State licensees were mailed survey forms during July and August 1986. Of the forms mailed, 19 were returned marked "unable to locate." Those who did not respond by October 1986 were sent a second request letter. Of the surveys presumably received by the addressees, 1085 responses were received (86.4%); 171 forms were not returned. All of the nuclear utilities responded to the survey.

The survey was prepared in order to identify volumes and types of GTCC LLW that would require disposal. The survey did not attempt to address NRC and Agreement State licensees (those states who set their own licensing criteria) who do not hold broad Type A NRC licenses or similar Agreement State licenses, or those who are in the category of "general licensees." General licensees are those who have legal possession of sealed sources manufactured by a specific licensee. These general licensees have not been routinely identified or tracked in the past by the licensing agencies.

3. GENERATORS

Survey respondents who were identified as potential generators of GTCC LLW were categorized into 13 business types. Nine of the potential GTCC LLW generator categories reported having waste, or the probability of generating future waste. Types of business included are:

- Carbon-14 users^c
- Fuel fabricators^c
- Medical research institutions^c
- Nuclear utilities^c
- Industrial research and development firms^c
- Nuclear research reactors
- Sealed source distributors^c
- Sealed source manufacturers^c
- Nonmedical academic institutions^c
- Nonresearch medical institutions
- Analytical laboratories
- Waste service companies
- Manufacturers of devices containing sealed sources.^c

3.1 Greater-Than-Class C Low-Level Radioactive Waste Generators

Generators who had multiple business activities were classified by the primary activity that generated LLW. For this study, a generator is a site where at least one facility generates LLW. For instance, if two or more nuclear reactors are at the same site and are owned by the same utility, each is classified as a separate generator.

The following types of generators reported having GTCC LLW on hand in 1986; definitions of their operations follow:

- **Nuclear Utilities.** Operators of light water reactors (pressurized water reactors and boiling water reactors) are potential GTCC LLW generators. Operating procedures of light water reactors vary from reactor to reactor. Different operating practices and manufacturer design of individual reactors dictate the potential GTCC LLW that may be produced. Future nuclear

c. Those generators who reported having GTCC LLW

reactor waste volumes are developed based on average operating practices for individual manufacturer designs. GTCC LLW may include activated metals from standard operation procedures, ion-exchange resins and cartridge filters from decontamination efforts, and decommissioning wastes.

- **Nonmedical Academic Institutions.** The use of TRU, ^{137}Cs , and ^{90}Sr sealed sources is common in classroom and laboratory instruction and in research and development. However, those devices may be too small for consideration as GTCC LLW.
- **Carbon-14 Users.** GTCC ^{14}C wastes are generated in the manufacture of ^{14}C "tagged" chemicals used in research.
- **Manufacturers of Devices Containing Sealed Sources.** Sealed sources used in instruments and gauges are common products (e.g., ^{241}Am sources are used in smoke detectors). Although an individual's possession and use of smoke detectors are exempted from regulation, large quantities of the ^{241}Am sources at the manufacturer's site are regulated by the NRC. Generators in this category have indicated that waste sources may be classified as GTCC LLW.
- **Fuel Fabricators or Burnup Labs.** Fuel burnup labs (hot cells) and fabricators of nuclear fuel handle TRU materials (Pu) leading to the generation of transuranically contaminated GTCC LLW. In the past, some fuel fabricators developed mixed oxide fuel pellets containing varying amounts of ^{235}U and ^{239}Pu . A burnup lab examines irradiated nuclear fuel rods to identify structural integrity and to ascertain the percent of ^{235}U and ^{239}Pu burnup during operation. The process of examination results in Pu-contaminated GTCC LLW. A limited amount of reactor hardware may also be produced as GTCC LLW.
- **Medical Research Institutions.** Medical research institutions use TRU, ^{90}Sr , and ^{137}Cs sealed sources for medical therapy research.
- **Industrial Research and Development Firms.** Industrial companies use sealed sources for radiographing in gas chromatography (^{63}Ni) and for well logging (^{241}Am).
- **Sealed Source Distributors.** Sealed source distributors dispense sealed sources, but do not manufacture sources. Distributors become LLW generators only when they have a leaking or damaged sealed source returned to them that cannot be returned to the manufacturer.
- **Sealed Source Manufacturers.** Sealed source manufacturers develop and produce sources for use in gauges and devices, or for use in research or medical treatment. Used or obsolete sources are received by manufacturers on a case-by-case basis for storage or disposal.

4. WASTE CATEGORIES

Any facility that produces LLW containing 10 CFR Part 61 Class C radionuclides (see Tables 2-1 and 2-2) can theoretically generate GTCC LLW. For this study, the current and prospective routinely generated waste components are grouped as: (a) activated metals, (b) process wastes, (c) dry contaminated solids, and (d) sealed sources. Another waste form discussed in this study is DOE-held potential GTCC LLW. This waste is not identified in the modeling projections data base, but is included in the overall GTCC LLW total volume. Each of those categories may have dominant 10 CFR Part 61 Class C radionuclides.

Other GTCC LLWs reported, but not always routinely generated, are mixed oxide fuel pellets and organic liquids. Those wastes are identified in the projected data.

The waste categories listed in this study can further be classified as operations wastes or decommissioning wastes. Operations wastes are generated through normal/standard operating procedures of a facility. Decommissioning wastes are generated as the result of closure of a potential GTCC LLW generator, a one-time event that can account for a major volume of potential GTCC LLW. Wastes that are generated as a result of nonstandard procedures, and are generally produced on a one-time basis, are also included as operational waste. Those one-time wastes represent a small volume of material as compared to normal operations or decommissioning categories, and include the mixed waste included for this study (see Section 10.4).

The 10 CFR Part 61 Class C radionuclides of interest for each waste category are as follows:

Activated Metals

The primary activation product and other radionuclides ^{59}Ni , ^{63}Ni , ^{94}Nb , and ^{14}C

Process Wastes

The fission product radionuclides ^{137}Cs , ^{129}I , ^{99}Tc , and ^{90}Sr , the activation product radionuclide ^{63}Ni , and transuranics

Contaminated Solids

Transuranics (TRU)

Sealed Sources

Two distinct groups dependent on source type: transuranics (^{241}Am , ^{244}Cm , ^{238}Pu , and ^{239}Pu), and ^{137}Cs , ^{90}Sr .

4.1 Activated Metals

Activated metals components from nuclear power facilities, which are considered GTCC LLW in this study, can be categorized into operations waste and decommissioning waste forms. The waste forms are also categorized as originating from boiling water reactors (BWR) and pressurized water reactors (PWR). The potential GTCC LLW components associated with each of the waste categories are detailed in Appendix F, with more technical information in Appendix G. A summary of those components follows.

BWR Operations Waste

- Control rod blades
- Local power range monitors
- IRM/SRM instruments, dry tubes

PWR Operations Waste

- Thimble plug assemblies
- Incore instrument strings
- Primary sources

BWR Decommissioning Waste

- Core shroud

PWR Decommissioning Waste

- Core shroud
- Core barrel (high case only).

4.2 Process Wastes

Process wastes consist of the wet waste generated from the cleanup of liquids containing soluble and insoluble radioactive constituents. Those wastes are in the form of ion-exchange media, filter media, and cartridge filters, and are potential GTCC LLW because of the concentrations of ^{137}Cs and ^{90}Sr . The Class C limits of cartridge filters are also affected by the presence of ^{63}Ni . The process wastes generated by nuclear facilities include:

BWR Decontamination Resins**PWR Decontamination Resins****BWR Cartridge Filters**

- Control rod drive
- Fuel pool
- Underwater vacuum

PWR Cartridge Filters

- Seal water injection
- Seal water return
- Reactor coolant system
- Back-flush letdown
- Fuel pool

4.3 Dry Contaminated Solids

Dry contaminated solids are routinely generated at most facilities in the nuclear industry and by other commercial users of radioactive materials. These materials are in the form of contaminated plastics, paper, cloth, glass, metals, and wood, and are historically the largest volume of LLW generated. The routine generation of these waste forms is a result of surface contamination, and their specific activity is low, rarely exceeding Class A limits. For most facilities, it is unlikely that Class C limits for these waste forms will be exceeded. However, there are some fuel development

and hot cell facilities that could generate GTCC LLW as a result of transuranic surface contamination. Examples of those wastes are:

Contaminated Nonmetals

Contaminated nonmetal wastes are compactible and in most cases combustible. Those wastes that contain TRU radionuclides are usually segregated and evaluated separately from the other dry wastes before commingling for storage or disposal.

Contaminated Metals

Contaminated metal wastes are not easily compactible or combustible. The same types of facilities that generate contaminated nonmetal waste generate contaminated metal wastes. These wastes are specific to a few facilities, which were recognized in the EIA survey.

4.4 Sealed Sources

Sealed sources consist of small capsules, usually stainless steel that encapsulate relatively high concentrations of a single nuclide. Sealed sources are used in a wide range of applications, including industrial and medical applications, and become waste when they are no longer usable.

Two distinct groups of those sources have been identified in this study: (a) those containing TRU radionuclides and (b) those containing other radionuclides. Typical uses of each category are as follows:

Transuranic-Bearing Sources

- TRU gamma gauges
- X-Ray fluorescence sources
- Well logging devices
- Moisture gauges

Other Sources

- Medical therapy
- Gamma gauges
- Beta gauges
- Well logging devices
- Moisture gauges
- Calibration devices
- Gamma irradiators VI
- Gas chromatography
- Beta eye applicators

4.5 DOE-Held Potential GTCC LLW

GTCC LLW accepted by DOE from NRC and Agreement State licensees is summarized in Table 4-1. Those data were compiled in late 1990 and reflect information considered current on that date. The specific cases listed are included in the total volume data (see Section 10, Table 10-23). The volumes listed include waste stored by DOE until a disposal facility is in operation. It is unclear whether all of those wastes will require disposal in an NRC-licensed facility; some may be owned by DOE and may be acceptable for disposal in a DOE facility.

Table 4-1. DOE-held potential GTCC LLW

Volume	1.08 E+03 m ³
Radionuclide Activity	5.38 E+05 Ci
Major Radionuclides	Am, Pu, Sr, Be, Cs

4.6 Modeling and Projections

To facilitate modeling and data clarity, the waste forms listed in this study are categorized as follows:

1. Nuclear Utility Waste
Operations and decommissioning waste from:

BWR

General Electric

PWR

Babcock and Wilcox

Combustion Engineering

Westinghouse

2. Sealed Sources
3. DOE-Held Potential GTCC LLW
4. Other Generators Waste

Specific waste components associated with each category, or subcategory, are listed in Section 10, with volumes and radionuclide activities for each component.

5. UNCERTAINTIES IN GTCC LLW DATA FROM EIA SURVEY

Data collected in the 1986 EIA survey process was initially intended to provide information about many of the potential GTCC LLW generators. However, initial analysis of those data revealed incomplete, and in several cases, inconsistent trends and incomplete results in the data reported by specific generators. Therefore, the results were recognized as invalid and not used as the basis of this report. Uncertainties in those data may be attributed to one or all of several factors.

- Generators unaware of GTCC LLW definitions
- Lack of time or effort in completing the survey
- Lack of detailed information to characterize waste on hand
- Small generators have operating procedures that can vary with time.

The greatest uncertainty in EIA survey results comes from a lack of complete decommissioning data for use in determining the volumes and radionuclide concentration of reactor internal components. In 1986, little was known about other GTCC LLW inventory. That waste as explained in the 1986 EIA Survey had not been fully characterized because of the expense and the concern of radiation exposure to workers.

Specific factors contributing to the uncertainties of that data were identified for clarification before pursuing the project. Those factors are discussed in the following subsections.

Research in those uncertainties identified in this report are presented in Sections 6 and 7, with results in Section 10.

5.1 Concentration Averaging

Concentration averaging is the practice of placing similar LLW materials together in a container and averaging the radionuclide concentrations of those materials. For example, when GTCC LLW activated metals are combined with Class C activated metals, the resulting packaged waste may meet Class C standards. However, only like materials or components are averaged, no commingling of unlike material is acceptable.

There are three LLW disposal facilities in operation that accept NRC and Agreement State licensed waste. Those facilities will discontinue receiving waste from outside their state/compact regions at the end of 1992. Therefore, LLW disposal facilities are being developed by other states/compacts; these facilities are scheduled to open in 1996. It is unclear to what extent concentration averaging will be allowed at each of those disposal facilities. When guidance and criteria are established for each of those facilities, the effects of concentration averaging on disposal volumes of GTCC LLW will be better understood.

5.2 Estimating Sealed Sources

A second uncertainty found in the EIA survey data results is sealed source projections. The 1986 projections are low because the EIA survey did not reach a representative population of sealed source users; only a few with broad Type A NRC and similar Agreement State licenses were included. In a 1989 published report (Appendix D) the NRC has performed a more detailed sealed

source survey, the results of which have identified several orders of magnitude more sealed sources than did the EIA survey.

5.3 Disposition of DOE-Held Potential GTCC LLW

In some circumstances, DOE has accepted radioactive waste generated by licensees of the NRC and state regulatory agencies in response to requests from those agencies. DOE has also accepted, on a limited basis, waste by-products from licensees, generally as part of contracts for radioactive materials processing services by those contractors. Currently, GTCC LLW held by DOE is being stored. P.L. 99-240 requires that GTCC LLW generated by licensees of the NRC be disposed of in a facility licensed by that agency, but also authorizes DOE to dispose of radioactive waste that it "owns" in DOE-managed facilities. The method used to dispose of DOE-held potential GTCC LLW will be determined on an individual basis for each waste stream.

5.4 Projection of Operations Waste

The EIA Survey information provided by certain waste generators did not predict generation of any GTCC LLW from operations for the period 1985 to 2020. However, those waste generators had GTCC LLW in inventory from past operations. Several generators in the same category listed waste in inventory as of 1986 and predicted future generation of GTCC LLW. To clarify the discrepancy between similar generators, it was necessary to determine if there will be continued production of those wastes by all generators, or if they were produced through irregular practices or one-time events.

5.5 Timing of Receipt of Decommissioning Waste

The NRC generally grants operating reactor licenses for a period of 40 years. For the purposes of this report, it is assumed that decommissioning will occur at the end of a 40-year reactor working life, and the GTCC LLW will come to DOE after the first three years following reactor shutdown. There are 129 reactors either closed, operating, licensed to operate, or under construction. By the year 2035, assuming no new orders, all will be decommissioned. Life extensions, if granted, would affect the time DOE receives the GTCC LLW decommissioning volumes by up to 20 years, pushing decommissioning of some of those facilities back to 2055.

5.6 Nonfuel-Bearing Components

The exact volume of GTCC LLW generated from nonfuel-bearing components is uncertain. The concentrations of nickel and niobium are inconsistent from component to component, and little is known about exact or average concentrations of those elements. Activation of these elements generally controls the waste classification of activated metals. A lack of data from measurements of radionuclide concentrations in many irradiated components, especially reactor internals, adds to the uncertainty in the volume estimates for that waste form.

5.7 Ion-Exchange Resins

GTCC LLW may be generated by nuclear utilities in the form of ion-exchange resins and filters during routine operations, periodic decontamination, and final decommission. The ion-exchange resins and filters are the result of decontaminating the recirculation piping or the steam generator

channel head. It is unclear as to the level of radionuclide activity reached by those wastes. Operational practices can increase or reduce radionuclide activity levels in those materials.

5.8 Waste Packaging

NUREG/CR-0130 and NUREG/CR-0672 (References 3 and 4) provide estimated decommissioning volume estimates of reactor vessel internals that may be GTCC LLW. The packaged volumes presented in the NUREGs are high in part because of the assumptions that the waste would be packaged for near-surface disposal. More reasonable packaging factors are necessary for projections of GTCC LLW storage and disposal volumes.

6. TECHNICAL REVIEW PROCESS

Many of the areas of uncertainty described above cannot be resolved with available information. Therefore, in order to finalize this report, assumptions regarding the method of dealing with several of those areas were developed. The DOE GTCC LLW Program initiated a multiphased technical review process to aid in developing those assumptions.

A technical review team, composed of DOE personnel and representatives from external organizations representing regulators and waste generators, was organized. DOE offices that may be affected by the GTCC LLW projections (i.e., DOE-RW and DOE-EM) were asked to participate. DOE also requested cooperation from the NRC and from private industry.

The review process used a multiphased approach. In Phase I, DOE team members met to discuss specific issues related to the areas of uncertainty. Those issues included concentration averaging, the disposition of GTCC LLW currently held at DOE locations, timing of DOE receipt of decommissioning waste, nonfuel-bearing components, GTCC LLW from accidents, and ion-exchange resins. The DOE members of the technical review team developed assumptions for this study. Where necessary, the issues were carried over into future phases of the technical review process.

In Phase II, NRC representatives joined the DOE team members to address certain GTCC LLW issues. These issues included concentration averaging, packaged volume of decommissioning waste, radionuclide concentrations of reactor vessel internal components, and sealed source estimates.

In Phase III, DOE, NRC, and industry representatives met to discuss the remaining areas of uncertainty. Through this multiphased approach, preliminary assumptions regarding each of the issues were determined; further discussion of recommendations is found in Section 7, and application of these assumptions is discussed in Section 10.

7. RESEARCH TOPICS

A technical review process and further research into each of the areas of uncertainty were done to establish a guideline for assumptions used in the GTCC LLW projection process. Independent research into several topics is included in this study. Research into activated metals, packaging factors, and concentration averaging uncertainties was conducted by independent subcontractors for EG&G Idaho, Inc. A summary of the findings of those studies is included in this section. Data from those research efforts are included in the summation in this section and in the appendices of this study. Findings of the research studies are included in the modeled data projections. Discussions of recommendations by the technical review process are included in this section.

7.1 Operations Waste

The EIA survey data were reviewed for types and amounts of operations GTCC LLW produced by potential generators. Several uncertainties in the reported EIA survey data were addressed. Cost and ALARA considerations have hindered full characterization of routinely generated waste components that may still be in generator inventory. EIA survey information provided by several generators shows GTCC LLW in current inventory but predicts no additional generation of this waste in the future. Other generators in the same category reported GTCC LLW as current inventory and also predicted future production of this waste. In other examples, several generators reported GTCC LLW volumes equal to, or greater than, the total volume of waste reported by all of the other generators surveyed. Clarification of these discrepancies was necessary to determine estimates for the future generation rates of GTCC LLW.

Research was undertaken to provide insights into the operating practices of the various facilities in question. Survey forms were reviewed to determine which generators reported waste production practices that were seen as anomalous to the other generators in that specific category. The generators were later contacted by telephone to determine if the data reported on the EIA Survey were still accurate. In several cases, the generator reported a change in the original estimate of GTCC LLW. Reductions in GTCC LLW included compactible trash (not considered GTCC LLW when reevaluated) and projected fuel rod consolidation waste (not considered GTCC LLW, and in most cases the process was not expected to take place).

Generators that listed GTCC LLW in inventory, but expected no future generation of those wastes, were compared to those in the same category that had both present and future GTCC LLW estimates. In several cases there was a reduction in the amount of GTCC LLW predicted for the future, but for those generators who predicted no future waste in the EIA survey, no revisions were made by those generators for additional waste generated in the future.

As a result of the telephone contact, many generators made changes in the volumes or types of waste that had been reported. Several generators had included waste volumes that were much greater than all of the other generators in the same category. Followup contacts with the generators concluded that the initial estimates were too large, and a reduction in the original volume was appropriate. A reduction of a factor of ten was reported by two generators, both sealed source manufacturers. The largest reduction in those waste volumes was from compactible trash. Disposal history of that waste stream from 1986 to 1991 suggests that compactible trash will not reach Class C limits. Several nuclear power reactors had listed fuel rod consolidation material as GTCC LLW. Reevaluation of that estimate by the generator concluded that waste would not be disposed of as GTCC LLW if the procedure were developed and used. The original EIA estimate of GTCC LLW inventory was reduced primarily from sealed source manufacturers and nuclear power stations.

It is assumed that reactor core decontamination wastes from nuclear reactor shutdown procedures will be included as operations waste. The result of such an assumption will be that hardware from reactor internals will constitute the only nuclear utility decommissioning waste material considered for this study.

7.2 Ion-Exchange Resins

Calculational models were developed to estimate potential GTCC LLW volumes and to perform 10 CFR Part 61 waste classifications for filter and ion-exchange resin wastes generated in evaporator bottoms and during routine operations and decontamination and decommissioning activities. Worst cases for PWR and BWR reactor coolant purification wastes and liquid radioactive waste evaporator bottoms were developed based on industry fuel performance standards. For decontamination wastes, two PWR and BWR base-case decontaminations were developed as follows:⁵

<u>BWR</u>	<u>PWR</u>
Recirculation Piping	Steam Generator Channel Head
- full system/fuel in (FSFI)	- full system/fuel in (FSFI)
- full system/fuel out (FSFO)	- full system/fuel out (FSFO)

7.2.1 Operations Wastes

- Filter and ion-exchange resin wastes, generated for worst case PWR conditions, could exceed Class C concentration limits if not mixed with lower concentration filter or resin wastes in disposal packages, or if administrative limits are not established to limit the concentration on radionuclides in those wastes.
- The reactor water cleanup system wastes from the worst case BWR do not exceed Class C concentrations.
- Worst case PWR evaporator bottoms do not exceed Class A limits.
- Worst case BWR evaporator bottoms, although somewhat higher than PWR values, still do not exceed Class A limits.
- In accordance with 10 CFR Part 61.55 (a)(7), the sum of the fractions for the ratio to Class C classification (those values above 1 exceed Class C limits) for the worst case PWR and BWR are as follows:

	<u>BWR</u>	<u>PWR</u>
Resins	0.11	1.13
Filters	--	7.01

- It is a typical practice for PWRs to mix primary and purification filters and ion-exchange resins with lower activity filter and resin wastes or limit service life of the process filter or resin media so that waste classification for disposal packaging does not exceed Class C.

- If mixing wastes in packaging (with resultant lower average waste concentrations) is prohibited, then plant operators may have to increase change-out frequencies of purification filters to avoid generating GTCC LLW.
- A few plants with extreme core conditions may find it impractical to increase the change-out frequency to the degree required to avoid generating GTCC LLW and would, therefore, generate GTCC filter waste.

7.2.2 Decontamination Activities

- For both PWRs and BWRs, 10 CFR 61 Table 1 isotopes, particularly TRU, dominate the 10 CFR Part 61 waste classification determination (see Table 2-1 and 2-2).
- For PWRs, it is estimated that there will be no GTCC LLW generated from any decontamination activity during operation or at plant decommissioning.
- BWRs performing recirculation piping and full system/fuel out decontaminations, including those performed in conjunction with decommissioning, are estimated to generate no GTCC LLW.
- Full system/fuel in decontaminations would not be conducted at plant decommissioning; therefore, it is estimated that no filter or ion-exchange resin wastes generated at decommissioning will be GTCC LLW.
- Full system/fuel in decontaminations in BWRs are not expected to occur for the next five to ten years, and only then on a limited basis or under special circumstances wherein significant maintenance activities are expected and large radiation exposures are projected.
- For those BWRs with higher specific activity concentrations in the corrosion film removed in the decontamination, the wastes from full system/fuel in decontaminations can exceed 10 CFR Part 61 Class C concentrations.

7.3 Activated Metals/Nonfuel-Bearing Components^d

A large source of GTCC LLW is irradiated metal components from nuclear reactors; the commercial nuclear power industry is the largest source. Nonfuel-bearing hardware internal to the pressure vessel in nuclear reactors absorbs neutrons during reactor operation and becomes highly radioactive. The bulk of the activity is ⁶⁰Co; other radionuclides more important to waste classification usually have concentrations several orders of magnitude below its level. If the neutron flux seen by the hardware is sufficiently high, the other radioactive constituents could reach concentrations high enough to classify the material as GTCC LLW. Irradiated hardware that is potential GTCC LLW falls into two categories, that discharged in normal refueling and maintenance operations (operations waste) and that discharged in decommissioning the reactor at the end of its operating lifetime (decommissioning waste). Activity levels and volumes of nonfuel-bearing components from commercial power generation that could be classified as GTCC LLW material are evaluated from both categories.

d. Summary of a study by James Cline, *Evaluation of Potential for Greater-Than-Class C Classification of Irradiated Hardware Generated by Utility Operated Reactors*, Appendix E.

The nuclides ^{59}Ni , ^{63}Ni , ^{94}Nb and, to a lesser extent, TRU, by virtue of surface contamination, generally control the GTCC LLW classification of activated hardware. Nickel and TRU nuclides emit only weak x-ray, beta, or alpha activities that are nearly impossible to measure in the presence of the dominant cobalt radiation. Although it emits energetic gamma rays the niobium isotope is present at such low levels that it, too, is overwhelmed by the cobalt activity. Thus, practical waste classification techniques for irradiated hardware (as for most other utility-generated waste) relate all other activities to the observable cobalt concentration through scaling factors. Since cobalt is present in most reactor structural materials as trace material that varies widely from component to component (as is niobium in all but inconel), current classification schemes give rise to uncertainties in estimating concentrations for the desired activities. A lack of data from measurements of radioactivity in many irradiated components adds to uncertainty in the estimates, particularly for those items discharged at decommissioning. Little research has been done in the area of activation levels of reactor internals; further study is required to resolve uncertainties associated with those components. Many of the estimates rely solely on activation calculations that use estimates of the elemental compositions of the material, the average neutron fluence seen by that component and, in some cases, the neutron activation cross section for the reaction. Uncertainties in those values increase total uncertainty in estimated activities of the waste and, therefore, in estimated volumes of GTCC LLW.

The present study has compiled and evaluated data from many additional sources to expand a base of data from which to estimate the activity concentrations and volumes of GTCC LLW. Sources of those data include measurements made by or for utilities with irradiated hardware classified for disposal in commercial disposal sites, measurements of neutron flux in the appropriate regions of the reactor pressure vessel, analyses of elemental constituents of the particular structural material used for the components, and the activation-analysis calculations done for hardware (see Appendix G). Evaluations include results and assumptions for the activation analyses.

7.3.1 Radionuclide Concentrations

Table 7-1^e summarizes the evaluation of radioactive concentrations of five radionuclides important for waste classification of reactor-activated hardware removed in normal plant operation: ^{59}Ni , ^{63}Ni , ^{94}Nb , ^{99}Tc , and TRU (transuranic activity is a component of the surface contamination acquired from contact with the reactor coolant). The table lists one PWR and seven BWR components identified as having the greatest potential for classification as GTCC LLW: the control rod blade (CRB); three incore instruments, local power range monitor, intermediate range monitor, and short range monitor (LPRM, IRM, and SRM respectively); poison curtains; fuel support piece; and BWR fuel flow channel. The table expresses concentrations as ratios to the Class C activity concentration upper limits for that nuclide and highlights values that exceed 1.0 (GTCC LLW).

Data in the table are for typical discharge frequencies. Values for the ranges result from statistical analyses of measurement results. The table demonstrates the dominance of ^{63}Ni , a 10 CFR Part 61 Table 2 nuclide (see Table 2-2), in classifying waste. The other four activities are 10 CFR Part 61 Table 1 nuclides (see Table 2-1) and contribute primarily through the sum-of-fractions rule. The data in the table indicate that only the BWR in-core instrumentation may have concentrations that exceed Class C limits.

Table 7-1 has two entries for the in-core instruments, LPRM, IRM and SRM. These components are about 43 feet long, only 12 feet of which are in the reactor core (hot end). The portion labeled "hot end" represents this 12-foot section; the other entry averages the concentration over the entire

e. From Appendix E.

43-foot long piece. Although the in-core instrument tubes are GTCC LLW, they are low-volume items. The utilities usually dispose of them as Class C waste under the factor-of-ten rule (the high and low radionuclide activity must be within a factor of ten for any disposal liner configuration) by mixing them with the larger volume CRBs and averaging over the volume of stainless steel-304 (SS-304) in the disposal containers.

Table 7-2 lists reactor internal components with greatest potential for GTCC LLW classification and summarizes the radionuclide activity expected at the end of the reactor lifetime. This table expresses concentrations as ratios to the Class C upper concentration limits for that nuclide and highlights values that exceed 1.0. All values assume a 40-year reactor lifetime and a 75% duty cycle of full power operation. The spread between the high and low values is very large and represents the uncertainties from a lack of measured data. The high values are principally from computational estimates of neutron fluxes and cross sections, and the low estimates rely on measured scaling factors and a measurement set of neutron flux in a region outside the core. The base values are a log-mean average of the two and represent the best estimate that can be made at this time.

7.3.2 Elemental Constituents in Reactor Hardware Alloys

The report in Appendix G summarizes the data and analyses presented for elemental constituents of alloys used for most reactor hardware. The relative abundances of those elements, many of which are only trace constituents, are very important for classifying the irradiated pieces. The tables in Appendix A of that report include the nominal value, i.e., the manufacturer's specification, measured values previously reported⁶ (see Reference from Appendix G) for SS-304, and the average values from two additional sources, material chemically analyzed for its components and constituents deduced from measured activated radioactivity. The latter analyses include the activation cross sections and are a more pragmatic indicator of the expected radioactivities from activation.

Table 7-1. Summary of average radionuclide concentrations for irradiated hardware generated in normal operations

			Ratio of Activities to Class C Limits				
			Radionuclide				
<u>Component</u>			<u>Ni-59</u>	<u>Ni-63</u>	<u>Nb-94</u>	<u>Tc-99</u>	<u>TRU</u>
<u>BWR</u>							
<u>CRB</u>	Min:		0.02	0.10	0.03	0.0003	0.0032
	Max:		0.29	0.46	0.14	0.0200	0.0200
	Avg:		0.09	0.24	0.07	0.0100	0.0100
<u>LPRM</u> Hot End	Min:		0.11	0.54	0.16	0.0018	0.0180
	Max:		6.16	9.86	2.80	0.0300	0.3200
	Avg:		1.59	4.29	1.20	0.0100	0.1400
with Cold End	Min:		0.03	0.15	0.05	0.0005	0.0050
	Max:		1.72	2.75	0.78	0.0084	0.0023
	Avg:		0.44	1.20	0.33	0.0028	0.0008
<u>IRM</u> Hot End	Min:		0.85	4.00	1.15	0.0100	0.1300
	Max:		6.43	10.29	2.95	0.0300	0.3400
	Avg:		2.22	6.00	1.70	0.0200	0.2000
with Cold End	Min:		0.24	1.12	0.32	0.0028	0.0360
	Max:		1.79	2.87	0.82	0.0080	0.0950
	Avg:		0.62	1.67	0.47	0.0056	0.0560
<u>SRM</u> Hot End	Min:		1.13	5.29	1.55	0.0200	0.1700
	Max:		3.57	5.71	1.65	0.0200	0.1900
	Avg:		2.06	5.57	1.60	0.0200	0.1800
with Cold End	Min:		0.32	1.50	0.43	0.0050	0.0470
	Max:		1.00	1.60	0.60	0.0060	0.0530
	Avg:		0.57	1.55	0.50	0.0056	0.0500
<u>Poison Curtains</u>	Min:		0.80	0.36	0.10	0.0012	0.0120
	Max:		0.42	0.67	0.19	0.0023	0.0220
	Avg:		0.19	0.51	0.15	0.0016	0.0170
<u>Fuel Support Piece</u>	Avg:		0.004	0.200	0.005	0.0001	0.0050
<u>Fuel Flow Channel</u>	Min:		1.1E-04	1.7E-04	0.002		0.0870
	Max:		1.1E-03	2.9E-04	0.040		0.1400
	Avg:		3.6E-04	9.7E-04	0.010		0.0490

Note: Shaded numbers indicate activity ratios over Class C limits.

Table 7-2. Summary of high and low radionuclide estimates for reactor internal components

			Ratio of Activities to Class C Limits			
			Radionuclide			
<u>Component</u>			<u>Ni-59</u>	<u>Ni-63</u>	<u>Nb-94</u>	<u>Tc-99</u>
<u>PWR</u>						
Shroud	High:		8.5	13.3	10.3	0.22
	Base:		0.62	1.7	0.79	9.3E-03
	Low:		0.045	0.21	0.06	9.3E-04
Core Barrel	High:		1.1	1.8	0.77	0.01
	Base:		0.085	0.23	0.06	1.3E-03
	Low:		6.4E-03	0.03	4.6E-03	1.8E-03
<u>BWR</u>						
Core Shroud	High:		2.7	4.3	1.7	2.7E-03
	Base:		0.13	0.55	0.13	2.7E-04
	Low:		5.9E-03	0.07	0.01	2.7E-05

Note: Shaded numbers indicate activity ratios over Class C limits.

7.4 Packaging Factors^f

Packaging efficiency of GTCC LLW will dictate the volumes of material available for disposal. Inefficient packaging configurations could increase the volume of waste several times, resulting in disposal of containers with large internal void spaces. Reducing those void spaces is necessary to reduce the overall disposal volumes of GTCC LLW.

Estimates are developed in this section for a set of three values that represent a range for the packaging factor (PF) for several potential GTCC LLW streams. The PF is defined as the volume of a GTCC LLW disposal container divided by the original, as-generated or "unpackaged volume" of the wastes loaded into that container.

Methods for determining a PF consider the unpackaged volume of a potential GTCC LLW prior to any treatment and packaging for disposal. A PF estimates what the unpackaged volume will become when the wastes are actually processed (if processed at all) and packaged into disposal containers.

The PF for a specific GTCC LLW stream is the number of times an unpackaged volume will be increased, or decreased, to take into account a predisposal treatment (if any) and packaging for disposal. Packaging factors take into account any processes that reduce or increase an original unpackaged volume of waste, the volume inside a container not occupied by the waste, and the volume of the waste container itself.

A set of three values is proposed as the range for a reasonable estimate for the PF for each potential GTCC LLW. Values are determined considering: (a) existing low-level radioactive waste processing, handling, packaging, and transportation technologies, and (b) the uncertainties in forecasting which technologies will be used in the future when the GTCC LLW is actually generated, treated, packaged, and transported to disposal. The three values are the base case or most likely value for a PF, a high case PF that corresponds to the largest volume the GTCC LLW might be expected to become for disposal, and a low case PF for the smallest volume expected. The high and low cases are not proposed as worst case extreme values but rather as reasonable values for the range of the base case or most likely value for the PF.

PFs are the reciprocal of packaging efficiencies. Packaging efficiency is an unpackaged volume of a GTCC LLW inside a waste disposal container divided by the volume of the outside dimensions of the waste disposal container. The assessments of reasonable PFs are developed below by determining the packaging efficiencies and then taking the corresponding reciprocals.

The three types of GTCC LLW evaluated are: (a) activated metals, (b) sealed sources, and (c) all other wastes. The specific types of materials evaluated in each category and the generators are listed in Table 7-3. Packaging factors for each of these categories are included in Table 7-4.

f. Summary of a study by Geoff Quinn and Phil Grant, *Packaging Factors for Greater-Than-Class C Low-Level Wastes*, Appendix F.

Table 7-3. Specific types of materials evaluated for packaging factors in each category

POTENTIAL GTCC LLW AND GENERATORS	
<u>GTCC LLW Components</u>	<u>Generator</u>
<u>Activated Metals</u>	Nuclear Power Plants
BWR Operations	
Control Rod Blades	
Incore Instruments	
PWR Operations	
Thimble Plugs	
Incore Instruments	
Primary Sources	
BWR Decommissioning	
Core Shroud	
PWR Decommissioning	
Core Shroud	
Core Barrel	
<u>Sealed Sources</u>	Manufacturers and Users
<u>All Other GTCC LLW</u>	
Decontamination Resins	Nuclear Power Plants
Cartridge Filters	Nuclear Power Plants
Aqueous Liquids	Industrial and Academic Users
Solidified Liquids	Fuel Testing and Burnup Evaluation Facilities and Industrial Users

Table 7-4. Packaging factors for categories

GTCC LLW PACKAGING FACTORS USED IN VOLUME PROJECTIONS			
Potential GTCC LLW Stream	Base Case	High Case	Low Case
<u>Activated Metals</u>			
BWR Operations			
Control Rods	5	10	4
Incore Instruments	5	10	3
PWR Operations			
Thimble Plug Assemblies	20	30	10
Incore Instruments	5	10	3
Primary Sources	1	1	1
BWR Decommissioning			
Core Shroud	2	3	1.4
PWR Decommissioning			
Core Shroud	2	3	1.4
Core Barrel	2	3	1.4
<u>Sealed Sources</u>	380	1,000	2
<u>All Other GTCC LLW</u>			
Decontamination Resins	1	2	0.2
Cartridge Filters	2	4	0.1
Aqueous Liquids	0.2	3	0.1
Solidified Liquids	1	1.6	0.8

7.4.1 Activated Metals

The large components, core barrels and core shrouds, are evaluated in Appendix H, Section 2.1. The small components--BWR control rods, incore instruments, etc.--are considered in Appendix H, Section 2.2. For each type of component, background information and assumptions on the scenarios for packaging of such materials are presented. The calculational models are then described and results provided. Estimates of the base, high, and low cases for PFs used in disposal volume projections are made from an evaluation of the results.

An important consideration in the analyses is the assumption that different waste streams of GTCC LLW are not mixed in the same waste disposal container. This assumption does not reflect current practices for the packaging of LLW for burial at commercial LLW disposal sites. Within the guidelines acceptable at each disposal site, various small-size activated metal components are normally mixed together and disposed of as commercial LLW. While the acceptability of this practice for commercial and DOE disposal in the future will influence the types and quantities of potential GTCC LLW that are disposed of as GTCC LLW, this study ignores the possibility that activated metal waste streams will be mixed in determining the reasonable PFs for each waste stream examined.

In using this approach, a higher PF than with mixing will result for each waste stream, i.e., more containers of GTCC LLW will be predicted than will be generated without this assumption. Mixing different waste streams would allow small components to be placed into containers with pieces of large components and other small components to fill in void spaces in those waste containers. Taking advantage of mixing to utilize costly disposal volume seems a reasonable practice that would be applied to GTCC LLW in the future.

However, the PFs for unmixed waste streams are higher than the cases where mixing would be performed. As will be seen by the range of PF values in the results, such a small difference should not affect the usefulness of the values proposed for the PFs. Also, this approach is helpful since it allows computer modeling of an individual waste stream and prevents the problem of mathematical permutations--the many possibilities that would arise if waste streams were evaluated as mixed.

7.4.2 Sealed Sources

The principal references of data for this section are provided by the NRC survey (Appendix D) on surplus sources and direct input from manufacturers and users of sealed sources within the industry.^g Also included are the major types of sources, sizes specific to the sources, the devices, and the packages provided for handling, transport, and storage. To simplify the process in determining packaging scenarios and packaging factors based on the significantly large number of sealed source configurations (>2000), the sources and devices were characterized under three categories. The three major categories are moisture/density gauges, well logging units, and fixed/test gauges. These three types of sources are analyzed and modeled to determine the scenarios for the base case packaging factor, as well as for the high and low cases.

g. Information on sealed sources was gathered from the following industrial sources: Troxler Electronic Labs, Inc., Cornwallis Road, Alexander Drive, Research Triangle Park, North Carolina, 27709; Nuclear Environmental Engineering, Inc., Registry of Sealed and Device Sources; Amersham/Searle, private communication with Brian Baker; Parkwell Laboratories, Inc., Registry of Sealed Sources and Devices; New England Nuclear, Registry of Sealed Sources and Devices; CPN Company, 2830 Howe Road, Martinez, California, 94533; Monsanto Research Corporation, Dayton Laboratory, Dayton, Ohio.

Previous estimates of packaging factors for surplus sealed sources ranged from encapsulating nonreusable sources of approximately 40 cm³ (currently approximately 3700 of this type projected) with cement in 55-gal containers (PF of 6600) to extracting and consolidating unshielded sources in separate HLW canisters (PF of 14.5). The high case encapsulation scenario was based on stabilization requirements for commercial disposal of Class C LLW and will not be considered applicable to the GTCC LLW projections. Those previous cases did not specifically consider the practical condition of some surplus sources contained in devices, gauges, and deployment units and the potential return of such devices to DOE in the original package and transport containers.

The NRC survey concluded that a large number of sealed sources that will be declared GTCC LLW exist. Approximately 6% of those sources and devices are awaiting disposal or transfer to a storage location. The predominant source materials are ²⁴¹Am, ¹³⁷Cs, ²⁴⁴Cm, and ²³⁸Pu and ²³⁹Pu. Of the inactive sources, some have been returned to the manufacturer for storage, but the majority are being held by the users because of cost constraints and/or manufacturers' refusal to accept returns. The users usually store those sources and devices in their original receiving configuration. Of those returned to the manufacturer, the majority of devices and source gauges have been disassembled to the extent practical to remove the electronics and ancillary equipment for optimum storage and consolidation.

The majority of GTCC LLW sources and devices are transported from the manufacturer in 7A (Type A) containers. However, some sources have been transported in Type B containers because of the higher specific activity and associated dose rate/transport index and special shielding requirements. The majority of those Type A containers are retained by the users, while the Type B containers are usually returned to the manufacturer. Generally, the users and manufacturers retain the shipping containers for storage and transport of the devices/sources.

7.4.3 Types, Quantities, and Sizes of Sealed Sources

The NRC survey represented a sampling (approximately 25%) of their licensed users and manufacturers of sealed sources. Those sample data were then extrapolated for both specific and general licensed users, including those under Agreement State licensing, to project the current estimate of GTCC source quantities. The NRC survey focused primarily on users who maintain a specific license for use of sealed source material. General license projections were based on NRC knowledge and data base records.

The greatest quantity of sealed sources and devices currently in use under specific licenses is identified with well logging units, density/moisture gauges, and the fixed/test gauges. The major sealed sources under the general (broad) license are of the fixed/test gauge type.

Sealed sources used in devices present a unique problem because most sources are small but require much larger containers for shielding, handling, deployment, storage, and eventual disposal. The actual sources are usually small, sealed into stainless steel jackets, and contain one or more radionuclides.

A typical moisture/density gauge source contains ¹³⁷Cs and ²⁴¹Am. The maximum activities typically are 10 mCi ¹³⁷Cs and 60 mCi ²⁴¹Am. The maximum threshold activity used by the NRC to determine if those type of sources exceed Class C limits, (based on draft guidance) is 27 mCi for ²⁴¹Am and other transuranics and 910 Ci for ¹³⁷Cs. A typical moisture/density source is 1.5 cm in length and 1.0 cm in outside diameter (~1.2 cm³).

A typical oil well logging source contains ^{241}Am , ^{238}Pu , or ^{137}Cs . The maximum activities range from 10 to 20 Ci of transuranic material and 2 Ci of ^{137}Cs . The sizes of oil well logging sources are larger than the other type sources and typically are 2.54 cm in diameter and 7.5 cm in length (38.6 cm^3).

A typical GTCC test/fixed gauge source contains ^{137}Cs , ^{60}Sr , ^{241}Am , and ^{244}Cm . For the most part, the sources are smaller and contain significantly smaller quantities of source material than oil well logging sources. The test/fixed gauges typically are 0.8 cm in length and 4 cm in outside diameter, with a total volume of 10 cm^3 . A detailed description of major sources used for analytical and industrial applications is provided in Appendix H.

Packaged sealed sources are normally contact handled, but the sealed source capsule or jacket is not normally contact handled. The source is usually located within a container or device that is shielded to allow contact handling. The size of those devices is usually a function of the type of source, the shielding and containment requirements, and the manner in which the unit is deployed. Those devices are usually transported in special cases or packages provided by the manufacturer to protect the device. Tables 7-5 and 7-6 provide a summary of the range of sizes for the individual source, the shielded container, the device, and the transport package (Appendix H) for each of the major sealed sources identified within the NRC survey.

7.4.4 All Other GTCC LLW

The volume of GTCC LLW generated by NRC and Agreement State licensees may not be the volume disposed of by the DOE. The volume of the waste containers into which GTCC LLW is placed will constitute the disposal volume. Empty volume in a waste container will be minimized by the generators if it is cost-effective compared to costs for storage and disposal of containers having the empty space.

Since many GTCC LLW forms are similar to Classes A, B, and C wastes that can be disposed of in licensed commercial disposal facilities, and since disposal costs at such facilities are high enough to require waste generators to pursue volume reduction operations, the current industry practices for volume reduction of Classes A, B, and C wastes will likely be considered for treatment of GTCC LLW. If there is a relatively higher cost for disposal of GTCC LLW, additional cost-effective technologies for volume reduction of GTCC LLW might be introduced in the future. Likewise, some available but as yet uneconomical technologies may become cost-effective in the future.

A generic base case for a packaging factor for "Other Generator" GTCC LLW is a factor of 1.0. This value reflects the uncertainty in the volume reduction technologies that can be used for GTCC LLW and in the amounts of the different types of materials that will be generated. From current projections of unpackaged "Other Generator" GTCC LLW, the mix of such wastes will be compactible and noncompactible materials. A reasonable assumption is that the volume decrease from compaction, incineration, and other volume potential reduction technologies will offset the volume increase from packaging noncompactible materials into waste containers. With this assumption, the gains from volume reduction technologies would offset the historical increase in the overall volume of wastes from packaging the wastes.

The low value for the packaging factor for all other GTCC LLW is 0.5, or a prediction that a net reduction will occur from the amount generated to the amount disposed. This value reflects the significant advances in supercompaction and radioactive waste incineration that have recently become available for low-level waste treatment in the United States. With volume reduction factors of 300

to 1 for incineration and 10 to 1 for supercompaction, treatment of significant amounts of the GTCC LLW by these methods would bring the PF down considerably. The proposed value reflects that these new technologies will be available and used to some extent by the GTCC LLW generators.

Those wastes that were reported as packaged were given a packaging factor (PF) of 1.0. No information is available to determine packaging procedures for that waste material.

The high value for the packaging factor for all other GTCC LLW is expected to be 1.5 or a packaging efficiency of 0.67. This value represents an inability to use volume reduction technologies to a significant extent for GTCC LLW because of the radionuclide concentrations that make the waste GTCC LLW in the first place. Commercial treatment centers may be unwilling to allow treatment because of the potential for significant contamination of expensive equipment. Also, GTCC LLW generators may not have the volume of GTCC LLW to justify installing volume reduction equipment. The proposed value reflects the unique nature of this waste and the potential difficulty in processing it economically.

Table 7-5. Source and package size ranges

MOISTURE/DENSITY GAUGES (approximately 14,000 in existence)		
	Minimum	Maximum
Source Size	0.75 cm ³	4.6 cm ³
Shield Size	3,670 cm ³	4,140 cm ³
Gauge Size	14,680 cm ³	20,700 cm ³
Package Size	96,500 cm ³	104,950 cm ³

Table 7-6. Major GTCC LLW sealed/device sources

<u>TYPES</u>	<u>SPECIFIC LICENSE</u> <u>Estimated number of sources</u>
	(27,156)
1. <u>TRU-Bearing Sources</u>	
Well logging units	~2,000
Density/moisture gauges	~14,200
Gamma gauges	~200
Pacemakers	~130
X-Ray fluorescent	<1% of total
Smoke detectors	--
2. <u>Non-TRU-Bearing Sources</u>	
Test gauges (industrial/analytical)	~3,300
Fixed Gauges (gamma/beta, thickness)	~800
Calibration sources	~480
Category I sources	~450
Irradiator sources (Cat. II, III, IV)	~440

7.5 Concentration Averaging^h

The concentrations of radionuclides within a disposal liner are determined by dividing the estimated radioactivity by the volume or mass of the waste material within the package. This calculation assumes that the waste material within the package is homogeneous and that the calculated concentrations are averaged over the packaged waste. In some cases, high concentration wastes are combined with low concentration wastes in the same package; in this instance the concentration calculations are based on the combined radioactivity divided by the combined waste volumes. That practice is referred to as "concentration averaging." How that practice is applied to different types of waste affects the 10 CFR Part 61 waste classification and, for the purposes of this study, significantly affects wastes that will meet or exceed NRC waste form Class C limits.

To reflect current practices and disposal site criteria, possible GTCC LLWs were divided into four major categories:

- Activated Metals
- Process Wastes
- Contaminated Solids
- Sealed Sources.

To provide a framework for DOE volume projections, three concentration averaging scenarios were considered:

High-Volume Case -- Disposal criteria that reflect prospective practices leading to high GTCC LLW volumes

Base-Volume Case -- Disposal criteria that reflect current practices leading to moderate GTCC LLW volumes

Low-Volume Case -- Disposal criteria that reflect practices leading to low GTCC LLW volumes.

7.5.1 General NRC Guidelines

The documents published by the NRC allude to concentration averaging but do not specifically discuss the concept. The regulation, 10 CFR 61.55(a)(8), provides that "the concentration of a radionuclide may be averaged over the volume of the waste." The NRC Branch Technical Position (BTP)ⁱ on waste classification expands on 61.55(a)(8). It indicates that the principal consideration is whether the distribution of radionuclides is reasonably homogeneous. In one example, the BTP indicates that a small concentrated source, such as a gauge or check source, may be commingled with trash and averaged over the trash volume. In another example, the use of stabilization media as part

h. Summary of a study by Peter Tuite, *Impact of Concentration Averaging Greater-Than-Class C Low-Level Radioactive Waste Volume Projections*, Appendix G.

i. U.S. Nuclear Regulatory Commission, "Final Waste Classification and Waste Form Technical Position Papers," May 1983.

of the concentration calculation for filters and sealed sources is discussed. Limits on concentration averaging are not defined, but the BTP recognizes that the concentrations of hard-to-detect radionuclides, which are the primary drivers of classification status, must be accurate to within a factor of 10.

Additional guidance is provided in a draft regulatory guide on waste classification that was never published. This draft indicates that concentration averaging or blending is acceptable, provided the concentrations of the blended waste are within several orders of magnitude.

While each disposal site has applied Part 61 rules and NRC guidelines differently, concentration averaging of packaged waste is routinely done.

7.5.2 Current Practices

Current practices arise from the requirements in 10 CFR Part 61, the NRC Branch Technical Position on waste classification, and the licensed disposal site application of those requirements to specific commercial disposal sites. Current practices by waste category are listed below.

Activated Metals

Different types of components, and different pieces from different types of components, are usually packaged in the same disposal liner. In some cases, activated metals can also be encapsulated within a medium like cement. The Part 61 classification status of the stabilized metal can sometimes be determined by dividing the metal activity by the total volume, including the stabilization medium.

Process Wastes

Different batches of resin are combined in the same disposal liner. Similarly, different types of filter cartridges are combined in the same package. In some cases, spent filters and spent resins, or spent filters and activated metals, are combined in the same package. Both spent resins and filter cartridges can be stabilized within a medium like cement. The Part 61 classification status of the stabilized waste can be determined by dividing the waste activity by the total volume, including the stabilization medium.

Contaminated Solids

Current practices vary for dry waste and can include both incineration and compaction to reduce disposal volumes. Compaction can increase concentrations by a factor of 10, and incineration could increase concentrations by as much as a factor of 100. Wastes are segregated primarily by physical form so as not to differentiate between waste classes. When dry waste is incinerated, the incinerator ash is sometimes stabilized. The Part 61 classification status of the stabilized ash can be determined by dividing the ash activity by the total volume, including the stabilization medium.

Sealed Sources

By design, sealed sources contain high radioactivity in small volumes. This leads to high concentrations of Part 61 radionuclides. Under NRC guidelines, those sealed sources can be stabilized or encapsulated within a medium like cement. When encapsulated, the Part 61 classification status is determined by dividing the activity of the sources by the encapsulated volume.

Practices vary with waste category. Concentration averaging criteria vary among waste categories and among the commercial disposal sites. Current criteria at the commercial disposal facilities are discussed below.

7.5.2.1 Barnwell Disposal Facility. The Barnwell disposal facility in South Carolina received about 68% of the commercial low-level radioactive waste disposed of in 1989. Its application of 10 CFR Part 61 requirements to concentration averaging has a significant impact on how wastes are packaged for disposal. The criteria applied to commingling waste vary with the waste category as follows:

Activated Metals

Disposal liners containing activated metals must satisfy what is commonly referred to as the "Rule of 10." Application of this rule allows concentration averaging among the same types of components but limits, to a factor of 10, concentration averaging among individual components. For example, control rod blades of LPRM strings with high concentrations of Part 61 radionuclides cannot be combined with control rod blades of lower concentrations, unless the concentrations of Part 61 Table 1 radionuclides and Part 61 Table 2 radionuclides (see Tables 2-1 and 2-2) for the two components are within a factor of 10 of each other. Once the Rule of 10 is satisfied, the Part 61 classification status can be determined by dividing the total activity of all the components by the total volume.

Process Wastes

Where different batches of resins or filters are placed in the same disposal liners, the Part 61 classification status is determined by dividing the total activity of the waste by the total volume of the waste. Where different types of waste (i.e., resins and filters) are combined in the same disposal liner, the Part 61 classification status of each waste type is determined separately. The radioactivity of each waste type is divided by the volume for each waste type, and documented separately.

The addition of cement increases the volume of the waste and reduces the concentration of Part 61 radionuclides. Under NRC guidelines, the Part 61 classification status of the stabilized waste can be determined by dividing the total activity by the total volume, including the stabilization medium. Where spent resins are stabilized, the total volume can be applied. However, where filter cartridges are stabilized, Part 61 classification status must be determined by dividing the total activity by the volume of the filter cartridges only.

Contaminated Solids

The packaged waste, with and without volume reduction, is considered homogeneous; the entire volume is used for classification determination. Incinerator ash may need stabilization; criteria for resins would apply.

Sealed Sources

Requests for disposal of encapsulated sources are addressed on a case-by-case basis. Generally, sources containing non-TRU radionuclides with activities up to 10 Ci, and sources containing TRU radionuclides with activities up to 50 mCi, are approved for encapsulation.

7.5.2.2 Richland Disposal Facility. The Richland, Washington, disposal facility received about 25% of the commercial LLW disposed of in 1989. Its application of 10 CFR Part 61 requirements to concentration averaging is similar to the NRC guidelines. The criteria applied to commingling waste vary with the waste category as follows:

Activated Metals

The Part 61 classification status of disposal liners containing activated metals can be determined by dividing the total activity of all the components by the total volume of the components in the package.

Process Wastes

In all cases, the Part 61 classification status of spent resins and filter cartridges is determined by dividing the total activity of the waste by the total volume of the waste.

Both spent resins and filter cartridges can also be stabilized. The Part 61 classification status of the stabilized waste can be determined by dividing the total activity by the total volume, including the stabilization medium.

Contaminated Solids

The packaged wastes are considered homogeneous, and the entire volume is used for classification determination.

Sealed Sources

Non-TRU-bearing sources can be encapsulated, and the encapsulation medium volume is included in the concentration calculation. Prior approval on encapsulation medium and method is required for sources containing TRU radionuclides.

7.5.2.3 Beatty Disposal Facility. The Beatty, Nevada, disposal facility received about 7% of the commercial low-level radioactive waste disposed of in 1989 (Appendix I). Its application of 10 CFR Part 61 requirements to concentration averaging is identical to that of the Richland facility, except that prior approval is not required for encapsulated TRU sources.

7.5.3 Prospective Averaging Practices

Beyond 1992, the concentration averaging practices will be based on the criteria in effect within the different compact regions and sited states. Therefore, disposal criteria may vary to a much greater extent than at present. On the one hand, the Richland site will continue to be the disposal site for the Northwest Compact and, in the absence of any new NRC criteria, current practices will remain essentially unchanged. On the other hand, the Appalachian Compact Region, hosted by Pennsylvania, intends to prohibit any form of concentration averaging. Some of the other compact regions/states that represent large volumes of LLW, like the Southeast, Midwest, and New York, have not yet stated their positions on concentration averaging. It is unlikely that regulatory positions will differ from those already defined and bounded by the Northwest Compact Region and the Appalachian Compact Region.

The "no concentration averaging" position that may be proposed by some states or compacts is the most restrictive. Applying that position to the different waste categories discussed above would lead to the following criteria:

Activated Metals

Each activated metal component would be classified under Part 61 as if it were the only component in the disposal liners. Radionuclide concentrations would be determined by dividing the component activity by the component volume.

Process Solids

Where resins and filter cartridges are stabilized, the volume of the stabilization medium would be excluded from classification determination.

Contaminated Solids

Commingling with other contaminated solids would be prohibited. Inclusion of the stabilization medium volume to classify incinerator ash would probably be unacceptable for classification determination.

Sealed Sources

Encapsulation of sealed sources in a stabilization medium would probably be acceptable. However, radionuclide concentrations would be determined by dividing the source activity by the source volume.

7.5.4 GTCC LLW Projection Scenarios

Establishing the limits of the volumes of GTCC LLW that could be generated requires making some assumptions concerning prospective disposal criteria. For the purposes of determining the impact of alternative concentration averaging criteria, the following bounding conditions were assumed:

Base-Volume Case

The criteria in effect at the Barnwell site represent the current practices for most waste (68% by volume in 1989) that could be classified as GTCC LLW. Those criteria were selected to represent the base-volume case. Use of the Rule of 10 is provided for activated metals; use of the stabilization medium volume is provided for resins and incinerator ash, commingling of contaminated solids, and encapsulation of non-TRU-bearing sources with activities up to 10 Ci.

Low-Volume Case

The criteria in effect at the Richland site are consistent with the NRC reading of 10 CFR Part 61, even though they represent current practices for some waste (25% by volume in 1989) that could be considered GTCC LLW. Those criteria were selected to represent the low-volume case. They provide for determining Class C status based on disposal liner activity and volume, with stabilization medium included and all sources encapsulated.

High-Volume Case

The prospective criteria for compacts, like those suggested by the Appalachian Compact Region as described above, represent a very restrictive reading of 10 CFR Part 61 requirements. Those criteria were selected to represent the high-volume case. They provide for no concentration averaging. Contaminated solid commingling, use of stabilization medium volumes, and encapsulation would be prohibited.

Table 7-7 summarizes cases and criteria as they apply to the four different categories of waste discussed.

Table 7-7. GTCC LLW volume projection scenarios

<u>Waste Category</u>	<u>Low Volume Case</u>	<u>Base Volume Case</u>	<u>High Volume Case</u>
1. Activated Metals	Activity in liner divided by waste volume in liner Stabilization media volume INCLUDED	Activity in liner divided by waste volume in liner Concentrations of components/types within a FACTOR of 10	Activity in component divided by component by individual volume in liner
2. Process Waste	Activity in liner divided by waste volume in liner Stabilization media volume INCLUDED for Resins and filters	Activity in liner divided by waste volume in liner Stabilization media volume INCLUDED for Resins NOT for filters	Activity in liner divided by waste volume in liner Stabilization media volume NOT INCLUDED
3. Contaminated Solids	Activity in liner divided by waste volume in liner Commingling with non-TRU solids	Activity in liner divided by waste volume in liner Commingling with lower activity TRU solids	TRU bearing solids activity divided by solids volume No commingling
4. Sealed Sources	Activity in liner divided by waste volume in liner Encapsulation media volume INCLUDED for ALL	Activity in source divided by source volume Encapsulation media volume INCLUDED for non-TRU sources <10Ci	Activity in source divided by source volume Encapsulation media volume EXCLUDED

7.6 Fuel Assembly Hardware

Fuel rod consolidation is a proposed process of separating and consolidating spent fuel rods from spent fuel assemblies to gain maximum use of the storage space available in a fuel storage pool at a nuclear power generation facility. The process generates waste from fuel disassembly hardware in the form of skeletal assemblies, end fittings, grid spacers, springs, screws, and miscellaneous small hardware. For purposes of this study, the waste generated through fuel rod consolidation, as decided by the technical review process, is not considered GTCC LLW. Followup contact with nuclear utilities after the EIA survey suggests that fuel rod consolidation is unlikely to occur at those facilities that indicated intended use of this process in the 1986 EIA survey. Waste generated through fuel rod consolidation, if any, would likely be considered high-level waste and would be disposed of under the Standard Contract (10 CFR 961).

7.7 Timing of Decommissioning Waste

The working life of a nuclear power reactor is assumed to be 40 years, using current NRC licensing practices. The assumption that decommissioning begins within five years following shutdown, and that the reactor vessel internals are available for disposal as GTCC LLW following the first three years after closure of the facility, are used in this study. Using that scenario, and assuming no new orders, the final closure date for all currently operating reactors, or those licensed to begin operation, will be the year 2035.

NRC is working on a proposed rule for life extension of nuclear power reactors, with a maximum extended life of approximately 20 years. The National Energy Policy, under development by DOE, assumes that 70% of all reactors will apply for some form of life extension. It is uncertain as to how many nuclear reactors will receive permits for life extensions, or what the exact length of the extended life will be, but the assumption here is that there will be 20 years of life extension for each of those facilities where life extension is granted.

The decommissioning of nuclear reactors, using this timing scenario, will make all waste available for disposal by the year 2035 in the low and base cases. The high case, assuming a maximum life extension of 20 years, will make all nuclear utility waste available for disposal by the year 2055.

7.8 Disposal Location of DOE-Held Potential GTCC LLW

A number of commercial facilities have generated GTCC LLW and, through contractual arrangements with DOE or for health and safety reasons, the waste is being stored at DOE facilities. The total packaged volume of this stored waste is approximately 1077 m^3 ($1.08\text{E}+03$). Not all the materials included in DOE-held potential GTCC LLW may require disposal in an NRC-licensed facility. A determination will be required on each case to decide whether disposal in an NRC-licensed facility or in a DOE facility is appropriate. Some of that waste was accepted for DOE research purposes and will not require disposal in an NRC-licensed facility. In the absence of a legal determination at this time, the total volume of this waste will be included in all scenarios for this study. At a future time, a determination will be made regarding the exact disposal location of each of these waste streams on a case-by-case basis.

7.9 Sealed Source Estimates^j

In 1989, the NRC conducted a survey to determine the number of potential sealed sources that may be available for disposal as GTCC LLW (Appendix D). Of the 2,202 specific licensees surveyed, 1,332 had 7,340 potential GTCC LLW sources or devices. Of the 1,332 specific licensees having GTCC LLW sources, 54 were having difficulty disposing of their sources or devices. These licensees have 715 sources in storage. Difficulties with disposal include the cost, the inability to find anyone to take the sources, or that the manufacturer of a source or device will not take the source back.

The bulk of the sealed sources identified in the NRC survey (6,728) are ^{241}Am , with approximately 18,741 Ci of activity. There are 122 ^{137}Cs sources with an approximate activity of 233,042 Ci. A total of 420 Ci of ^{238}Pu and 37 Ci of ^{239}Pu have been identified. There are approximately 2 Ci of ^{244}Cm .

Of the potential GTCC LLW sources or devices, 54.6% were portable gauges, mainly moisture gauges; the ^{241}Am activity totaled about 272 Ci (1.5% of the total ^{241}Am activity). Most of the radionuclide activity is from well logging sources. Well logging sources accounted for 24.2% of the total number of potential GTCC LLW sources or devices, 80.4% of the total ^{241}Am activity, and 30.4% of the total ^{238}Pu activity.

Based on findings of the 1989 NRC report on sealed sources (estimated projections from sample population to total licensee population), there are approximately 4,928 licensees with GTCC sealed sources and/or devices. It is estimated that each licensee has 5.51 sealed sources, yielding 27,156 GTCC sealed sources or devices.

The basis for projection of sealed source volumes assumes that all sources will be available for disposal. Timing of disposal of those sources was not identified in the NRC survey. A 90% confidence interval (Section 10.2) was used to determine the high and low limits for the NRC data, and the estimated data from the NRC report will be the base case.

The NRC estimates that there are 27,156 sealed sources or devices considered potential GTCC LLW. A 90% confidence interval for this total yields a range of GTCC LLW of sources from 24,900 to 30,800. NRC assumed that each source has an average volume of 0.603 cm^3 . Using those estimates, the total unpackaged volume for all GTCC LLW sealed sources or devices will range from $15,014\text{ cm}^3$ ($1.5\text{E-}2\text{ m}^3$) to $18,572\text{ cm}^3$ ($1.8\text{E-}2\text{ m}^3$), with a projected base case volume of $16,375\text{ cm}^3$ ($1.6\text{E-}2\text{ m}^3$).

7.10 Commercial Light Water Reactor GTCC LLW Generation^k

The generation of potential GTCC LLW from light water reactors (LWR) is plant type and fuel cycle dependent. All LWRs, both operating and planned for startup, are considered in this study, totaling 113 LWRs (36 BWRs and 77 PWRs).

j. Summary of a study by the Nuclear Regulatory Commission, *Above Class C Source/Device Inventory Survey*, Appendix H.

k. Summary of a study by Peter Tuite, *Greater-Than-Class C Low-Level Radioactive Waste Light Water Reactor Projections*, Appendix C.

Based on the results of a previous study (Appendix I), three categories of GTCC LLW were considered:

- Routinely generated activated metals
- Process wastes
- Decommissioning activated metals.

The results in Appendix G indicate that GTCC LLW volumes would be much lower than previously estimated by DOE; the evaluation in Appendix F confirms those results. Routinely generated activated metals and process wastes (in the form of cartridge filters) were found to represent a relatively small portion of the LLW routinely generated at LWRs. Estimated generation rates at BWRs range from a low of 0.1 m³ to a high of 0.2 m³ per plant per fuel cycle. Generation rates at PWRs varied with reactor vendor but were lower than those at BWRs. Estimated generation rates at PWRs range from a low of 0.07 m³ to a high of 0.1 m³ per plant per fuel cycle. Appendix F summarizes the generation rates at LWRs by waste type and reactor vendor. That study also presents estimates of the total volumes of GTCC LLW that may be generated through plant closure. Those cumulative volume estimates are based on the generation rates shown and the number of fuel cycles remaining through plant closure, assuming 40-year plant life cycles. Under the "no concentration averaging" volume scenario, the cumulative volumes are relatively small, amounting to about 170 m³.

All indications are that full reactor coolant system (RCS) decontamination projects, with fuel assemblies remaining in the vessel, will also result in GTCC LLW in the form of deep bed resins. Full RCS decontamination projects have never been performed and may not be until the mid-1990s. Initially, those operations will be performed with fuel out, at a frequency of one per plant every 10 years. After those decontamination processes have been fully demonstrated with fuel assemblies removed, nuclear power plants will consider full RCS decontaminations with fuel assemblies in the vessel. To project RCS decontamination resin volumes, it was assumed that plants with fewer than 20 years remaining through closure would not perform full RCS decontaminations with fuel assemblies in the vessel. This assumption eliminated 17 BWRs and 26 PWRs as potential generators of GTCC decontamination resin waste. It was assumed that the remaining 70 plants (19 BWRs and 51 PWRs) would perform one full RCS decontamination project, with fuel assemblies in the vessels, before shutdown. The estimated volume of decontamination resin GTCC LLW for a typical BWR was 2.83 m³, and for a PWR, 4.25 m³. These estimates led to cumulative volumes of about 270 m³ of full RCS decontamination resin for all LWRs through shutdown.

Decommissioning activated metals represent the largest source of GTCC LLW in terms of volume and activity. For this study a detailed analysis was performed for six different plants typical of LWRs. Two BWRs and four PWRs were analyzed in depth to estimate volumes and determine activities of GTCC LLW components. Appendix F summarizes the results of this analysis extended to all LWRs in terms of volumes. The cumulative volume of decommissioning activated metals for all plants considered is about 510 m³. Those values are conservative since only the larger plants of each LWR type were evaluated in detail.

The results for decommissioning activated metals were somewhat different from those reported in the standard NUREG reference reports (Reference 7). For BWR components, the results of this study indicate that, when calculated fluence are normalized to surveillance capsule data, the calculated integrated fluxes are about 40% lower than those previously reported, thereby resulting in lower activities. Thus, while the core shroud exceeds Class C limits, it does so by factors of less than two.

Those results would indicate that further investigations to better define BWR flux levels are warranted. Additionally, some of the older, smaller BWRs with long operating histories should be modeled and analyzed using surveillance capsule data to normalize activation fluxes.

For PWR components, the results of this evaluation confirm that the core baffle, or shroud, is well above Class C limits for all PWRs considered and also confirm that the thermal shield is not GTCC LLW. Mixed results were found for the core barrel, a major PWR component. For the two Westinghouse PWRs, the core barrel was found to barely exceed Class C limits. Since those plants were considered representative of 55 PWRs, further evaluations could eliminate the core barrel as GTCC LLW in this group of PWRs, provided there is no added activation from life extension.

Prior methodology employed for the analysis of decommissioning activated metals relied on NUREG 3474 (see Reference 8) for materials compositions. Those data have not been applicable to other activated metals components for some time, and actual materials heat data, or composites thereof, are usually used for other activation calculations. Where components are marginal GTCC LLW, like the BWR shroud and upper grid plate and certain PWR core barrels, bounding radionuclide content as a function of available materials heat data for the components of interest would reduce current uncertainties.

The results from the reactor models evaluated also demonstrate that plants should be modeled and analyzed individually because of variances in geometry and operating history, which can influence the results.

8. ACCIDENT SCENARIO AT A NUCLEAR FACILITY

It is difficult, if not impossible, to predict the likelihood of an accident at a nuclear facility. Attempting to make assumptions about an accident is beyond the scope and capability of this study. In fairness to the subject of accidents at nuclear facilities, a brief summary of the assumed GTCC LLW materials that have been identified as originating at Three Mile Island Unit 2 (TMI-2) is included in this section.

The sources of identified wastes from the TMI-2 accident are filters and resins used in the decontamination of the reactor coolant.⁸ Information exists as to the full extent of damage in the reactor core, however, no decision has been reached regarding actions to remove and dispose of the reactor internals and damaged fuel debris still in the reactor.¹

The filter and resin materials that produced GTCC LLW during the cleanup and decontamination efforts are listed below:

- **EPICOR Pre-Filters**

- Approximately 49.5 m³ of waste packaged for disposal
- Organic resins

- **Submersible Demineralizer System (SDS)**

- Approximately 7.5 m³ of packaged waste
- Zeolite resin filters

- **SDS Cuno Filters**

- Approximately 1.2 m³ of packaged waste
- Potential for mixed waste, lead material.

Filter and resin wastes were produced through processing of contaminated water found in the building and reactor pool. Concentrations of radionuclide activities were estimated to be approximately 6.00E+5 Ci for the waste that was produced from filters and liners. The total volume of packaged materials that are considered GTCC LLW is estimated at 58.2 m³. At decommissioning of TMI-2, a characterization of the reactor internals will be made and disposal requirements will be established.

The estimated volume of GTCC LLW from filters and resins will be included as potential accident waste. There are no volume estimates of potential GTCC LLW for the decommissioning of TMI-2, and because of the unique nature of the debris in the core, no attempt will be made to estimate future waste volumes.

The volumes given in this section include those components considered GTCC LLW. Volumes reported in Table 10-23 include all components shipped to DOE, several of which were accepted for research. Therefore, the volumes in Section 8 are smaller than those in Table 10-23.

1. Personal communication with Bill Conaway, General Public Utilities Three Mile Island, April 11, 1991.

9. MODELING APPLICATIONS

The base case scenario considers the most realistic data available and reflects current operating, decommissioning, and disposal practices for potential GTCC LLW. The high and low cases account for upper and lower limits of the base case data. The high case includes some components that were not recognized as GTCC LLW in the base case, but may be a potential waste consideration.

For each scenario, the volumes and radionuclide activities of the generator's current inventory and future generation rates are included in the projections data. The data are projected to a future date (2035 for low and base cases, and 2055 for high case), reported on an annual basis, and compiled to a five-year cycle. Waste volumes are projected for unpackaged, packaged, and concentration averaged categories; radionuclide activities are projected for unpackaged and packaged waste categories only. There are insufficient data to account for the effects of concentration averaging on radionuclide activity totals. One could assume an average activity per volume and multiply by the concentration averaged volume to get an average radionuclide concentration. However, this study has chosen to report activities rather than concentration levels.

The unpackaged waste volumes are multiplied by a component-specific packaging factor that will increase or decrease the volume of unpackaged waste. Packaged volume data are reported as a total for component-specific categories, which are included as either operations or decommissioning waste.

Concentration averaging factors are applied to the packaged waste volumes, resulting in the concentration averaged category. The volumes of that category of waste will change according to the concentration averaging scenario that is applied. In each case the data is reported on a component-specific level.

Total component volumes are calculated using existing waste inventories at each facility and adding an incremental annual volume for each component to that waste volume. This step is repeated on an annual cycle, resulting in a new volume of waste each year. Radionuclide concentrations, which contribute to the potential GTCC LLW classification of each component (for each nuclide that influences the component) are accumulated, along with each of these waste volumes, annually. Before additional waste is added to a new annual waste volume, the radionuclide concentrations are decayed for one year, using the Bateman equations to account for parent and daughter radionuclide behaviors. The Bateman equations assume that each parent nuclide decays as an exponential function of the decay constant and half-life for that nuclide; the daughter product increases as a result of the parent nuclide decay, but is assumed not to exceed the value of the parent at any point during the parent decay. The parent nuclide decay results in daughter products that are either stable isotopes or potential isotopes of GTCC LLW concern; the potential GTCC LLW daughter nuclides are included in the radionuclide inventory associated with that component and accumulated and decayed on an annual basis.

The modeled data are stored in a data base and controlled by an integrated data base management system. This system allows retrieval of modeled information by addressing specific data sorting requirements. For example, the data may be retrieved by any one, or combination of, the following requirements:

- Generator category
- Specific generator

- Component
- Facility type
- Activated nuclide
- Nuclide content.

Standard reporting of the modeled data is by waste type (operations or decommissioning), component, and volumes associated with a five-year cycle. Data can be retrieved through numerous other combinations of data requirements, such as yearly totals or by a specific waste type.

10. PROJECTIONS

Research and technical review of various uncertainties associated with GTCC LLW were used in this study to establish a guideline for assumptions applied in the GTCC LLW projection process. Recommendations from the technical review process that were applied include:

Fuel Consolidation

Assume that all waste generated as a result of fuel rod consolidation will be included in the high-level waste inventory, not as GTCC LLW.

Timing of Decommissioning Waste

Assume that 70% of all operating nuclear power plants will apply for licenses for life extension of the facilities. Those life extensions could be up to 20 years. It is assumed that all life extensions will be for 20 years and that data will apply to the projection of high case only.

Disposition of DOE-Held Potential GTCC LLW

Several commercial facilities have generated GTCC LLW and, through contractual arrangements, for health and safety reasons, DOE takes possession of those wastes and stores them at DOE facilities. The packaged volume of that waste is approximately 1077 m³, the unpackaged volume is not known. A determination will be required in each case to establish whether disposal in an NRC-licensed facility is necessary. For the purposes of this study, it will be assumed that all waste inventory will be disposed of in an NRC-licensed facility. However, it is unlikely that all waste will be subject to that disposal requirement.

Sealed Sources

Because of the lack of detailed information on sealed sources from the EIA survey, an alternate source of information was recommended. NRC has recently conducted a survey of GTCC sealed source users, and those results were recommended for this study.

Accidents at a Nuclear Facility

Little information is available to predict the waste forms that may be generated by an accident at a nuclear facility. It was recommended that this study use data from the TMI-2 accident. The wastes that were classified as GTCC LLW are referenced in Section 8.0.

Areas of uncertainty, where no recommendation was reached by the technical review process, include:

- Concentration averaging
- Waste packaging
- Activated metals/nonfuel-bearing components
- Operations waste from routine procedures.

Clarification of those issues with no recommendation from the technical review process was made by contracting private consultants recognized in the nuclear industry as having the recognized expertise. Results and application of that research are included in this study.

Each category of projections in this study includes component lists, total volumes of material associated with that component (in cubic meters), total radionuclide activities for each component

(in curies), and projections to 2035 for low and base cases and projections to 2055 for the high case. Data included are classified into four categories, which include all components described in Section 4. Those categories are:

- Nuclear Utilities
 - Operations waste
 - Decommissioning waste
- Sealed Sources
- DOE-Held Potential GTCC LLW
- Other Generators Waste (those generators not listed in the three categories above).

The projected data for the Other Generators Waste category rely on estimates made in the EIA survey to establish inventories as of 1986 and waste generation rates to closure of that facility. Those data are necessary to establish a baseline for future waste projections of those waste generators. Nuclear utility GTCC LLW generator data from the EIA survey do not provide adequate detail to estimate waste production rates, or 1986 inventory volumes. Wastes produced by those generators are disposed of on a regular basis (as described in Section 7.5), making it difficult to establish a baseline datum for initial inventory.^m

Existing LLW disposal sites will stop receiving waste from generators outside their State/compact boundaries at the end of 1992. All nuclear utilities will be required to dispose of their LLW at the designated disposal sites in their State/compact region. Those sites may not be open until after 1996, making onsite storage of LLW necessary at each nuclear power reactor site. To ensure adequate storage of future waste, the existing LLW inventory at operating nuclear utility reactors will be shipped for disposal before the end of 1992.ⁿ

All nuclear utility GTCC LLW that can be concentration averaged for disposal will be shipped to one of the existing disposal facilities before 1992. Waste generated before 1993 will not be easily quantified and cannot be efficiently estimated in the projections report. Data used to project GTCC LLW from nuclear utilities will come from estimates established by existing disposal practices. Average waste production rates (as described in Appendix F) are applied in the modeling process. Those averages will be applied to the total life of the reactor. However, it should be noted that a majority of the GTCC LLW produced before 1993 may be disposed of in an existing facility under current guidelines.

Projected GTCC LLW scenarios include unpackaged, packaged, and concentration averaged volumes. The radionuclide activity for each component is considered constant throughout each scenario. No assumptions are made to determine the effects of packaging or concentration averaging on individual component radionuclide activities.

m. Waste generation rates differ with each reactor design and with reactor operating procedures. Average waste generation rates were established for this study (see Section 10.1 and Appendix C).

n. Personal communication with Peter Tuite, WMG Inc., February 14, 1991.

The decommissioning waste radionuclide activities included in Table 10-8 are calculated using a computer code that predicts activation of components in an estimated flux field. The flux field was modeled and normalized to the observed data from surveillance capsules inside the reactor core (Appendix F). In most cases, the normalized flux data were less than the modeled flux (Items 4 and 5 on Table 10-8). The Westinghouse 1 reactor type normalized data increased. The operating practice for that reactor type generated a flux larger than necessary. Consequently, the operating practices have changed to reduce the flux field.

No data are available from surveillance capsules for the BWR 4 reactor type listed in Table 10-8. For that component, the modeled flux is used to calculate the radionuclide activity; for the other decommissioning components, the normalized flux data are used.

Unpackaged waste data depict the total displaced volume^o of each component. Packaged-waste data are based on the volume of the container in which each component is packaged for disposal. The concentration averaged data reflect the change in volume of GTCC LLW from application of that scenario to the packaged volume. All assumptions used in each case are outlined in Section 7.

10.1 Nuclear Utilities

Nuclear utility projections data are divided into PWR and BWR reactor types. BWR nuclear reactors are manufactured by General Electric Co., while PWR nuclear reactors are manufactured by three companies, Babcock & Wilcox Co., Combustion Engineering, and Westinghouse. Each manufacturer has unique reactor designs, with the potential for different waste component generation from each reactor type.

Two distinct BWR designs are listed, with the primary differences between the two designs being volumes of materials associated with the reactor core shroud (a reactor internal decommissioning component). PWRs are designed and built by the three different manufacturers listed above. One Babcock & Wilcox, one Combustion Engineering, and two Westinghouse designs are included in this study. Waste materials are unique to each reactor design; each design has different operations waste and decommissioning waste volumes. This study assumes that operations waste is generated through normal operating procedures, that the waste will become available for storage or disposal at the end of each refueling cycle, and that decommissioning waste will be available for disposal within three years after reactor closure. Nuclear power reactor refueling cycles, on average, are 20 months for BWRs and 19 months for PWRs. BWR and PWR components generated as waste are included in Tables 10-1 and 10-2, while component volumes and radionuclide concentrations are detailed in Appendices F and G.

Waste volumes and radionuclide activities for all components are scaled to annual generation rates and projected on a yearly cycle rather than projecting on fuel cycles. That assumption allows for a constant time scale for all GTCC LLW projections.

Table 10-3 summarizes the waste volumes used for projections of nuclear utility operations waste. Those data are broken down to be reactor specific.

o. Based on the principle of fluid displacement: A solid object placed in liquid displaces a volume of liquid proportional to its own volume.

The operations waste component list developed for this study came from existing disposal records using radionuclide concentrations, generation rates, and material volumes data. That waste is classified into two categories called activated metals and process wastes. Activated metals components include such things as control rod blades and incore instruments from BWRs, and primary sources, incore instruments, and thimble plug assemblies from PWRs. Process wastes include cartridge filters and decontamination resins from both BWRs and PWRs (see Tables 10-4 through 10-7). Data from those tables are used as a basis for estimating average production rates per reactor (based on reactor type). Those data are used to project volumes of GTCC LLW and radionuclide activities associated with that waste.

Several components included in that list are also listed in the HLW Standard Contract. It is important to note that those components are included in the EIA Survey and are replaced or removed, on occasion, separate from the final bundle or HLW component. Those components are included only to retain the integrity of the data base; no interpretation of the standard contract is intended.

Decommissioning wastes from nuclear reactors are generated at the end of the normal operating life of a reactor and are produced through one-time events. Each reactor design has the potential for differing volumes and radionuclide contents that may exceed Class C limits (see Tables 10-8 and 10-9). Independent research (Appendices F and G) suggests that BWR core shrouds may exceed Class C limits. Those studies also suggest that no other BWR reactor internals will exceed Class C limits. Data from studies detailing PWR reactor internals suggest that the core baffle may exceed Class C limits, in some cases, by a factor of 45. Core barrel data suggest that radionuclide concentrations are near Class C limits. However, uncertainty associated with modeling cannot predict if the core barrel will become GTCC LLW in each case; therefore, the core barrel will be included in the high case only. Data suggest that thermal shields do not activate to GTCC LLW and will not be included as GTCC LLW.

Material volumes for individual components, considered as GTCC LLW, were calculated from engineering drawings or recorded from final safety analysis documents. Radionuclide concentrations were calculated using ANISN and ORIGEN2 codes. Surveillance capsule data, taken from inside active reactor vessels, was used to normalize the modeled data to a known baseline standard (Appendix F). Adjustments to the modeled radionuclide activity data, compared to the surveillance capsule data, suggest a decrease of E+1 in magnitude of radionuclide activities for several components, while the remaining components show no change in magnitude between the two data sets. To ensure the most realistic data possible, the adjusted data were used in calculating the radionuclide content of each component.

It is assumed that nuclear reactor types, components (operations and decommissioning), and radionuclide activities listed are the current data for estimating projections data for the nuclear utilities category, and are used in the modeling process.

Table 10-1. BWR components generated as GTCC LLW

<u>BWR Component</u>	<u>Case</u>		
	<u>Low</u>	<u>Base</u>	<u>High</u>
BWR Operations			
Control rod blades	X	X	X
Local power range monitor	X	X	X
IRM/SRM dry tubes	X	X	X
Cartridge filters	X	X	X
Control rod drives	X	X	X
Fuel pool/vacuum	X	X	X
BWR Decommissioning			
Core shroud	X	X	X
Thermal shield	NA ^a	NA	NA

a. Not applicable--not included in this case.

Table 10-2. PWR components generated as GTCC LLW

PWR Component	Case		
	Low	Base	High
PWR Operations			
Babcock & Wilcox			
Incore instruments	X	X	X
Primary sources ^a	X	X	X
Crud tank filters	X	X	X
Combustion Engineering			
Instrument strings	X	X	X
Primary sources ^a	X	X	X
Westinghouse			
Thimble plug assemblies ^a	X	X	X
Thimble tubes ^a	X	X	X
Primary source rods ^a	X	X	X
Common to all PWRs			
Cartridge Filters			
Reactor coolant system	X	X	X
Seal water injection	X	X	X
Cavity drain	X	X	X
Spent fuel pool	X	X	X
PWR Decommissioning			
Babcock & Wilcox			
Core baffle	X	X	X
Core barrel	NA ^b	NA	X
Thermal shield	NA	NA	NA
Combustion Engineering			
Core baffle	X	X	X
Core barrel	NA	NA	X
Thermal shield	NA	NA	NA
Westinghouse			
Core baffle	X	X	X
Core barrel	NA	NA	X
Thermal shield	NA	NA	NA

a. Listed in Part 961 "Standard Contract for Disposal of Spent Fuel and/or High-Level Radioactive Waste," Nuclear Regulatory Commission. Included as data from the EIA Survey.

b. Not applicable--not included in this case.

Table 10-3. Light water reactor summary projections--routine GTCC LLW

<u>Routine Activated Metals</u>	<u>Number of Plants</u>	<u>Unit Volume (m³)</u>	<u>Units Per Cycle</u>	<u>Percent GTCC</u>	<u>GTCC Units/ Cycle</u>	<u>GTCC/ cycle (m³)</u>
1. BWR operations	36					
a. Control rod blades-control cell		0.017	15	25	4	0.060
b. Control rod blades-normal		0.017	10	10	1	0.017
c. LPRM strings		0.003	12	75	9	0.025
d. IRM/SRM dry tubes		0.002	2	100	2	0.004
Subtotal						0.115
2. Westinghouse PWR operations	55					
a. Thimble plug assemblies ^a		0.001	16	10	1.6	0.001
b. Thimble tubes ^a		0.002	2	100	2	0.003
c. Primary source rods ^a		0.0002	0.4	100	0.4	0.0001
Subtotal						0.004
3. Combustion Eng. PWR operations	15					
a. Instrument strings		0.002	24	50	12	0.024
b. Primary sources ^a		0.001	0.4	100	0.4	0.0002
Subtotal						0.0242
4. Babcock & Wilcox PWR operations	7					
a. In-core detectors		0.003	16	50	8	0.023
b. Primary sources ^a		0.0001		100	0.4	0.0001
Subtotal						0.023
<u>Routine Process Wastes</u>						
1. BWR cartridge filters	36					
a. Control rod drive		0.002	30	50	15	0.030
b. Fuel pool/vacuum		0.006	12	25	3	0.021
Subtotal						0.051
2. PWR cartridge filters	77					
a. Reactor coolant system		0.013	8	25	2	0.025
b. Seal water injection		0.003	18	25	4	0.011
c. Cavity drain		0.003	2	25	0.5	0.001
d. Spent fuel pool		0.024	4	25	1	0.024
Subtotal						0.061
3. B&W crud tank filter	7	0.027	0.5	100	0.5	0.013

a. Listed in Part 961 "Standard Contract for Disposal of Spent Fuel and/or High-Level Radioactive Waste," Nuclear Regulatory Commission. Included as data from the EIA Survey.

Table 10-4. PWR hardware components--activity estimates

PWR Vendor		Westinghouse		Combustion Engineering		Babcock & Wilcox	
1. Component type	Thimble ^c plugs	Thimble ^c tubes	Primary ^c source rods	Instrument strings	Primary ^c sources	Incore detectors	Primary ^c sources
2. Volume (m ³)	0.0007	0.0023	0.0002	0.0020	0.0006	0.0028	0.0001
3. Weight (kg)	5.45	15.64	1.59	15.45	4.54	22.72	0.45
4. Number of cycles ^a	5	7	5	3	5	3	5
5. Exposure - EFPD ^b	2000	2800	2000	1290	2150	1200	2000
6. Part 61 status							
- Table 1 fraction	1.04	3.33	300000	3.25	750000	0.822	550000
- Table 2 fraction	1.28	4.29	NEG	4.05	NEG	2.059	NEG
7. Activity (curies)							
C-14	7.80E-03	5.70E-02	NA	1.83E-02	NA	1.56E-02	NA
Mn-54	1.65E+00	8.40E+00	NA	2.08E-01	NA	1.89E-01	NA
Fe-55	8.85E+01	4.90E+02	NA	9.52E+00	NA	8.41E+00	NA
Co-60	5.45E+01	5.10E+02	NA	3.35E+02	NA	3.11E+02	NA
Ni-59	4.40E-02	4.20E-01	NA	3.26E-01	NA	3.33E-01	NA
Ni-63	5.30E+00	4.10E+01	NA	3.99E+01	NA	4.08E+01	NA
Nb-94	6.30E-05	5.80E-05	NA	1.34E-04	NA	1.24E-04	NA
TRU	NA ^d	NA	50(Pu-Be)	NA	340(Pu-Be)	NA	25(Am-Be)
Total curies	1.50E+02	1.06E+03	5.00E+01	3.85E+02	3.40E+02	3.61E+02	2.50E+01

a. Number of cycles component is in service.

b. Equivalent full-power days.

c. Listed in Part 961 "Standard Contract for Disposal of Spent Fuel and/or High-Level Radioactive Waste," Nuclear Regulatory Commission. Included as data from the EIA Survey.

d. Not applicable.

Table 10-5. BWR hardware components--activity estimates

<u>BWR Vendor</u>	<u>General Electric</u>			
1. Component type	<u>Control rod blade</u>	<u>Control rod blade (cc)^c</u>	<u>LPRM string</u>	<u>IRM/SRM dry tube</u>
2. Volume (m ³)	0.0170	0.0170	0.0028	0.0023
3. Weight (kg)	93.18	93.18	22.72	17.73
4. Number of cycles ^a	6	2	3	6
5. Exposure - EFPD ^b (390 EFPD/cycle)	2340	780	1170	2340
6. Part 61 status				
- Table 1 fraction	1.36	1.28	1.57	4.61
- Table 2 fraction	0.71	0.68	1.37	4.01
7. Activity (curies)				
H-3	2.20E+01	2.30E+01	NA	NA
C-14	1.20E-01	1.10E-01	3.40E-02	8.10E-02
Mn-54	2.00E+01	4.50E+01	8.80E+01	1.20E+01
Fe-55	1.21E+03	1.78E+03	5.30E+02	9.05E+02
Co-60	1.23E+03	1.49E+03	3.60E+02	7.15E+02
Ni-59	5.30E-01	5.50E-01	1.80E-01	3.90E-01
Ni-63	8.50E+01	8.20E+01	2.70E+01	6.40E+01
Nb-94	3.75E-03	3.50E-03	6.24E-04	1.50E-03
TRU	NA	NA	3.00E-05	NA
Total curies	2.58E+03	3.42E+03	9.30E+02	1.69E+03

a. Number of cycles component is in service.

b. Equivalent full-power days.

c. Control-Cell

Table 10-7. Decontamination resin activity

1. Resin type	PWR RCS fuel in		BWR RCS fuel in
2. Decon method	CAN DEREM ^a	LOMI ^b	LOMI ^b
3. Volume (m ³)	4.24	4.24	2.85
4. Weight (kg)	3409	3409	2272
5. Part 61 status			
- Table 1 fraction	4.68	1.56	5.84
- Table 2 fraction	NA	NA	NA
6. Activity (curies)			
C-14	3.28E+00	1.09E+00	1.00E-02
Mn-54	5.91E+01	1.97E+01	4.27E+00
Fe-55	5.88E+02	1.95E+02	4.51E+01
Co-58	4.10E+02	1.37E+02	1.51E+00
Fe-59	2.63E+01	8.74E+00	1.20E-01
Co-80	1.17E+03	3.90E+02	2.10E+02
Ni-63	7.22E+01	2.40E+01	5.53E+00
Zn-66	1.64E+01	5.46E+00	2.23E+00
Sr-90	6.57E+00	2.18E+00	2.00E-02
Nb-95	4.93E+01	1.64E+01	2.40E-01
Ca-134	6.57E+00	2.18E+00	6.00E-02
Ca-137	9.85E+00	3.28E+00	7.00E-02
Ce-141	2.96E+01	9.83E+00	4.00E-02
Ce-144	6.57E+01	2.18E+01	6.00E-01
Pu-241	3.30E+00	1.10E+00	2.92E+01
Cm-242	1.60E+00	5.00E-01	2.00E-02
TRU	1.50E+00	5.00E-01	4.90E-01
Total curies	2.52E+03	8.39E+02	3.00E+02
Activity $\mu\text{Ci/cc}$	5.94E+00	1.97E+00	1.06E+00

a. Proprietary decontamination method developed in Canada, using ion-exchange method.

b. Low oxidation metal ion.

Table 10-8. PWR decommissioning waste--activity estimates for Westinghouse, Babcock & Wilcox, and Combustion Engineering

<u>PWR Vendor</u>	<u>Westinghouse 1</u>		
1. Component type	<u>Core baffle</u>	<u>Core barrel</u>	<u>Thermal shield</u>
2. Volume (m ³)	1.27	2.63	3.11
3. Weight (kg)	10,136	28,045	24,773
4. Flux (n/cm ² -sec) (ANISN) ^a	1.83E+14	1.94E+12	3.49E+11
5. Normalized flux ^b (Surveillance capsule)	2.06E+14	2.91E+12	5.22E+11
6. Part 61 status			
- Table 1 fraction	45.41	1.12	0.20
- Table 2 fraction	37.36	0.94	0.16
7. Activity (curies)			
C-14	5.95E+02	3.14E+01	4.62E+00
Mn-54	3.09E+05	1.21E+04	1.94E+03
Fe-55	3.09E+06	1.69E+05	2.49E+04
Co-60	1.46E+06	1.18E+05	1.80E+04
Ni-59	1.47E+03	1.84E+02	2.81E+01
Ni-63	3.33E+05	2.33E+04	3.49E+03
Nb-94	8.74E+00	5.42E-01	8.53E-02
Total curies	5.19E+06	3.23E+05	4.84E+04

Table 10-8. (continued)

<u>PWR Vendor</u>	<u>Westinghouse 2</u>		
1. Component type	<u>Core baffle</u>	<u>Core barrel</u>	<u>Thermal shield</u>
2. Volume (m ³)	2.01	3.54	1.27
3. Weight (kg)	16,000	20,909	10,090
4. Flux (n/cm ² -sec) (ANISN) ^a	1.53E+14	7.90E+12	4.72E+12
5. Normalized flux (Surveillance capsule)	7.32E+13	3.98E+12	2.26E+12
6. Part 61 status			
- Table 1 fraction	21.26	1.45	0.69
- Table 2 fraction	16.00	1.23	0.36
7. Activity (curies)			
C-14	3.52E+02	4.14E+01	4.46E+00
Mn-54	2.00E+05	1.77E+04	4.08E+03
Fe-55	1.93E+06	2.33E+05	2.42E+04
Co-60	1.14E+06	1.56E+05	2.18E+04
Ni-59	1.40E+03	2.39E+02	2.66E+01
Ni-63	2.25E+05	3.04E+04	3.21E+03
Nb-94	6.39E+00	7.04E-01	1.40E-01
Total curies	3.50E+06	4.37E+05	5.3E+04

Table 10-8. (continued)

PWR Vendor	Babcock & Wilcox		
	Core <u>baffle</u>	Core <u>barrel</u>	Thermal <u>shield</u>
1. Component type			
2. Volume (m ³)	1.47	2.55	2.72
3. Weight (kg)	20,181	11,636	21,636
4. Flux (n/cm ² -sec) (ANISN) ^a	1.26E+13	1.20E+14	4.45E+12
5. Normalized flux (Surveillance capsule)	7.34E+12	7.02E+13	2.60E+12
6. Part 61 status			
- Table 1 fraction	21.50	2.90	0.97
- Table 2 fraction	16.89	2.60	0.77
7. Activity (curies)			
C-14	2.99E+02	6.41E+01	1.98E+01
Mn-54	1.55E+05	2.77E+04	1.09E+04
Fe-55	1.70E+06	3.75E+05	1.15E+05
Co-60	8.56E+05	2.32E+05	7.94E+04
Ni-59	1.03E+03	3.44E+02	1.17E+02
Ni-63	1.73E+05	4.64E+04	1.47E+04
Nb-94	4.65E+00	1.01E+00	3.71E-01
Total curies	2.89E+06	6.82E+05	2.20E+05

Table 10-8. (continued)

<u>PWR Vendor</u>	<u>Combustion Engineering</u>	
1. Component type	<u>Core baffle</u>	<u>Core barrel</u>
2. Volume (m ³)	2.21	8.21
3. Weight (kg)	20,181	64.796
4. Flux (n/cm ² -sec) (ANISN)	1.31E+14	1.23E+13
5. Normalized flux (Surveillance capsule)	--	--
6. Part 61 status		
- Table 1 fraction	33.01	4.59
- Table 2 fraction	26.43	3.91
7. Activity (curies)		
C-14	5.50E+02	3.19E+02
Mn-54	4.09E+05	1.41E+05
Fe-55	3.84E+06	1.81E+06
Co-60	1.96E+06	1.13E+06
Ni-59	2.16E+03	1.62E+03
Ni-63	4.09E+05	2.25E+05
Nb-94	1.12E+01	5.27E-01
Total curies	6.63E+06	3.30E+06

Table 10-9. BWR decommissioning--activity estimates for General Electric

<u>BWR Vendor</u>		<u>General Electric</u>	
1. Component type		BWR 4 <u>baffle</u>	BWR 6 <u>barrel</u>
2. Volume (m ³)		2.89	4.98
3. Weight (kg)		22,909	39,545
4. Flux (n/cm ² -sec) (ANISN) ^a		5.85E+12	5.57E+12
5. Normalized flux (Surveillance capsule)		--	3.55E+12
6. Part 61 status			
- Table 1 fraction		2.76	1.58
- Table 2 fraction		3.37	1.81
7. Activity (curies)			
C-14		9.23E+01	8.39E+01
Mn-54		1.28E+04	1.36E+04
Fe-55		5.45E+05	4.89E+05
Co-60		3.60E+05	3.57E+05
Ni-59		4.81E+02	4.69E+02
Ni-63		6.81E+04	6.32E+04
Nb-94		9.22E-01	9.33E-01
Total curies		9.87E+05	9.23E+05
a. Appendix F.			

10.1.1 Results

Tables 10-10 through 10-20 summarize the results of projections of nuclear utility GTCC LLW. Tables 10-10 and 10-11 are projected unpackaged waste volumes for all BWR and PWR facilities. Tables 10-12 and 10-13 list packaged volumes for nuclear reactors in low, base, and high cases. Tables 10-14 and 10-15 include concentration averaging data for all cases. Tables 10-16 through 10-20 list radionuclide contents for nuclear reactor wastes.

Table 10-10. GTCC LLW projection scenario--BWR unpackaged waste volumes (m³)

<u>Component</u>	<u>Low to 2035</u>	<u>Base to 2035</u>	<u>High to 2035</u>	<u>High to 2055</u>
Control rod blade	8.83E+01	8.83E+01	8.83E+01	1.20E+02
Local power range monitor	1.93E+01	1.93E+01	1.93E+01	2.63E+01
Dry tubes	4.36E+00	4.36E+00	4.36E+00	5.80E+00
FI decontamination resin	5.66E+01	5.66E+01	5.66E+01	5.66E+01
Pool filters	1.68E+01	1.68E+01	1.68E+01	2.27E+01
Control rod drive strainers (outer)	1.12E+01	1.12E+01	1.12E+01	1.54E+01
Control rod drive strainers (inner)	2.55E-01	2.55E-01	2.55E-01	3.45E-01
Cartridge filters ^a	5.80E-01	5.80E-01	5.80E-01	5.80E-01
Control rod bearings ^a	2.86E-02	2.86E-02	2.86E-02	2.86E-02
Poison curtains ^a	6.78E-03	6.78E-03	6.78E-03	6.78E-03
Core shroud	1.29E+02	1.29E+02	1.29E+02	1.29E+02
Total BWR volume	3.26E+02	3.26E+02	3.26E+02	3.77E+02

a. Reported as 1986 inventory in the EIA survey.

Table 10-11. GTCC LLW projection scenarios--PWR unpackaged waste volumes (m³)

<u>Component</u>	<u>Low to 2035</u>	<u>Base to 2035</u>	<u>High to 2035</u>	<u>High to 2055</u>
<u>Babcock & Wilcox</u>				
Incore detectors	3.90E+00	3.90E+00	3.90E+00	5.30E+00
Primary sources ^a	1.13E-02	1.13E-02	1.13E-02	1.13E-02
Fuel in decon resins	8.48E+00	8.48E+00	8.48E+00	8.48E+00
Cartridge filters	1.32E+01	1.32E+01	1.32E+01	1.80E+01
Crud tank filter	2.32E+00	2.32E+00	2.32E+00	3.15E+00
Control rod drive ^b	3.20E-02	3.20E-02	3.20E-02	3.20E-02
Flux wire ^b	4.00E-01	4.00E-01	4.00E-01	4.00E-01
Metals misc. ^b	3.80E-02	3.80E-02	3.80E-02	3.80E-02
Core shroud	1.03E+01	1.03E+01	1.03E+01	1.03E+01
Core barrel	--	--	1.53E+01	1.53E+01
Total	3.87E+01	3.87E+01	5.40E+01	6.10E+01
<u>Combustion Engineering^c</u>				
Instruments strings	9.17E+00	9.17E+00	9.17E+00	1.22E+01
Primary sources ^a	7.47E-02	7.47E-02	7.47E-02	7.47E-02
Fuel in decon resins	4.66E+01	4.66E+01	4.66E+01	4.66E+01
Cartridge filters	2.30E+01	2.30E+01	2.30E+01	3.12E+01
Thimble plug assemblies ^a	4.00E-02	4.00E-02	4.00E-02	4.00E-02
Control rod drive ^b	7.40E-01	7.40E-01	7.40E-01	7.40E-01

Table 10-11. (continued)

<u>Component</u>	<u>Low to 2035</u>	<u>Base to 2035</u>	<u>High to 2035</u>	<u>High to 2055</u>
<u>Combustion Engineering (continued)</u>				
Flux wire ^b	6.00E-02	6.00E-02	6.00E-02	6.00E-02
Metals misc. ^b	3.00E-01	3.00E-01	3.00E-01	3.00E-01
Core shroud	3.32E+01	3.32E+01	3.32E+01	3.32E+01
Core barrel	--	--	1.23E+02	1.23E+02
Total	1.13E+02	1.13E+02	2.36E+02	2.47E+02
<u>Westinghouse</u>				
Thimble plug assemblies ^a	3.89E+00	3.89E+00	3.89E+00	4.44E+00
Source rods ^a	1.15E+00	1.15E+00	1.15E+00	1.15E+00
Fuel in decon resins	1.61E+02	1.61E+02	1.61E+02	1.61E+02
Cartridge filters	8.50E+01	8.50E+01	8.50E+01	1.14E+02
Incore instruments	4.47E+00	4.47E+00	4.47E+00	5.94E+00
Control rod drive ^b	1.72E+01	1.72E+01	1.72E+01	1.72E+01
Metals misc. ^b	1.25E+00	1.25E+00	1.25E+00	1.25E+00
Core shroud	8.93E+01	8.93E+01	8.93E+01	8.93E+01
Core barrel	--	--	1.98E+02	1.98E+02
Total	3.63E+02	3.63E+02	5.61E+02	5.92E+02

Table 10-11. (continued)

<u>Component</u>	<u>Low to 2035</u>	<u>Base to 2035</u>	<u>High to 2035</u>	<u>High to 2055</u>
Total PWR Volume	5.15E+02	5.15E+02	8.51E+02	9.00E+02
Total Unpackaged Volume for All Reactors	8.41E+02	8.41E+02	1.18E+03	1.28E+03

a. Listed in Part 961 "Standard Contract for Disposal of Spent Fuel and/or High-Level Radioactive Waste," Nuclear Regulatory Commission. Included as data from the EIA Survey.

b. Reported as 1986 inventory in the EIA survey.

c. An inconel flow skirt will also be GTCC LLW; however, little information is available for activation levels. The total unpackaged volume is less than 2 m³, which is 0.76% of the total decommissioning volume.

Table 10-12. GTCC LLW projection scenarios--BWR packaged waste volumes (m³)

<u>Component</u>	<u>Low to 2035</u>	<u>Base to 2035</u>	<u>High to 2035</u>	<u>High to 2055</u>
Control rod blade	3.53E+02	4.41E+02	8.83E+02	1.20E+03
Local power range monitor	5.80E+01	9.67E+01	1.93E+02	2.63E+02
Dry tubes	1.31E+01	2.13E+01	4.36E+01	5.80E+01
FI decontamination resin	1.13E+01	5.66E+01	1.13E+02	1.13E+02
Pool filters	1.68E+00	3.36E+01	6.72E+01	9.08E+01
Control rod drive strainers (outer)	1.12E+00	2.22E+01	4.55E+01	6.16E+01
Control rod drive strainers (inner)	2.55E-02	5.09E-01	1.02E+00	1.38E+00
Cartridge filters	5.80E-02	1.16E+00	2.32E+00	2.32E+00
Control rod bearings ^a	1.42E-04	1.42E-04	1.42E-04	1.42E-04
Poison curtains ^a	6.78E-03	6.78E-03	6.78E-03	6.78E-03
Core shroud	1.80E+02	2.57E+02	3.86E+02	3.86E+02
Total BWR volume	6.18E+02	9.30E+02	1.73E+03	2.18E+03

a. Reported as 1986 inventory in the EIA survey.

Table 10-13. GTCC LLW projection scenarios--PWR packaged waste volumes (m³)

<u>Component</u>	<u>Low to 2035</u>	<u>Base to 2035</u>	<u>High to 2035</u>	<u>High to 2055</u>
<u>Babcock & Wilcox</u>				
In-core detectors	1.17E+01	1.95E+01	3.90E+01	5.30E+01
Primary sources ^a	1.13E-02	1.13E-02	1.13E-02	1.13E-02
Fuel in decon resins	1.70E+00	8.48E+00	1.70E+01	1.70E+01
Cartridge filters	1.32E+00	2.64E+01	5.29E+01	7.20E+01
Crud tank filter	2.32E-01	4.64E+00	9.28E+00	1.26E+01
Control rod drive ^b	3.20E-02	3.20E-02	3.20E-02	3.20E-02
Flux wire ^b	4.00E-01	4.00E-01	4.00E-01	4.00E-01
Metals misc. ^b	3.80E-02	3.80E-02	3.80E-02	3.80E-02
Core shroud	1.44E+01	2.06E+01	3.09E+01	3.09E+01
Core barrel	-- ^c	-- ^c	4.59E+01	4.59E+01
Total	2.98E+01	8.01E+01	1.95E+02	2.32E+02
<u>Combustion Engineering^d</u>				
Instruments strings	2.75E+01	4.59E+01	9.17E+01	1.22E+02
Primary sources ^a	7.47E-02	7.47E-02	7.47E-02	7.47E-02
Fuel in decon resins	9.34E+00	4.66E+01	9.33E+01	9.33E+01
Cartridge filters	2.30E+00	4.59E+01	9.19E+01	1.25E+02
Thimble plug assemblies ^a	4.00E-01	8.00E-01	1.20E+00	1.20E+00
Control rod drive ^b	7.40E-01	7.40E-01	7.40E-01	7.40E-01

Table 10-13. (continued)

<u>Component</u>	<u>Low to 2035</u>	<u>Base to 2035</u>	<u>High to 2035</u>	<u>High to 2055</u>
Flux wire ^b	6.00E-02	6.00E-02	6.00E-02	6.00E-02
Metals misc. ^b	3.00E-01	3.00E-01	3.00E-01	3.00E-01
Core shroud	4.63E+01	6.62E+01	9.93E+01	9.93E+01
Core barrel	-- ^c	-- ^c	3.69E+02	3.69E+02
Total	8.70E+01	2.07E+02	7.48E+02	8.11E+02

Westinghouse

Thimble plug assemblies ^a	3.89E+01	7.78E+01	1.17E+02	1.33E+02
Source rods ^a	1.15E+00	1.15E+00	1.15E+00	1.15E+00
Fuel in decon resins	3.24E+01	1.61E+02	3.22E+02	3.22E+02
Cartridge filters	8.50E+00	1.70E+02	3.34E+02	4.56E+02
Incore instruments	1.34E+01	2.15E+01	4.47E+01	5.94E+01
Control rod drive ^b	1.72E+01	1.72E+01	1.72E+01	1.72E+01
Metals misc. ^b	1.25E+00	1.25E+00	1.25E+00	1.25E+00
Core shroud	1.25E+02	1.79E+02	2.68E+02	2.68E+02
Core barrel	-- ^c	-- ^c	5.95E+02	5.95E+02
Total	2.38E+02	6.29E+02	1.70E+03	1.85E+03

Table 10-13. (continued)

<u>Component</u>	<u>Low to 2035</u>	<u>Base to 2035</u>	<u>High to 2035</u>	<u>High to 2055</u>
Total PWR Volume	3.55E+02	9.16E+02	2.64E+03	2.89E+03
Total Packaged Volume for All Reactors	9.73E+02	1.85E+03	4.37E+03	5.07E+03

a. Listed in Part 961 "Standard Contract for Disposal of Spent Fuel and/or High-Level Radioactive Waste," Nuclear Regulatory Commission. Included as data from the EIA Survey.

b. Reported as 1986 inventory in the EIA survey.

c. Not included in this case.

d. An inconel flow skirt will also be GTCC LLW; however, little information is available for activation levels. The total unpackaged volume is less than 2 m³, which is 0.76% of the total decommissioning volume.

Table 10-14. GTCC LLW projection scenarios--BWR concentration averaged waste volumes (m³)

<u>Component</u>	<u>Low to 2035</u>	<u>Base to 2035</u>	<u>High to 2035</u>	<u>High to 2055</u>
Control rod blade	0.00E+00	1.76E+02	8.83E+02	1.20E+03
Local power range monitor	0.00E+00	1.29E+01	1.93E+02	2.63E+02
Dry tubes	0.00E+00	2.13E+00	4.36E+01	5.80E+01
FI decontamination resin	0.00E+00	2.26E+01	1.13E+02	1.13E+02
Pool filters	0.00E+00	1.34E+01	6.72E+01	9.08E+01
Control rod drive strainers (outer)	0.00E+00	4.44E+00	4.55E+01	6.16E+01
Control rod drive strainers (inner)	0.00E+00	1.02E-01	1.02E+00	1.38E+00
Cartridge filters	0.00E+00	4.64E-01	2.32E+00	2.32E+00
Control rod bearings ^a	1.42E-04	1.42E-04	1.42E-04	1.42E-04
Poison curtains ^a	6.78E-03	6.78E-03	6.78E-03	6.78E-03
Core shroud	1.80E+02	2.57E+02	3.86E+02	3.86E+02
Total BWR volume	1.80E+02	4.89E+02	1.73E+03	2.18E+03

a. Reported as 1986 inventory in the EIA survey.

Table 10-15. GTCC LLW projection scenarios--PWR concentration averaged waste volumes (m³)

<u>Component</u>	<u>Low to 2035</u>	<u>Base to 2035</u>	<u>High to 2035</u>	<u>High to 2055</u>
<u>Babcock & Wilcox</u>				
In-core detectors	0.00E+00	2.60E+00	3.90E+01	5.30E+01
Primary sources ^a	1.13E-02	1.13E-02	1.13E-02	1.13E-02
Fuel in decon resins	0.00E+00	3.39E+00	1.70E+01	1.70E+01
Cartridge filters	0.00E+00	1.06E+01	5.29E+01	7.20E+01
Crud tank filter	0.00E+00	1.86E+00	9.28E+00	1.26E+01
Control rod drive ^b	3.20E-02	3.20E-02	3.20E-02	3.20E-02
Flux wire ^b	4.00E-01	4.00E-01	4.00E-01	4.00E-01
Metals misc. ^b	3.80E-02	3.80E-02	3.80E-02	3.80E-02
Core shroud	1.44E+01	2.06E+01	3.09E+01	3.09E+01
Core barrel	^c	^c	4.59E+01	4.59E+01
Total	1.49E+01	3.95E+01	1.95E+02	2.32E+02
<u>Combustion Engineering^d</u>				
Instruments strings	0.00E+00	6.11E+00	9.17E+01	1.22E+02
Primary sources ^a	7.47E-02	7.47E-02	7.47E-02	7.47E-02
Fuel in decon resins	0.00E+00	1.87E+01	9.33E+01	9.33E+01
Cartridge filters	0.00E+00	1.84E+01	9.19E+01	1.25E+02
Thimble plug assemblies ^b	0.00E+00	3.20E-01	1.20E+00	1.20E+00
Control rod drive ^b	7.40E-01	7.40E-01	7.40E-01	7.40E-01
Flux wire ^b	6.00E-02	6.00E-02	6.00E-02	6.00E-02

Table 10-15. (continued)

<u>Component</u>	<u>Low to 2035</u>	<u>Base to 2035</u>	<u>High to 2035</u>	<u>High to 2055</u>
<u>Combustion Engineering (continued)</u>				
Metals misc. ^b	3.00E-01	3.00E-01	3.00E-01	3.00E-01
Core shroud	4.63E+01	6.62E+01	9.93E+01	9.93E+01
Core barrel	^c	^c	3.69E+02	3.69E+02
Total	4.75E+01	1.11E+02	7.48E+02	8.11E+02
<u>Westinghouse</u>				
Thimble plug assemblies ^a	0.00E+00	3.11E+01	1.17E+02	1.33E+02
Source rods	1.15E+00	1.15E+00	1.15E+00	1.15E+00
Fuel in decon resins	0.00E+00	6.44E+01	3.22E+02	3.22E+02
Cartridge filters	0.00E+00	6.80E+01	3.34E+02	4.56E+02
Incore instruments	0.00E+00	2.87E+00	4.47E+01	5.94E+01
Control rod drive ^b	1.72E+01	1.72E+01	1.72E+01	1.72E+01
Metals misc. ^b	1.25E+00	1.25E+00	1.25E+00	1.25E+00
Core shroud	1.23E+02	1.79E+02	2.68E+02	2.68E+02
Core barrel	^c	^c	5.95E+02	5.95E+02
Total	1.43E+02	3.65E+02	1.70E+03	1.85E+03

Table 10-15. (continued)

<u>Component</u>	<u>Low to 2035</u>	<u>Base to 2035</u>	<u>High to 2035</u>	<u>High to 2055</u>
Total PWR Volume	2.05E+02	5.16E+02	2.64E+03	2.89E+03
Total Concentration Averaged Volume for All Reactors	3.85E+02	1.01E+03	4.37E+03	5.07E+03

a. Listed in Part 961 "Standard Contract for Disposal of Spent Fuel and/or High-Level Radioactive Waste," Nuclear Regulatory Commission. Included as data from the EIA Survey.

b. Reported as 1986 inventory in the EIA survey.

c. Not included in this case.

d. An inconel flow skirt will also be GTCC LLW; however, little information is available for activation levels. The total unpackaged volume is less than 2 m³, which is 0.76% of the total decommissioning volume.

Table 10-16. Radionuclide activity--cartridge filters (curies)

<u>Radio-nuclide</u>	<u>BWR Outer CRD Strainers</u>	<u>BWR Inner CRD Strainers</u>	<u>BWR Fuel Pool</u>	<u>BWR Operational</u>
Am-241	8.06E-04	7.92E-04	9.55E-04	3.35E-04
C-14	5.11E-04	5.11E-04	1.64E+00	1.89E-03
Cm-242 ^a	0.00E+00	0.00E+00	0.00E+00	^b
Co-58	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	1.86E+01	1.80E+01	4.04E+01	6.04E-01
Cs-137	3.43E-01	3.40E-01	1.16E+01	6.94E-01
Fe-55	1.31E+01	1.27E+01	9.59E+00	0.00E+00
H-3	3.06E-02	3.00E-02	5.72E-01	0.00E+00
Mn-54	2.19E-04	2.18E-04	9.50E-04	0.00E+00
Nb-94 ^a	0.00E+00	0.00E+00	0.00E+00	7.59E-04
Ni-59 ^a	0.00E+00	0.00E+00	0.00E+00	1.32E-01
Ni-63	0.00E+00	0.00E+00	1.34E+02	4.68E+00
Pu-238 ^a	0.00E+00	0.00E+00	0.00E+00	^b
Pu-241	4.91E-01	4.83E-01	5.82E-01	2.04E-01
Sr-90	8.11E-03	8.04E-03	7.55E-01	1.63E-01
Tc-99 ^a	0.00E+00	0.00E+00	0.00E+00	3.80E-03
TRU	2.02E-01	2.02E-01	5.21E-01	1.34E-01
Zn-65	0.00E+00	0.00E+00	2.38E-04	0.00E+00
Total curies	6.76E+01	6.85E+01	2.00E+02	6.62E+00

Table 10-16. (continued)

<u>Radionuclide</u>	<u>Combustion Engineering</u>	<u>Babcock & Wilcox</u>		
	<u>Operational</u>	<u>Operational</u>	<u>Crud Tank Filters</u>	<u>Miscellaneous</u>
Am-241	2.09E-03	5.64E-03	7.45E-04	7.54E-04
C-14	7.15E+00	1.56E-01	3.34E+00	3.34E+00
Co-58 ^a	7.62E-15	0.00E+00	2.38E-30	2.38E-30
Co-60	1.02E+01	1.02E-02	2.07E+00	2.07E+00
Cs-137	1.20E-01	2.74E-01	4.99E-02	4.99E-02
Fe-55 ^a	9.40E-01	0.00E+00	6.63E-02	6.63E-02
H-3	1.91E-01	8.12E-03	6.58E-02	6.58E-02
Mn-54 ^a	3.45E-05	0.00E+00	1.11E-08	1.11E-08
Ni-63	6.29E+01	2.86E+02	2.84E+01	2.84E+01
Pu-241	1.27E+00	3.43E+00	4.60E-01	4.60E-01
Sr-90	5.76E-02	1.36E+00	2.38E-02	2.38E-02
Tc-99	0.00E+00	1.73E+00	0.00E+00	0.00E+00
TRU ^a	4.20E-01	0.00E+00	1.96E-01	1.96E-01
Zn-65 ^a	1.17E-06	0.00E+00	4.24E-11	4.24E-11
Total curies	8.33E+01	2.93E+02	3.47E+01	3.47E+01

Table 10-16. (continued)

<u>Radionuclide</u>	<u>Westinghouse Operational</u>	<u>Westinghouse Miscellaneous</u>
Am-241	3.97E-06	7.84E-03
C-14	0.00E+00	2.59E+01
Cm-242	0.00E+00	0.00E+00
Co-58	0.00E+00	5.54E-04
Co-60	3.35E-02	4.58E+01
Cs-137	0.00E+00	4.38E-01
Fe-55	0.00E+00	6.82E-01
H-3	0.00E+00	7.26E-01
Mn-54	0.00E+00	9.09E-03
Ni-63	3.73E+00	2.28E+02
Pu-238	0.00E+00	0.00E+00
Pu-241	2.42E-03	4.78E+00
Sr-90	0.00E+00	2.11E-01
TRU	1.20E-04	1.52E+00
Zn-65	0.00E+00	1.52E-03
Total curies	3.77E+00	3.08E+02

a. Reported on EIA Survey.

b. Less than E-30.

Table 10-17. Radionuclide activity--decontamination resins (curies)

<u>Radio-nuclide</u>	<u>PWR B&W</u>	<u>PWR CE</u>	<u>PWR WEST</u>	<u>BWR RCS</u>
Am-241	1.44E-02	3.99E-02	1.43E-01	8.36E-01
C-14	2.29E+01	4.91E+01	1.77E+02	3.89E-02
Ce-141	3.19E-01	8.96E+00	5.08E+01	3.43E-02
Ce-144	2.72E-05	1.76E-01	2.89E+00	7.28E-03
Cm-242	4.55E-12	2.76E-05	1.57E-03	2.42E-06
Co-58	1.76E-24	5.64E-09	1.47E-05	7.82E-10
Co-60	5.95E+02	2.92E+03	1.15E+04	1.22E+03
Cs-134	6.34E-02	1.88E+00	1.08E+01	4.89E-02
Cs-137	4.34E+01	1.04E+02	3.72E+02	1.83E+00
Fe-55	2.61E+01	3.70E+02	1.82E+03	7.36E+01
Fe-59 ^a	0.00E+00	1.41E-16	2.43E-11	1.89E-16
Mn-54	9.09E-05	2.83E-01	4.05E+00	8.81E-02
Nb-95 ^a	0.00E+00	5.53E-21	2.07E-14	3.69E-20
Ni-63	4.40E+02	9.74E+02	3.49E+03	1.90E+02
Pu-238	8.70E-14	5.29E-07	3.00E-05	4.62E-08
Pu-241	8.76E+00	2.43E+01	8.72E+01	5.09E+02
Sr-90	2.83E+01	6.83E+01	2.44E+02	5.12E-02
TRU	1.05E+01	2.25E+01	8.10E+01	1.91E+01
Zn-65	5.23E-07	1.44E-02	3.09E-01	9.74E-03
Total curies	1.18E+03	4.54E+03	1.78E+04	2.02E+03

a. Less than E-30.

Table 10-18. Radionuclide activity--BWR components (curies)

<u>Radionuclide</u>	<u>Control Rod Blade</u>	<u>Local Power Range Monitor</u>	<u>Dry Tube</u>
Am-241	0.00E+00	2.74E-02	3.81E-02
C-14	1.64E+02	5.34E+01	1.26E+02
Cm-242 ^a	0.00E+00	^b	0.00E+00
Co-60	5.29E+04	1.22E+04	2.58E+04
Cs-137 ^a	0.00E+00	8.31E+03	2.52E+00
Fe-55	7.46E+03	2.17E+03	3.82E+03
H-3	5.16E+03	3.84E+00	1.04E-03
Mn-54	3.17E-01	8.91E-01	8.44E-02
Nb-94	5.24E+00	9.90E-01	2.31E+00
Ni-59	8.34E+02	3.03E+02	1.49E+03
Ni-63	9.55E+04	3.32E+04	7.71E+04
Pu-238 ^a	0.00E+00	^b	0.00E+00
Pu-241 ^a	0.00E+00	1.67E+01	1.10E+00
Sr-90 ^a	0.00E+00	7.35E+03	1.17E-01
Tc-99 ^a	0.00E+00	4.38E+00	3.60E-03
TRU	0.00E+00	2.37E+03	8.00E-02
Total curies	1.62E+05	6.65E+04	1.08E+05

Table 10-18. (continued)

<u>Radionuclide</u>	<u>Control Rod Roller Bearings</u>	<u>Poison Curtains</u>
C-14	1.08E-02	3.95E-01
Co-60	3.82E+00	1.43E-01
Fe-55	6.70E-05	1.47E-05
H-3	0.00E+00	4.66E-01
Nb-94	3.32E-04	3.77E-03
Ni-59	4.80E-02	1.77E+00
Ni-63	5.05E+00	1.52E+02
Mn-54	3.78E-19	1.39E-25
TRU	0.00E+00	0.00E+00
Total curies	8.93E+00	1.55E+02

a. Reported on EIA Survey.

b. Less than E-30.

Table 10-19. Radionuclide activity--PWR components (curies)

<u>Radionuclide</u>	<u>Westinghouse</u>		<u>Combustion Engineering</u>		<u>Babcock & Wilcox</u>	
	<u>Thimble Plug Assembly^a</u>	<u>Primary Source^a</u>	<u>Incore Instrument</u>	<u>Primary Source^a</u>	<u>Incore Instrument</u>	<u>Primary Source^a</u>
Am-241	0.00E+00		0.00E+00		0.00E+00	
C-14	2.17E+01		1.06E+01		3.78E+00	
Cm-242 ^b	0.00E+00		0.00E+00		0.00E+00	
Co-60	3.86E+03		5.32E+03		9.99E+02	
Cs-137 ^b	0.00E+00		0.00E+00		9.02E-04	
Fe-55	1.00E+03		1.61E+01		7.02E-01	
H-3 ^b	0.00E+00		0.00E+00		1.70E-03	
Mn-54	3.46E-01		3.50E-04		7.80E-08	
Nb-94	2.95E-01		7.82E-02		3.18E-02	
Ni-59	1.21E+02		1.90E+02		1.69E+02	
Ni-63	1.16E+04		1.84E+04		1.63E+04	
Pu-238 ^b	0.00E+00		0.00E+00		0.00E+00	
Pu-241 ^b	0.00E+00		0.00E+00		0.00E+00	
Sr-90 ^b	0.00E+00		0.00E+00		8.38E-04	
Tc-99 ^b	0.00E+00		0.00E+00		3.50E-04	
TRU	0.00E+00	6.73E+06	0.00E+00	9.26E+06	1.67E-02	1.21E+04
Total curies	1.66E+04	6.73E+06	2.39E+04	9.26E+06	1.75E+04	1.21E+04

Table 10-19. (continued)

<u>Radio-nuclide</u>	<u>Westinghouse</u>			<u>Combustion Engineering</u>		
	<u>Control Rod Drive</u>	<u>Incore Instruments</u>	<u>Metals Misc.</u>	<u>Control Rod Drive</u>	<u>Flux Wire</u>	<u>Metals Misc.</u>
C-14	6.06E+00	1.15E+02		4.27E+00		
Co-60	1.74E+02	3.40E+04		3.93E+02		
Fe-55	6.87E-04	5.10E+03		9.92E+00		
H-3	5.24E+00	0.00E+00		8.67E+01		
Mn-54	3.94E-21	7.13E-01		5.00E-08		
Nb-94	1.63E+00	1.26E-01		1.36E-01		
Ni-59	2.65E+02	8.64E+02		2.14E+01		
Ni-63	2.72E+04	8.19E+04		9.17E+02		
TRU	6.73E+06	0.00E+00				
Total curies	6.76E+06	1.22E+05	no data available	1.45E+03	no data available	no data available

Table 10-19. (continued)

<u>Radio- Nuclide</u>	<u>Babcock & Wilcox</u>		
	<u>Control Rod Drive</u>	<u>Flux Wire^b</u>	<u>Metals Misc.</u>
C-14	1.17E-01	0.00E+00	
Co-60	2.94E-01	3.61E+00	
Cs-137	2.27E-03	0.00E+00	
Fe-55	0.00E+00	0.00E+00	
H-3	4.28E-03	0.00E+00	
Mn-54 ^c	0.00E+00	0.00E+00	
Nb-94	5.89E-03	0.00E+00	
Ni-59	5.29E+00	1.52E+02	
Ni-63	6.08E+02	1.53E+04	
Si-90	2.11E-03	0.00E+00	
Tc-99	9.00E-04	0.00E+00	
TRU	0.00E+00	3.34E-02	
Total curies	6.14E+02	1.55E+04	no data available

a. Listed in Part 961 "Standard Contract for Disposal of Spent Fuel and/or High-Level Radioactive Waste," Nuclear Regulatory Commission. Included as data from the EIA Survey.

b. Reported on EIA Survey.

c. Less than E-30.

Table 10-20. Radionuclide activity--decommissioning components (curies)

Radionuclide	PWR									BWR
	Westinghouse			Combustion Engineering			Babcock & Wilcox			General Electric
	Core Baffle	Core Barrel	Core Baffle	Core Baffle	Core Barrel	Core Barrel	Core Baffle	Core Barrel	Core Shroud	
C-14	2.53E+04	2.05E+03	8.91E+03	8.91E+03	4.70E+02	2.39E+03	3.84E+02	3.52E+03		
Co-60	1.19E+07	1.65E+06	3.56E+06	2.89E+05	1.03E+05	1.89E+06				
Fe-55	6.48E+06	9.03E+05	1.90E+06	9.99E+04	8.31E+04	7.26E+05				
Mn-54	1.40E+04	4.26E+03	1.48E+03	3.61E+01	4.16E-02	2.17E+02				
Ni-59	7.74E+04	1.20E+04	2.20E+04	2.76E+03	2.06E+03	1.86E+04				
Ni-63	5.90E+06	1.37E+06	4.48E+04	3.14E+05	2.43E+05	2.29E+06				
Nb-94	4.06E+02	3.52E+01	1.31E+02	8.13E+00	6.06E+00	3.60E+01				
Total curies	2.44E+07	3.94E+06	5.54E+06	7.06E+05	1.78E+06	4.93E+06				

10.2 Sealed Sources

There are many types of sealed sources/devices in use. Classifications of those materials (with GTCC LLW potential) are listed in Appendix D. The data suggest that there are various GTCC LLW generators with little chance of disposing of those materials that are, or may become, waste. An uncertainty analysis of the data provided for that study is used to develop high and low cases for the projections model. Uncertainty is not assumed for the radionuclide activity; those data reported in the NRC study are used in all cases. No detail is available for timing of sealed source GTCC LLW generation; therefore, no time scale was developed for those wastes.

Methods used to compute the confidence intervals for GTCC LLW sealed sources are as follows. The proportion of licensees with sealed sources are based on a binomial distribution. This is a distribution for data that can take two values; in this case the two values are either the licensee possesses, or does not possess GTCC sealed sources. The confidence interval ($\hat{p}$) is constructed by first approximating the variance of the estimated proportion of licensees possessing GTCC sealed sources. The variance of $\hat{p}$ is given by

$$VAR(\hat{p}) = \frac{\hat{p}(1 - \hat{p})(N - n)}{(n-1)N} . \quad (1)$$

An approximate 95% confidence interval on the proportion of licensees possessing GTCC sealed sources is given by

$$\hat{p} \pm 2\sqrt{VAR(\hat{p})} . \quad (2)$$

Thus, the NRC estimates from its survey that 21% ($\hat{p} = 0.21$) of the licensees have GTCC sealed sources. NRC sampled 2,202 licensees (n) from a population of approximately 24,000 specific licensees (N). From those data, variance ($\hat{p}$) = 6.85 E-05, yielding a confidence interval on $\hat{p}$ of between 0.193 and 0.227.

The number of GTCC sealed sources is assumed to be a Poisson distribution, which is often used to characterize discrete counting phenomena. The variance of a Poisson random variable equals its mean, with the upper and lower 95% confidence limits for the mean given by

$$\theta_b = \frac{1}{2} \chi^2_{2x_1} \delta/2 \quad (3)$$

$$\theta_u = \frac{1}{2} \chi^2_{2(x+1)_1} 1 - \delta/2 \quad (4)$$

where

x = is the number of observed GTCC sealed sources in the sample, and

$\chi^2_{u,b}$ = is the value of a chi-square random variable with u degrees of freedom corresponding to probability level b .

The NRC study states that 7,340 GTCC sealed sources were identified in the survey. From this the NRC estimates 27,156 total sources, or about 5.57 sources per licensee. The initial confidence interval is placed on the number of observed sealed sources and then extrapolated to the total population. The lower and upper 95% confidence limits are 7,173 and 7,510 respectively. Those numbers correspond to 5.385 and 5.638 GTCC sealed sources per licensee.

Since the lower 95% confidence limit on the proportion of licensees identified as having GTCC sealed sources is 0.193, and the upper limit is 0.227, an approximate number of sources projected over the population is

$$\begin{aligned}\text{Lower:} & 24,000 \times 5.385 \times 0.193 = 24,943 \\ \text{Upper:} & 24,000 \times 5.638 \times 0.227 = 30,716\end{aligned}$$

Assuming, as NRC has, that each source has an average volume of 0.603 cm^3 , the confidence interval volume is 0.015 m^3 to 0.018 m^3 , with a base case volume of 0.016 m^3 .

Data included in the NRC study (Appendix D) are assumed to be current information for sealed sources. Those data are used in estimating volumes and radionuclide activities for all sealed source projections, and are used in all cases based on an uncertainty analysis for determining upper and lower limits. Volumes determined from that analysis are included in Table 10-21. Those data show unpackaged and packaged waste volumes, and radionuclide activities. NRC suggests that many general licensees may have GTCC LLW sources, but little information is available about those source holders at this time. Therefore, this study considers specific licensees only.

10.3 DOE-Held Potential GTCC LLW Waste

DOE is currently storing radioactive waste (which may be GTCC LLW) generated by NRC and Agreement State licensees. Those wastes are either subject to disposal in either an NRC-licensed facility or a DOE facility. No determination has been made for disposal at this time because ownership of those wastes is unclear. For the purposes of this study, the volumes listed in Tables 10-22 and 10-23 will be included in the low, base, and high case scenarios, along with the radionuclide activities listed.

10.4 Other Generators Waste

Seven generators reported GTCC LLW in the EIA survey that were not sealed sources or nuclear industry waste. Those generators are classified as Other Generators for this study and are listed in Table 10-27. Waste components and volume data received from the EIA survey are listed in Tables 10-24 through 10-26. Those data reflect volumes reported in the EIA survey and updates reported through telephone contacts with each generator.

10.5 Projections Summary

Projected volumes and radionuclide activities for waste components identified with each generator are listed in Tables 10-28 and 10-29. All data are category specific.

Two waste components, mixed oxide fuel pellets and organic liquids, are included in this section and were reported as inventory in the EIA survey. The organic liquids are considered mixed waste (the only mixed waste recognized in this study) and are routinely generated. The mixed oxide fuel

pellets were generated as a one-time event. No future generation of those wastes was predicted by the generator.

This study assumes that data provided from the EIA survey, and followup contacts with each generator, are current information for these waste generators. Components, estimated volumes, and radionuclide activities are used to establish an "on hand" inventory for 1986, and generator estimates for waste generation rates. Those data are used to predict future waste volumes for each generator and individual waste components. No other data were recognized to provide better information.

Table 10-21. Sealed sources volumes and radionuclide activities

Unpackaged Volume (m³)

<u>Low</u>	<u>Base</u>	<u>High</u>
1.50E-02	1.60E-02	1.8E-2

Packaged Volume (m³)

<u>Low</u>	<u>Base</u>	<u>High</u>
3.00E-2	6.0E+00	1.80+01

Radionuclide Activities (curies)

Am-241	1.92E+04
Cs-137	2.81E+05
Cm-244	2.00E+03
Pu-238	1.97E+04
Pu-239	6.77E+02
Sr-90	1.00E-02
Total	3.03E+05

Table 10-22. DOE-held GTCC LLW

Total Volumes: 1.08E+03m³

Total Radionuclide Concentration: 5.38E+05 Ci

Major Radionuclides: Am, Pu, Sr, Be, Cs

Table 10-23. GTCC LLW from NRC or Agreement State licensed facilities (Currently held at U.S. Department of Energy sites)

Field Office	Generator/User	Estimated Quantity				Major Isotope	Form Description	Status	Comments
		Act (Ci)	Vol. (m ³)	Vol. (ft ³)	Mass (g)				
AL	None	0.00	0.00	0.00	0.00	None	None	None	None
CH	None	0.00	0.00	0.00	0.00	None	None	None	None
ID	J. C. Haynes	25.40	6.12	216.00	4	Am	5 drums in 2 boxes	Stored at ILTSF	Health & safety
88	Monsanto (MO)	453.37	5.67	200.21	50.86	Pu,Am	27 55-gal drums	Stored at ILTSF	Waste contract
		6.59	13.59	480.00	1.95	Pu,Am	4 bins	Stored at ILTSF	
OR	Babcock & Wilcox (VA)	46.30	3.99	141.00	487.92	Pu,Am	19 55-gal drums	Stored at ILTSF	Service contract
		43.34	1.20	42.30	105.90	MFP,Pu	3 casks	Stored at TAN	Waste contract
OR	NFS (IN) - forecasted	N/A	156.00	5500.00	N/A	Pu,Am	750 55-gal drums	Stored at NFS	Waste contract
RL	Sequoyah fuels (OK)	N/R	256.41	9054.03	9230.00	Pu	1221 55-gal drums	Retrievable storage	Waste contract
	GE Vallecitos (CA)	N/R	87.51	3090.00	0.80	Pu	8 boxes	Retrievable storage	Service contract
		60000.00	4.16	147.00	593.00	Pu,MFP	6 casks	Retrievable storage	
	Babcock & Wilcox (PA)	N/R	11.89	420.00	31.00	Pu	2 boxes	Retrievable storage	Purchase order
		N/R	63.84	2254.24	237.00	Pu	304 55-gal drums	Retrievable storage	
	Westinghouse (PA)	N/R	78.54	2773.31	528.00	Pu	374 55-gal drums	Retrievable storage	Waste contract
		N/R	173.60	6130.00	8.00	Pu	27 boxes	Retrievable storage	
	Rockwell-Canoga (CA)	N/R	0.21	7.42	0.25	Ra,Be	1 source in 55-gal drum	Retrievable storage	Service contract

Table 10-23. (continued)

<u>Field Office</u>	<u>Generator/User</u>	<u>Estimated Quantity</u>				<u>Major Isotope</u>	<u>Form Description</u>	<u>Status</u>	<u>Comments</u>
		<u>Act (Ci)</u>	<u>Vol. (m³)</u>	<u>Vol. (ft³)</u>	<u>Mass (g)</u>				
		N/R	40.95	1445.97	556.00	Pu	195 55-gal drums	Retrievable storage	
	TMI (GPU Nuc Corp)	477700.00	161.42	5700.00	N/R	Cs, Sr	14 casks	Disposed in trench	MOU
	Battelle Columbus	N/R	0.21	7.42	2.50	Ra,Be	1 source in 55-gal drum	Retrievable storage	Service contract
SAN	None	0.00	0.00	0.00	0.00	None	None	None	None
SR	Allied General	N/R	10.92	385.59	57.00	Pu	52 55-gal drums	Retrievable storage	DOE ltr (waste inseparable from other AGNS wastereceived via nine service contracts)
8	Nuclear Services								
TOTALS		538275.00	1076.26	37994.49	11898.02				

Notes:

Activity: Total activity of nuclear material in the waste
N/R: Not reported
Volume: Package volume
GCD: Greater Confinement Disposal
Mass: Total mass of nuclear material in the waste
ILTSF: Intermediate-Level Temporary Storage Facility
Major Isotopes: Isotopes that contribute most to waste's activity
TAN: Test Area North

Significant figures are furnished to provide as much information as possible and should not be taken to imply a high degree of precision.

Table 10-24. Other Generator waste, 1986 GTCC LLW inventory

<u>Component</u>	<u>Volume (m³)</u>
Metal shavings	5.00E+00
Cartridge filters	9.00E+00
Compactible trash	3.18E+01
Contaminated equipment	1.50E+00
Ion-exchange resins	5.00E+00
Filter media	1.50E+00
Mixed oxide fuel pellets	4.15E+00
Noncompactible trash	3.75E+00
Organic liquids ^a	1.38E+00
Solidified resins	2.00E+00
Metals from operations	3.40E-01
Reactor components	1.00E-02
Total	6.54E+01

a. Considered mixed waste.

Table 10-25. Other Generator waste, GTCC LLW packaged volumes projected to 2035

<u>Component</u>	<u>Low</u>	<u>Base</u>	<u>Volume (m³)</u> <u>High</u>
Metal shavings	5.00E+00	5.00E+00	5.00E+00
Cartridge filters	2.30E+00	4.60E+01	9.20E+01
Compactible trash	1.42E+02	1.42E+02	1.42E+02
Contaminated equipment	2.33E+01	2.33E+01	2.33E+01
Ion-exchange resins	1.00E+00	5.00E+00	1.00E+01
Filter media	9.30E+00	9.30E+00	9.30E+00
Mixed oxide fuel pellets	4.15E+00	4.15E+00	4.15E+00
Noncompactible trash	1.03E+01	1.03E+01	1.03E+01
Organic liquids ^a	6.22E+00	6.22E+00	6.22E+00
Solidified resins	1.60E+01	1.60E+01	1.60E+01
Metals from operations	2.95E+00	2.95E+00	2.95E+00
Total	2.23E+02	2.70E+02	3.21E+02

a. Considered mixed waste.

Table 10-26. Other Generator Waste, GTCC LLW components by generator

<u>Generator Types</u>	<u>GTCC LLW Components</u>
Carbon 14 users	Organic liquids ^a
Fuel fabricators	Metal shavings Cartridge filters Compactible trash Ion-exchange resins Mixed oxide fuel pellets
Industrial research and development	Mixed oxide fuel Pellets
Manufacturers of Devices with sealed sources	Cartridge filters Compactible trash Solidified resins
Nuclear research/test reactors	Reactor internal parts Control rod blade parts
Sealed source distributors	Organic liquids ^a Metal parts
Sealed source manufacturers	Compactible trash Contaminated equipment Filter media Noncompactible trash

a. Considered mixed waste.

Table 10-27. Other Generator waste, total curies

<u>Nuclide</u>	<u>Metal Shavings</u>	<u>Cartridge Filters</u>	<u>Compactible Trash</u>			<u>Contaminated Equipment</u>	<u>Ion-exchange Resins</u>	<u>Filter Media</u>	<u>Mixed Oxide Fuel Pellets</u>
			<u>Decommissioning</u>	<u>Operations</u>	<u>Trash</u>				
Am-241	9.23E-01	9.51E-01	7.37E+02	1.53E+02		2.06E+02	9.22E-01	3.25E+01	3.07E+01
C-14	9.94E-01	0.00E+00	0.00E+00	0.00E+00		0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cm-242	0.00E+00	0.00E+00	a	a		a	0.00E+00	0.00E+00	0.00E+00
Co-60	1.25E-02	3.06E-02	0.00E+00	1.39E-06		0.00E+00	7.33E-03	0.00E+00	0.00E+00
Cs-137	1.27E+00	1.12E+01	0.00E+00	6.85E+00		0.00E+00	1.24E+00	0.00E+00	0.00E+00
I-129	1.00E+00	1.00E+00	0.00E+00	0.00E+00		0.00E+00	1.00E+00	0.00E+00	0.00E+00
Ni-59	1.00E+00	1.00E+00	0.00E+00	0.00E+00		0.00E+00	1.00E+00	0.00E+00	0.00E+00
Ni-63	7.07E-01	7.07E-01	0.00E+00	0.00E+00		0.00E+00	7.02E-01	0.00E+00	0.00E+00
Pu-238	0.00E+00	0.00E+00	a	a		a	0.00E+00	0.00E+00	9.42E+00
Pu-239	0.00E+00	0.00E+00	0.00E+00	0.00E+00		0.00E+00	0.00E+00	0.00E+00	2.05E+02
Pu-240	0.00E+00	0.00E+00	0.00E+00	0.00E+00		0.00E+00	0.00E+00	0.00E+00	4.92E+00
Pu-241	1.34E-01	8.94E-02	1.36E+02	7.61E+00		3.57E+01	1.28E-01	0.00E+00	5.35E+01
Sr-90	2.98E-01	8.93E-01	0.00E+00	2.98E-04		a	2.91E-01	0.00E+00	0.00E+00
Tc-99	1.00E+00	1.00E+00	0.00E+00	0.00E+00		0.00E+00	0.00E+00	0.00E+00	0.00E+00
TRU	1.00E+00	4.00E+00	8.00E+02	1.32E+01		2.00E+02	1.50E-01	0.00E+00	3.88E+01
Total	8.34E+00	2.09E+01	1.67E+03	1.31E+02		4.42E+02	5.31E+00	1.23E+01	1.42E+02

a. Less than E-30.

Table 10-27. (continued)

<u>Nuclide</u>	<u>Noncompactible Trash</u>	<u>Organic Liquids</u>	<u>Solidified Resins</u>	<u>Metal from Operations</u>	<u>Reactor Components</u>
Am-241	5.11E+01	6.49E+00	2.77E-02	3.25E+01	0.00E+00
C-14a	0.00E+00	1.19E+01	0.00E+00	0.00E+00	0.00E+00
C-14n	0.00E+00	1.02E+01	0.00E+00	0.00E+00	0.00E+00
Cm-242	a	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	0.00E+00	0.00E+00	1.39E-06	0.00E+00	0.00E+00
Cs-137	0.00E+00	0.00E+00	1.71E+00	0.00E+00	0.00E+00
Ni-59	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.41E+00
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.27E+01
Pu-238	a	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pu-241	3.57E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	0.00E+00	1.56E-01	2.98E-04	3.24E-01	0.00E+00
TRU	2.00E+01	4.90E-03	1.00E-03	4.90E-03	0.00E+00
Total	7.47E+01	2.88E+01	1.74E+00	3.28E+01	8.41E+01

a. Less than E-30.

Table 10-28. Summation data--packaged volumes

<u>Generator</u>	Volume (m ³)			
	<u>Low 2035</u>	<u>Base 2035</u>	<u>High 2035</u>	<u>High 2055</u>
<u>Nuclear Utility</u>				
Operations	6.07E+02	1.33E+03	2.58E+03	3.28E+03
Decommissioning	3.66E+02	5.23E+02	1.79E+03	1.79E+03
<u>Sealed sources</u>	3.00E-02	6.08E+00	1.80E+01	1.80E+01
<u>DOE-Held Potential GTCC LLW</u>	1.08E+03	1.08E+03	1.08E+03	1.08E+03
<u>Other Generators Waste</u>	2.23E+02	3.07E+02	3.58E+02	3.58E+02
Total	2.28E+03	3.25E+03	5.83E+03	6.53E+03
<u>Nuclear Utility Waste Concentration Averaged (not included in total)</u>				
Operations	2.28E+01	4.81E+02	2.58E+03	3.28E+03
Decommissioning	3.66E+02	5.23E+02	1.79E+03	1.79E+03

Table 10-29. Summation data of all tables--radionuclide activity

<u>Generator</u>	<u>Radionuclide Activity (Ci)</u> <u>(Based on Unpackaged Waste Volumes)</u>
<u>Nuclear Utility</u>	
Operations	2.33E+07
Decommissioning	4.17E+07
Sealed sources	3.03E+05
DOE-Held Potential GTCC LLW	5.38E+05
Other Generators Waste	2.92E+03
Total Activity	6.58E+07

11. DATA ANALYSIS

Data presented in this study are based on the various assumptions outlined in Sections 7 and 10. The available disposal and technical review data have been used in estimating GTCC LLW volumes and radionuclide activities. The base case data presented in this study are considered representative of realistic estimates of GTCC LLW characteristics projections. Representation of the projected volumes of GTCC LLW for nuclear utility waste and the other generators categories are presented in graphical form as Figures 11-1 through 11-43. Because of the detail and ranges of data presented in this study (cases vs. scenarios), a series of figures are presented to facilitate interpretation of the results of the data modeling and projections efforts.

The data are presented in groups of three figures, with each group labeled by the GTCC LLW component of interest. The groups are arranged in columns, with one column (one component) per page. Each figure represents a specific scenario (unpacked, packaged, or concentration averaged) with the different data cases (low, base, high) included in each figure. That arrangement allows a quick scan analysis of all the projected data sets. A brief summary of those projected estimates follows.

11.1 Nuclear Utility Wastes

The largest base case packaged volume for any category of GTCC LLW reported in this study is from nuclear reactors. For PWRs (base and low cases) there are approximately $1.33\text{E}+02 \text{ m}^3$ of unpackaged decommissioning waste metals from the core shrouds. For the high case, which includes the core shroud and lower core barrel, there are approximately $4.69\text{E}+02 \text{ m}^3$ of unpackaged decommissioning waste. BWR unpackaged decommissioning wastes for the high, base, and low cases are approximately $1.29\text{E}+02 \text{ m}^3$.

Operations waste for BWRs and PWRs comes in the form of process wastes (filters, resins, etc.), and activated metals (in-core instruments, control rod blades, and primary sources). Unpackaged operations waste from nuclear reactors totals $5.79\text{E}+02 \text{ m}^3$. Fuel-in decontamination resins account for approximately $2.73\text{E}+02 \text{ m}^3$ of those wastes. It is unclear if all of those resins will be GTCC LLW because of various operating practices, but all projected resin volumes are included in the total volumes. Control rod blade volumes are approximately $8.83\text{E}+01 \text{ m}^3$. That component is produced by BWRs exclusively and contributes the largest volume of material for any single nuclear utility component type.

Packaging has the most significant impact on the volumes of GTCC LLW that will be available for disposal by the DOE. Waste volumes for individual components vary according to packaging (e.g., filters can be reduced to as much as 20% of their original volume, and some metals can be expanded to nearly 30 times their original unpackaged volume). The average increase in the base case volumes of GTCC LLW, through packaging, is by a factor of 2 (doubling the volume for all wastes). The high case scenario can increase from 3 to 6 times the unpackaged volume for each component.

Activated metals from decommissioning of nuclear reactors contribute significantly to the total packaged volumes of GTCC LLW available for disposal from nuclear utilities. It is estimated that decommissioning wastes will account for nearly 30% of all base case GTCC LLW generated by nuclear utilities. The high case (projected to 2035) decommissioning waste is estimated to account for nearly 40% of the total volume of GTCC LLW generated from nuclear utilities.

High case data projected to the year 2055 (assuming that 70% of all currently active nuclear reactors have a life extension of 20 years) suggests a 21% increase in operations waste over the 2035 high case projections. Decommissioning wastes volumes for life extension will remain the same, the timing of decommissioning activities will be delayed by the life extension. The life extension may increase the radionuclide activity of those components.

Low case data from nuclear utilities operations waste indicate a much smaller amount of packaged waste available for disposal with a higher packaging efficiency. In the low case, decommissioning wastes are nearly 40% of the total volume of GTCC LLW material generated by nuclear utilities.

Concentration averaging reduces the volume of waste in the base and low cases, but as stated earlier does not affect the high case; averaging of radionuclide concentrations between components is not considered in this case.

The high case (2035) is considered as the "no concentration averaging" case and assumes radionuclide activities averaged over a single component only. The base case reflects a decrease in the packaged volumes of disposable GTCC LLW by 45% through concentration averaging. The low case scenario reduces the packaged volume of GTCC LLW by approximately 60%.

Concentration averaging affects operations waste exclusively. Decommissioning waste (from reactor internals) is assumed to have large radionuclide activities and may prohibit any concentration averaging of those materials. The base case packaged volumes are reduced by $8.40\text{E}+02 \text{ m}^3$ by concentration averaging.

Total radionuclide activity found in nuclear utility operations waste is around $2.23\text{E}+07 \text{ Ci}$, and an estimated $4.17+05 \text{ Ci}$ for decommissioning wastes. It is uncertain what radionuclide activities will be found in reactor internals until further research is done in this area. Radionuclides of most concern are ^{59}Ni and ^{63}Ni along with ^{94}Nb . These nuclides tend to drive waste radionuclide activity to GTCC LLW limits. Large activity radionuclides (which may not be listed in 10 CFR Part 61) are included in the nuclide estimates. Those nuclides often are used as scaling references for estimating other radionuclide totals developed in this study (i.e., ^{60}Co is used as a reference scale for ^{63}Ni) and may have relatively short half-lives (in the order of years to decades). All radionuclides are included in the totals; those radionuclides not included in Tables 2-1 and 2-2 of this study are included as additional information.

Nuclear utility waste accounts for approximately 57% of all packaged base case GTCC LLW reported in this study. Radionuclide activity content for these wastes is approximately 99% of estimated total activity of GTCC LLW projections with a total of $6.50\text{E}+07 \text{ Ci}$.

Figures 11-1 through 11-40 represent the nuclear utility GTCC LLW projections. As presented, the data show the effects of packaging and concentration averaging on each component listed in the waste stream. Operations waste, in the low and base cases, are affected by concentration averaging; decommissioning waste is not. The range of results for each case and scenario can be clearly seen in those figures.

11.2 Sealed Sources

The disposal volume of waste from sealed sources is small compared to the overall waste volume projected for GTCC LLW. However, the radionuclide concentrations found in these wastes are relatively high for the small volumes. The total unpackaged volumes of GTCC LLW estimated to be generated by specific licensee having GTCC LLW sealed sources are between $1.50\text{E-}02\text{ m}^3$ and $1.80\text{E-}02\text{ m}^3$. Packaged scenarios increase these volumes to $1.80\text{E+}01\text{ m}^3$ for the high case, and $3.00\text{E-}02\text{ m}^3$ for the low case with the base case project to $6.00\text{E+}00\text{ m}^3$.

Estimated base case sealed source packaged volumes are less than 0.1% of the entire base case GTCC LLW projected packaged volumes. The radionuclide content for these sealed sources is approximately 0.5% of the total curie content estimated for GTCC LLW. The largest number of sealed sources/devices comes from portable gauges. The greatest radionuclide activity comes from well logging and irradiator sources. The nuclides of interest and greatest contribution are ^{241}Am and ^{137}Cs . The other sources/devices contribute a relatively minor effect on the overall volume and radionuclide content of the GTCC LLW stream.

Timing of disposal of sealed sources that are potential GTCC LLW is unclear. It is assumed that all sealed sources/devices will become GTCC LLW, however, timing has not been established.

11.3 DOE-Held Potential GTCC LLW

There is an estimated $1.08\text{E+}03\text{ m}^3$ of packaged GTCC LLW held at DOE facilities awaiting disposal. It is unclear how much of that waste will require disposal at an NRC licensed disposal facility. A detailed list of components is not available, but a list of volumes packaged and estimated radionuclide activities appears in Table 10-23.

Volume estimates for DOE-held GTCC LLW are approximately 30% of all base case packaged GTCC LLW projected. The packaging factors of those waste containers in which the DOE-held potential GTCC LLW is stored is unknown. Repackaging of those wastes would substantially reduce the volumes for disposal. As an example, less than 10 m^3 of filter material from the TMI-2 reactor are stored in containers totaling over 160 m^3 . The radionuclide activity is approximately 80% of the total curie content. The largest volumes of DOE-held potential GTCC LLW comes from Nuclear Fuel Services, Sequoyah Falls, Westinghouse Pennsylvania, and Three Mile Island waste. The largest radionuclide activity comes from Three Mile Island waste.

11.4 Other Generators Waste

Waste included in this category is produced by several different generator types. Most of that waste is reported in the EIA survey as a one-time production; no future generation expected. Those wastes are listed in the data tables and reflect no increase in volume. The largest volume of GTCC LLW comes from compactible trash, most of which is reported by sealed source manufacturers. Without further analysis, it will be unclear if all of the compactible trash will be GTCC LLW.

Organic liquids (considered mixed waste) were reported on the EIA survey with no further waste generation expected. That waste has been absorbed, solidified, and packaged. No future waste volume generation was reported for that waste component.

Radionuclide activity information is not available at this time for all of the GTCC LLW listed in this category; however, the total activity is assumed to represent the majority of the GTCC LLW volumes projected.

Figure 11-41 represents the low, base, and high case volumes of GTCC LLW in the other generator category. No concentration averaging or unpackaged waste volumes are included with those figures because of a lack of detailed or complete data.

DATA GRAPHS

Nuclear Utility GTCC LLW (BWR) Control Rod Blade

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-1. Nuclear utility GTCC LLW (BWR) control rod blades.

R91 0559

Nuclear Utility GTCC LLW (BWR)
Local Power Range Monitor

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

R91 0560

Figure 11-2. Nuclear utility GTCC LLW (BWR) local power range monitor.

Nuclear Utility GTCC LLW (BWR)

Dry tubes

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-3. Nuclear utility GTCC LLW (BWR) dry tubes.

R91 0561

Nuclear Utility GTCC LLW (BWR)
FI decontamination resin

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

R91 0562

Figure 11-4. Nuclear utility GTCC LLW (BWR) fuel in decontamination resin.

Figure 11-5. Nuclear utility GTCC LLW (BWR) pool filters.

R91 0563

Nuclear Utility GTCC LLW (BWR)
Control rod drive strainers (outer)

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-6. Nuclear utility GTCC LLW (BWR) control rod drive strainers (outer). R91 0564

Nuclear Utility GTCC LLW (BWR)
Control rod drive strainers (inner)

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

R91 0565

Figure 11-7. Nuclear utility GTCC LLW (BWR) control rod drive strainers (inner).

Nuclear Utility GTCC LLW (BWR)
Cartridge filters ^a

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

^a Reported as 1986 inventory in the EIA survey.

R91 0566

Figure 11-8. Nuclear utility GTCC LLW (BWR) cartridge filters.^a

Nuclear Utility GTCC LLW (BWR)
Control rod bearings ^a

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

^a Reported as 1986 inventory in the EIA survey.

R91 0567

Figure 11-9. Nuclear utility GTCC LLW (BWR) control rod bearings.^a

Nuclear Utility GTCC LLW (BWR) Poison curtains ^a

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

^a Reported as 1986 inventory in the EIA survey.

R91 0568

Figure 11-10. Nuclear utility GTCC LLW (BWR) poison curtains.^a

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-11. Nuclear utility GTCC LLW (BWR) core shroud.

R91 0569

GTCC LLW Projection Scenarios - Babcock and Wilcox

In-core detectors

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-12. GTCC LLW projection scenarios--Babcock & Wilcox incore detectors. R91 0594

GTCC LLW Projection Scenarios - Babcock and Wilcox
Primary sources

R91 0595

Figure 11-13. GTCC LLW projection scenarios--Babcock & Wilcox primary sources.

GTCC LLW Projection Scenarios - Babcock and Wilcox Fuel in decon resins

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-14. GTCC LLW projection scenarios--Babcock & Wilcox fuel in decontamination resins. R91 0596

GTCC LLW Projection Scenarios - Babcock and Wilcox

Cartridge filters

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-15. GTCC LLW projection scenarios--Babcock & Wilcox cartridge filters.

R91 0597

GTCC LLW Projection Scenarios - Babcock and Wilcox
Crud tank filter

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-16. GTCC LLW projection scenarios--Babcock & Wilcox
crud tank filter.

R91 0598

GTCC LLW Projection Scenarios - Babcock and Wilcox Control rod drive

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-17. GTCC LLW projection scenarios--Babcock & Wilcox control rod drive.

R91 0599

GTCC LLW Projection Scenarios - Babcock and Wilcox

Flux wire

Unpackaged

Packaged

Concentration averaged

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-18. GTCC LLW projection scenarios--Babcock & Wilcox flux wire.

R91 0600

GTCC LLW Projection Scenarios - Babcock and Wilcox

Metals misc.

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-19. GTCC LLW projection scenarios--Babcock & Wilcox miscellaneous metals.

R91 0601

GTCC LLW Projection Scenarios - Babcock and Wilcox

Core shroud

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-20. GTCC LLW projection scenarios--Babcock & Wilcox core shroud.

R91 0602

GTCC LLW Projection Scenarios - Babcock and Wilcox

Core barrel

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-21. GTCC LLW projection scenarios--Babcock & Wilcox core barrel.

R91 0603

GTCC LLW Projection Scenarios - Combustion Engineering Instruments strings

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-22. GTCC LLW projection scenarios--Babcock & Wilcox instruments strings.

R91 0584

GTCC LLW Projection Scenarios - Combustion Engineering

Primary sources

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-23. GTCC LLW projection scenarios--Combustion Engineering primary sources. R91 0585

GTCC LLW Projection Scenarios - Combustion Engineering

Fuel in decon resins

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-24. GTCC LLW projection scenarios--Combustion Engineering
fuel in decontamination resins.

R91 0586

GTCC LLW Projection Scenarios - Combustion Engineering Cartridge filters

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-25. GTCC LLW projection scenarios--Combustion Engineering cartridge filters. R91 0587

GTCC LLW Projection Scenarios - Combustion Engineering
Thimble plug assemblies

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-26. GTCC LLW projection scenarios--Combustion Engineering thimble plug assemblies. R91 0588

GTCC LLW Projection Scenarios - Combustion Engineering Control rod drive

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-27. GTCC LLW projection scenarios--Combustion Engineering R91 0589
control rod drive.

GTCC LLW Projection Scenarios - Combustion Engineering Flux wire

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-28. GTCC LLW projection scenarios--Combustion Engineering flux wire. R91 0590

GTCC LLW Projection Scenarios - Combustion Engineering

Metals misc.

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-29. GTCC LLW projection scenarios--Combustion Engineering R91 0593 miscellaneous metals.

GTCC LLW Projection Scenarios - Combustion Engineering Core shroud

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-30. GTCC LLW projection scenarios--Combustion Engineering R91 0591 core shroud.

GTCC LLW Projection Scenarios - Combustion Engineering Core barrel

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-31. GTCC LLW projection scenarios--Combustion Engineering R91 0592
core barrel.

GTCC LLW Projection Scenarios - Combustion Engineering

Thimble plug assemblies ^a

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-32. GTCC LLW projection scenarios--Combustion Engineering thimble plug assemblies.

R91 0570

GTCC LLW Projection Scenarios - Combustion Engineering

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-33. GTCC LLW projection scenarios--Combustion Engineering source rods. R91 0573

GTCC LLW Projection Scenarios - Combustion Engineering

Fuel in decon resins

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-34. GTCC LLW projection scenarios--Westinghouse fuel in decontamination resins.

R91 0574

GTCC LLW Projection Scenarios - Combustion Engineering

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-35. GTCC LLW projection scenarios--Westinghouse cartridge filters.

R91 0575

GTCC LLW Projection Scenarios - Combustion Engineering

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-36. GTCC LLW projection scenarios--Westinghouse incore instruments.

R91 0579

GTCC LLW Projection Scenarios - Combustion Engineering

Low to 2035 Base to 2035 High to 2035 High to 2055

Figure 11-37. GTCC LLW projection scenarios--Westinghouse control rod drive.

R91 0580

GTCC LLW Projection Scenarios - Combustion Engineering

Metals misc.

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-38. GTCC LLW projection scenarios--Westinghouse miscellaneous metals.

R91 0581

GTCC LLW Projection Scenarios - Combustion Engineering

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-39. GTCC LLW projection scenarios--Westinghouse core shroud.

R91 0582

GTCC LLW Projection Scenarios - Combustion Engineering Core barrel

Low to 2035
 Base to 2035
 High to 2035
 High to 2055

Figure 11-40. GTCC LLW projection scenarios--Westinghouse core barrel.

R91 0583

Other Generators, GTCC LLW Packaged Volumes
Projected to 2035

Figure 11-41. Other generators, GTCC LLW packaged volumes projected to 2035.

R91 0571

Other Generators, GTCC LLW Packaged Volumes
Projected to 2035

Low to 2035
 Base to 2035
 High to 2035

Figure 11-41. continued.

Summation Data - Packaged Volumes

Figure 11-42. Summation data--packaged volumes.

R91 0605

R91 0606

Figure 11-43. Summation data of all tables--radionuclide activity.

12. CONCLUSIONS

The total base case volume of GTCC LLW is much less than previous estimates indicated. Independent research in similar areas has concluded that fewer components from reactor internals will be included as GTCC LLW. Efficient and reasonable packaging factors are applied to the waste volumes to reduce the previous estimates of disposal volumes of GTCC LLW. Concentration averaging reduces the volumes of operations waste.

The waste streams of greatest concern come from nuclear power utilities. It is estimated that BWRs and PWRs will generate approximately the same volume of GTCC LLW through base case operations as will be generated through decommissioning. The largest volumes of materials, for specific components, comes from BWR control rod blades. Other components that contribute large volumes are Fuel-In decontamination resins and decommissioning components from reactor internals. The largest radionuclide activity will come from reactor decommissioning waste components. All projected waste will be available for disposal by 2035 or 2055, if life extension is granted to select reactor facilities barring new reactor orders.

It is uncertain as to whether all of the reactor internals listed in the high case will become GTCC LLW. Further study must be conducted to clarify this issue. It is unclear which components are included as high-level waste. The Standard Contract for these waste types should be defined to list all components that will be affected.

The waste generated by other than nuclear utilities does not significantly affect the total volumes of GTCC LLW projections. Information was not reported for radionuclide activities of all material listed in that waste category. Those cases are noted in the tables.

Sealed sources have relatively small volumes, whether unpackaged or packaged for disposal. However, the radionuclide activity is large for the small volumes. No data are available to estimate the timing of generation of those wastes.

DOE has accepted $1.08\text{E}+03 \text{ m}^3$ (1077 m^3) of waste that are classified as GTCC LLW. At this time it is unclear what disposal requirements will be applied to each of the wastes listed in the DOE-held GTCC LLW category. Portions of that waste may not require disposal in an NRC-licensed facility.

GTCC LLW is generator specific. Differing operational procedures at the various generator facilities listed in this study are established by economic, safety, and practical application standards. The data presented in this study are best estimates based on averaging the waste generation rates at those facilities.

The total volume and radionuclide activity estimated as GTCC LLW by this study are:

Volume (packaged base case to 2035): $3.25\text{E}+03 \text{ m}^3$

Radionuclide activity $6.58\text{E}+07 \text{ Ci}$

12.1 Recommendations for Efforts to Reduce Uncertainties in Estimated Volumes

- Perform underwater dose-rate profiles of the core shrouds in the three BWR plants retired from service: Humbolt Bay, Dresden Unit 1, and Lacrosse. These plants have the fuel removed from the core, and access to the shroud should be easy if the pressure vessel top were removed. Estimates of concentrations of the relevant nuclides could be determined for those reactor internals. Those values would be used to improve the neutron flux estimates for the larger reactors, as well as for scaling the measured activities to the larger reactors and longer service life. Similar studies could occur at the retired Rancho Seco PWR on the shroud and core barrel.
- Evaluate the available neutron cross-section data, thermal and resonance, for nickel and niobium activation to ^{59}Ni , ^{63}Ni , and ^{94}Nb to obtain the best activation values. The evaluation would use reactor activation measurements using neutron energy-selective absorbers to separate thermal from epithermal activation and would utilize the DOE National Data Centers to access additional data and evaluations of known data on these cross sections.
- Evaluate available data, and initiate further research into the elemental concentrations of nickel and niobium in metals and in nuclear reactors. That study would help reduce uncertainty with regard to activation of those components.
- Perform additional searches for industry data of measured or estimated neutron fluxes in the regions of core shrouds and barrels, particularly for PWRs. This would require additional contacts with the utility industry and with the three principal manufacturers of U.S. PWR plants.
- Commission an integrated study by several independent researchers, administrated by one office, to study the research data gathered in the tasks listed above. Apply these results to the GTCC LLW volumes to reflect the projections as reasonably as possible.
- Continue to refine estimates for numbers of sealed sources and disposition of sources after reaching expected working life of license (i.e., disposed of or recycled).
- Establish an all-inclusive list for components designated as high-level waste under the Standard Contract for high-level waste disposal.
- Establish a procedure for the use of concentration averaging.
- Determine the disposal requirements for DOE-held potential GTCC LLW.
- Perform an assessment on each category of GTCC LLW to determine whether any GTCC LLW should be considered a mixed waste.

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APPENDIX A

ENERGY INFORMATION ADMINISTRATION
SURVEY FORM

NE-869 GREATER THAN CLASS C LOW-LEVEL RADIOACTIVE WASTE AND RADIUM WASTE DATA FORM

NE-869

U. S. DEPARTMENT OF ENERGY
WASHINGTON, D. C. 20585

Form Approved
OMB No. 1901-0290
Expires: 9/30/1986

Data on this one time mandatory form are collected under the authority of the Federal Energy Administration Act of 1974 (15 USC Section 761 et. seq.), Section 170B of the Atomic Energy Act of 1954 (42 USC 2210b), and the Low-Level Radioactive Waste Policy Amendments Act of 1985. This form shall be returned by all addressees. Owners, generators, and holders of:

1. low-level radioactive waste having radionuclide concentrations that exceed Class C limits according to 10 CFR Part 61 "Licensing Requirements for Land Disposal of Radioactive Waste," OR
2. other waste materials that contain radium-226 in concentrations that exceed 100 nCi/g,

must complete appropriate sections of this form. Those who do not own, generate or hold such wastes should so indicate in Question 2.1, complete the appropriate parts of Section 1, and return the form.

This form is to be completed based on data as of December 31, 1985 and should be filed no later than six weeks after receipt of the form. A separate survey form should be filed for each facility or nuclear power plant site owned by a respondent. For example, a nuclear utility could own and operate several sites, and a site might include two nuclear power generating units; a university might include a research hospital facility and a science or engineering research facility.

This data will be used in preparation of the Congressionally mandated report. The Department of Energy (DOE) plans to use the data in preparing recommendations for further legislation necessary to DOE's fulfillment of its disposal obligations.

The definitions of Class C limits and waste to be included in this survey are given in Appendix A. Guidelines for classification of waste are in Appendix B. Appendix C is a glossary of terms used in this survey.

Please refer all questions regarding this form to Margaret Knecht at (208) 526-4531, or Winston Hickman at (208) 526-9641, on weekdays between 7:30 a.m. and 4:30 p.m. Mountain time (9:30 a.m. and 6:30 p.m. Eastern time).

Please return completed forms to:

Energy Information Administration - (EI-53)
U. S. Department of Energy
Attn: Greater Than Class C Waste
Mail Stop BG-094
Washington, D. C. 20585

NE-869 GREATER THAN CLASS C LOW-LEVEL RADIOACTIVE WASTE AND RADIUM WASTE DATA FORM

The data provided on Form NE-869 by nuclear utilities are not considered to be confidential.

The data provided on Form NE-869 by all respondents other than nuclear utilities will be considered confidential as follows:

The information contained in this form will be kept confidential to the extent that it satisfies the criteria set forth in the Freedom of Information Act (FOIA) exemption for trade secrets and confidential commercial information and Department of Energy (DOE) regulations implementing the FOIA and is prohibited from public release by the Trade Secrets Act, 18 U.S.C. §1905.

Upon receipt of a request for disclosure of this information under the FOIA, the DOE shall, in accordance with the procedures and criteria provided in 10 C.F.R. §1004.11, make a final determination whether the information is exempt from disclosure. To assist us in this determination, respondents should demonstrate to the DOE that their information constitutes trade secrets or commercial or financial information whose release would be likely to cause substantial harm to their company's competitive position. A letter accompanying the submission that explains (on an element-by-element basis, if possible) the reasons why the information would be likely to cause the respondent substantial competitive harm if released to the public would aid in this determination.

Requests from other federal agencies for information from this form shall be evaluated in accordance with the DOE Policy on the Disclosure of Individually Identifiable Energy Information in the Possession of the EIA [45 Fed. Reg. 59812 (1980)]. Respondents should be aware that this information will be disclosed to State agencies for certain purposes, provided they can demonstrate adequate safeguards to prevent disclosure of the data.

Except as otherwise provided by law, the information will also be made available in response to an order of a Court of competent jurisdiction, or upon written request, to the Congress, any Committee of Congress, the General Accounting Office, or other Congressional agencies authorized by law to receive such information.

NOTE: The term waste, as used throughout this form, refers to low-level radioactive waste that exceeds Class C limits and other waste that contains radium-226 in concentrations greater than 100 nCi/g.

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1.0 Respondent Data

1.1 Plant or Facility Name: _____
Company or Institution Name: _____
Mailing Address: _____

City / State / ZIP

1.2 Person who may be contacted if necessary to clarify information provided on this form:

Name: _____
Title: _____
Mailing Address: _____

City / State / ZIP
Telephone Number: (____) ____ - ____

1.3 Authorized Signature/Certification: I certify that the information contained herein is true and accurate to the best of my knowledge.

Name: _____
Title: _____
Mailing Address: _____

City / State / ZIP
Signature: _____ Date: _____

1.4 Plant or Facility Business or Activity Category

a. Nuclear Facility	Yes ☐ No ☐
b. Nuclear Research or Test Reactor	Yes ☐ No ☐
c. Nuclear Fuel Fabricator or Burnup Lab	Yes ☐ No ☐
d. Medical Institution, Research	Yes ☐ No ☐
e. Medical Institution, Nonresearch	Yes ☐ No ☐
f. Waste Service Company (e.g. broker, processor)	Yes ☐ No ☐
g. Sealed Source Manufacturer	Yes ☐ No ☐
h. Sealed Source Distributor	Yes ☐ No ☐
i. Manufacturer of Devices Containing Sealed Sources	Yes ☐ No ☐
j. Industrial Research and Development	Yes ☐ No ☐
k. Academic Institution, Nonmedical	Yes ☐ No ☐
l. Other	Yes ☐ No ☐

If other, please specify: _____

- 1.5 Are you licensed to possess any of the following radionuclides: C-14, Ni-59, Nb-94, Tc-99, I-129, Am-241, Pu-241, Cm-242, other transuranic nuclides with half-lives greater than 5 years, Ni-63, Sr-90, or Cs-137? Yes ☐ No ☐. If yes, are you licensed by the Nuclear Regulatory Commission? Yes ☐ No ☐; state? Yes ☐ No ☐. If by a state, please specify which state(s): _____

- 1.6 Are you licensed, registered, or otherwise authorized to possess radium-226? Yes ☐ No ☐. If yes, please give state(s): _____.

QUESTIONS 1.7 AND 1.8 ARE TO BE ANSWERED ONLY BY MANUFACTURERS AND DISTRIBUTORS OF SEALED SOURCES AND DEVICES CONTAINING SEALED SOURCES

- 1.7 MANUFACTURERS AND DISTRIBUTORS OF SEALED SOURCES AND DEVICES CONTAINING SEALED SOURCES: Do you have arrangements with your customers for them to return to you the sealed sources/devices that you provided to them? Yes ☐ No ☐. If yes, please provide any qualifying conditions: _____

- 1.8 MANUFACTURERS AND DISTRIBUTORS OF SEALED SOURCES AND DEVICES CONTAINING SEALED SOURCES: Will you accept discarded (waste) or damaged sealed sources or devices containing sealed sources which you did not originally supply?

Yes ☐ No ☐. If yes, please provide any qualifying conditions: _____

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QUESTION 1.9 IS TO BE ANSWERED ONLY BY MANUFACTURERS OF SEALED SOURCES

- 1.9 MANUFACTURERS OF SEALED SOURCES: Have you manufactured sealed sources that you estimate contain radionuclide concentrations that exceed Class C concentrations limits or contain radium in concentrations exceeding 100 nCi/g? Yes No .
(Please see Appendix B for assumptions and methods to be used for determining waste classification of sealed sources.) If yes, please provide the descriptive data for each source type below.

Number of Units	Source Size (Curies)	Radio- Nuclide	Neutron Source? (Yes/No)	Model Number*	Use*	Volume per Source (cm ³)	Mass per Source (grams)
_____	_____	_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____	_____	_____

- * "Model Number" refers to the manufacturer's model number.
"Use" refers to the current or former use (e.g., irradiator source, well logging source, gauge, startup neutron source, etc.)

Comments and Clarifications: _____

QUESTIONS 1.10 AND 1.11 ARE TO BE ANSWERED ONLY BY WASTE SERVICE COMPANIES

1.10 WASTE SERVICE COMPANIES: Have you been requested to provide disposal services for sealed sources, devices containing sealed sources, or other wastes that exceed Class C limits or contain radium-226 in concentrations that exceed 100 nCi/g? Yes___No___. If yes, please give as much of the following information as you can so that such waste can be taken into account in planning for future disposal capacity and such persons can be notified when disposal services are available. (Alternatively, you may forward a copy of this survey to such individuals and record their name below.)

Company Requesting Disposal Number*

Description**

[illegible]

* Number of items.

** Description of items, including size (curies), radionuclides, neutron source (Y/N), volume, mass, condition (leaking or intact).

1.11 WASTE SERVICE COMPANIES: Do you accept, for interim storage and future disposal, waste sealed sources, devices containing sealed sources, or other wastes that exceed Class C limits or that contain radium-226 in concentrations exceeding 100 nCi/g? Yes No . If yes, please give qualifying conditions:

2.0 Identification of Current Holders and Future Generators of Waste

- 2.1 Are you a current or future owner, generator, or holder of low-level radioactive waste that exceeds Class C limits or that contains radium-226 in concentrations exceeding 100 nCi/g? Yes___No___. (See Appendix A for definitions of Class C concentration limits and waste to be included in and excluded from this survey. See Appendix B for guidelines for determining concentrations of radionuclides in waste. See Appendix C for definitions of low-level radioactive waste and other terms used in this survey.)

If yes, please answer Questions 2.2 to 2.5 below.

If no, please complete Section 1.0 and return this form to the address indicated on the front page of this form.

- 2.2 Do you currently possess low-level radioactive waste that exceeds Class C concentration limits or that contains radium-226 in concentrations exceeding 100 nCi/g? Yes___No___. If yes, please complete Section 3.0.
- 2.3 Do you anticipate future possession of any additional low-level radioactive waste that exceeds the Class C concentration limits or that contains radium-226 in concentrations exceeding 100 nCi/g? Yes___No___. If yes, please complete Section 4.0.
- 2.4 Do you currently possess or expect to possess greater than Class C low-level radioactive waste or greater-than-100-nCi/g radium-226 wastes in quantities or under conditions that will exceed your present capacity to provide adequate storage? Yes___No___. If yes, please complete Section 5.0.
- 2.5 If, in the future, you ceased your current operations and decommissioned your facility, do you believe that any of the waste material generated at that time would exceed Class C limits? Yes___No___. If yes, please complete Section 6.0.

3.0 Current Waste Inventories

PLEASE COMPLETE A SEPARATE SECTION 3.0 FOR EACH INDIVIDUAL RADIOACTIVE WASTE STREAM (AT THIS FACILITY OR NUCLEAR POWER PLANT SITE) THAT IS IDENTIFIED IN QUESTION 3.1 BELOW. ONLY ONE WASTE STREAM SHOULD BE SHOWN IN QUESTION 3.1 AND REPORTED IN EACH SECTION 3.0

Waste stream category; please mark the category in 3.1.1 or 3.1.2 that best describes the waste stream for which this Section 3.0 is being completed. Complete a separate Section 3.0 for each individual waste stream.

3.1 Does the waste stream contain neutron-activated metal pieces or parts?
Yes No . If no, please mark a category in 3.1.1. If yes, please check a category in 3.1.2.

3.1.1 Waste NOT containing neutron-activated metal pieces or parts:

a. <u> </u> Solidified liquid	n. <u> </u> Organic liquid
b. <u> </u> Filter media	o. <u> </u> Animal carcasses
c. <u> </u> Dewatered ion exchange resin	p. <u> </u> Other biological material
d. <u> </u> Solidified resin	q. <u> </u> Foils or absorbed gas
e. <u> </u> Absorbed liquid	r. <u> </u> Wire
f. <u> </u> Compactible trash	s. <u> </u> Paint or plating
g. <u> </u> Noncompactible trash	t. <u> </u> Demolition rubble
h. <u> </u> Contaminated equipment	u. <u> </u> Glassware
i. <u> </u> Incinerator ash	v. <u> </u> Other. If other, please
j. <u> </u> Soil	specify: <u> </u>
k. <u> </u> Cartridge filter	<u> </u>
l. <u> </u> Sealed source	<u> </u>
m. <u> </u> Aqueous liquid	<u> </u>

3.1.2 Waste containing neutron-activated metal pieces or parts (see Appendix A for explanation of utility waste to be included in and excluded from this survey):

a. <u> </u> Control rod blade	h. <u> </u> Other in-core instruments
b. <u> </u> Poison curtain	i. <u> </u> Burnable poison rod
	assembly
c. <u> </u> Local power range monitor	j. <u> </u> Primary/secondary source
d. <u> </u> Flux wire	assembly
e. <u> </u> Rod control cluster assembly	k. <u> </u> Other. If other, please
f. <u> </u> Fuel channels	specify: <u> </u>
g. <u> </u> Thimble plugs or assembly	<u> </u>

FOR SEALED SOURCES THAT ARE CONSIDERED TO BE WASTES, PLEASE PROVIDE DATA IN QUESTIONS 3.11 TO 3.13;

FOR OTHER WASTES, PLEASE PROVIDE DATA IN QUESTIONS 3.2 TO 3.10.

3.2 If the waste stream is compactible trash, has the waste been compacted?
Yes No Not applicable

3.3 What is the process that generated this waste? _____

3.4 Was this waste generated by an accident or unexpected incident (as opposed to routine operations)? Yes ___ No ___. If yes, please briefly describe the accident or unexpected incident _____

3.5 Principal chemical form of the waste: _____

(if activated metals, check type of metal in list below):

a. ___ Haynes-25	g. ___ Stellite-3
b. ___ Inconel-718	h. ___ Zircaloy-2
c. ___ Inconel X, 750	i. ___ Zircaloy-4
d. ___ Nicobraz-50	j. ___ Other. If other, please
e. ___ Stainless Steel-302	specify: _____
f. ___ Stainless Steel-304	_____

3.6 Has this waste been absorbed or solidified? Yes ___ No ___. If yes, please check the absorbent or solidification agent in the list below:

a. ___ Speedi-Dry	j. ___ Dow media
b. ___ Celatom (MP-78)	k. ___ Cement
c. ___ Floor Dry/Super Fine	l. ___ Asphalt
d. ___ Hi Dri	m. ___ Delaware Custom Media
e. ___ Florco or Florco-X	n. ___ Envirostone
f. ___ Instant Dri	o. ___ Krolite
g. ___ Safe T Sorb	p. ___ Other. If other, please
h. ___ Oil-Dri (Safe n Dri)	specify: _____
i. ___ Zonolite Grades 2,3,4	_____

3.7 Does the waste contain hazardous chemical waste as defined by EPA in 40 CFR Part 261? Yes ___ No ___. If yes, please indicate the hazardous waste (information on defining hazardous waste is in Appendix D) and its approximate concentration in ppm or weight percent: _____

- 3.8 Does the waste contain more than 0.1% by weight of chelating agent?
 Yes ___ No ___. If yes, please give name and percent by weight for each chelating agent(s). See Appendix C for a definition of chelating agents: _____

- 3.9 Please provide waste data below as of December 31, 1985 (see Appendixes A and B for guidelines on determining waste volume, mass, and activity for the purposes of waste classification):

- a. Total approximate waste volume _____ (cubic meters)
 b. Total approximate waste mass: _____ (kilograms)
 c. Total approximate activity in waste _____ (curies) as of _____
 (approximate date of activity given above, if not December 31, 1985)
 d. Please check each radionuclide that is present and provide an estimate of the total activity in curies for each item checked:

Long-Lived Radionuclides

Nuclide	Activity (Curies)
C-14	_____
Ni-59	_____
Nb-94	_____
Tc-99	_____
I-129	_____
Am-241	_____
Pu-241	_____
Cm-242	_____
Other TRU*	_____
Ra-226	_____

Short-Lived Radionuclides

Nuclide	Activity (Curies)
H-3	_____
Co-60	_____
Ni-63	_____
Sr-90	_____
Cs-137	_____
Nuclides with half life less than 5 years (total) _____	

* TRU: Alpha-emitting transuranic radionuclides having half-lives exceeding 5 years.

- e. By what mechanism have you determined the activity in this waste?
1. Direct measurement of specific nuclides _____,
 2. Measurement of some nuclides, scaling of others _____,
 3. Calculations based on activation analysis _____,
 4. Estimates based on a known quantity of radioactive material involved in a particular process _____,
 5. Other (please describe) _____

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3.9 Continued

f. What is the estimated accuracy of the activity determination?

1. Within 10% _____ 2. Within 100% _____ 3. Within a factor of 10 _____
 4. Other _____. If other, please specify _____
 5. Unknown _____.

g. Does this waste contain fissile material? Yes ___ No ___. If yes, give amounts in grams per kilogram of waste:

U-233 _____, U-235 _____, Pu-239 _____,
 Other (in grams of Pu-239 equiv.) _____.

h. Please estimate the approximate percentage of this waste that would require remote handling _____. (For unpackaged waste streams, assume any standard package type and use a value of 200 mrem/hr at the package surface as the maximum level for contact-handled waste packages.)

3.10 Packaging. Is the waste now packaged? Yes ___ No ___. If yes, please indicate the types of packages, volumes of individual packages and number of each type of package according to the following categories:

Package Type	Number of Packages	Volume per Package (feet ³)
Drum	_____	_____
Metal Liner	_____	_____
High Integrity Container (HIC)	_____	_____
Metal Box	_____	_____
Wooden Box	_____	_____
Other (specify)	_____	_____

QUESTIONS 3.11 TO 3.13 APPLY ONLY TO WASTE SEALED SOURCES. PLEASE LEAVE BLANK IF YOU DO NOT POSSESS ANY WASTE SEALED SOURCES.

3.11 SEALED SOURCES. Are you currently in possession of any sealed sources that are considered waste, AND are expected to exceed Class C limits? Yes No .
(See Appendix B for guidelines on determining Class C limits for sealed sources.)

If no, please skip the remaining questions in Section 3 and continue on to Section 4.

If yes, please use the tables below to specify the number of sources, size (radioactivity per source), and other characteristics indicated below for these waste sealed sources. (Examples of "use" are irradiator source, well logging source, gauge, neutron startup source, etc.) Do NOT include sealed sources for which you have an arrangement to (a) recycle the source, (b) recycle the radioactive material in the source, (c) transfer the source to another party for their use, or (d) return the source to an appropriate manufacturer or distributor.

	Number	Size (Curies)	Nuclide	Neutron Source? (Yes/No)	Single Source Volume (cm ³)	Single Source Mass (Grams)
a.	_____	_____	_____	_____	_____	_____
b.	_____	_____	_____	_____	_____	_____
c.	_____	_____	_____	_____	_____	_____
d.	_____	_____	_____	_____	_____	_____
e.	_____	_____	_____	_____	_____	_____
f.	_____	_____	_____	_____	_____	_____

For each type of source listed above line by letter "a," "b," etc., please give manufacturer, model number (or other model identification if model number is not known), state whether the source is known to be leaking, and give the approximate year of manufacture or purchase:

	Manufacturer	Model Number	Leaking? (Yes/No)	Mfg. Year	Former Use, if Known
a.	_____	_____	_____	_____	_____
b.	_____	_____	_____	_____	_____
c.	_____	_____	_____	_____	_____
d.	_____	_____	_____	_____	_____
e.	_____	_____	_____	_____	_____
f.	_____	_____	_____	_____	_____

- 3.12 Do any of the waste sealed sources listed in Question 3.11 contain Cs-137?
 Yes No . If yes, please check the chemical form of the radioactive
 material below for each source type identified by the same line letter ("a,"
 "b," etc.) as used in Question 3.11.

Source Type	Number of Units	Chemical form of the Cs-137			
		Unknown	Cesium Chloride	Ceramic	Other (Please Specify)
_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____

- 3.13 Are any of the sealed sources listed in Question 3.11 contained in instruments,
 equipment, or other containers from which they will be extremely difficult to
 remove? Yes No . If yes, did the volume and mass in the table above
 include such material? Yes No . If yes, please describe the instrument,
 equipment, or other container for each such source type, and identify which
 units in Question 3.11 they are. _____

- a. ☐ Control rod blade
- b. ☐ Poison curtain
- c. ☐ Local power range monitor
- d. ☐ Flux wire
- e. ☐ Rod control cluster assembly
- f. ☐ Fuel channels
- g. ☐ Thimble plugs or assembly
- h. ☐ Other in-core instruments
- i. ☐ Burnable poison rod assembly
- j. ☐ Primary/secondary source assembly
- k. ☐ Other. If other, please specify: _____

FOR SEALED SOURCES THAT ARE CONSIDERED TO BE WASTES, PLEASE PROVIDE DATA IN QUESTIONS 4.13 TO 4.16
FOR OTHER WASTES PROVIDE DATA IN QUESTIONS 4.2 TO 4.12.

4.2 What is the activity or process that will generate this waste? _____

4.3 Principal chemical form of the waste: _____

(if activated metals, check type of metal in list below):

a. ☐ Haynes-25	g. ☐ Stellite-3
b. ☐ Inconel-718	h. ☐ Zircaloy-2
c. ☐ Inconel-X, 750	i. ☐ Zircaloy-4
d. ☐ Nicobraz-50	j. ☐ Other. If other, please
e. ☐ Stainless Steel-302	specify: _____
f. ☐ Stainless Steel-304	
k. ☐ Unknown	

4.4 Is it expected that this waste will be absorbed or solidified?
Yes ☐ No ☐ Don't know ☐. If yes, please check the expected absorbent or solidification agent in the list below, if known:

a. ☐ Speedi-Dry	j. ☐ Dow media
b. ☐ Celatom (MP-78)	k. ☐ Cement
c. ☐ Floor Dry/Super Fine	l. ☐ Asphalt
d. ☐ Hi Dri	m. ☐ Delaware Custom Media
e. ☐ Florco or Florco-X	n. ☐ Envirostone
f. ☐ Instant Dri	o. ☐ Krolite
g. ☐ Safe T Sorb	p. ☐ Other. If other, please
h. ☐ Oil-Dri (Safe n Dri)	specify: _____
i. ☐ Zonolite Grades 2,3,4	
q. ☐ Unknown	

- 4.5 Will this waste contain hazardous chemical waste as defined by EPA in 40 CFR Part 261? Yes___ No___ Don't know___. If yes, please indicate the hazardous waste (information on defining hazardous waste is in Appendix B) and its approximate concentration in ppm or weight percent:_____
- 4.6 Will this waste contain more than 0.1% by weight of chelating agent? Yes___ No___ Don't know___. If yes, please give name and percent by weight for each chelating agent(s). (see Appendix C for a definition of chelating agents)_____
- 4.7 Will this waste contain fissile material? Yes___ No___ . If yes, please give the estimated concentration in grams per kilogram of waste:
- U-233___ U-235___ Pu-239___ Other (in grams of Pu-239 equivalent)_____
- 4.8 Please estimate the approximate percentage of this waste that will require remote handling___. (For unpackaged waste streams, assume any standard package type and use a value of 200 mrem/hr at the package surface as the maximum level for contact-handled waste packages.)

QUESTIONS 4.10 AND 4.11 ARE ALTERNATIVE QUESTIONS ON RATE OF GENERATION OF FUTURE WASTE.

QUESTION 4.10 APPLIES TO WASTE THAT IS USUALLY GENERATED EVERY YEAR AND CAN BE REPRESENTED BY AN ANNUAL RATE OF GENERATION.

QUESTION 4.11 APPLIES TO WASTE THAT IS GENERATED IN AN ISOLATED INSTANCE OR IS GENERATED INTERMITTENTLY AND IS EASIER TO EXPRESS IN TERMS OF SEPARATE PROJECTED OCCURRENCES.

PLEASE ANSWER THE QUESTION THAT BEST REPRESENTS THE EXPECTED RATE OF GENERATION OF THE FUTURE WASTE STREAM BEING ADDRESSED. EACH QUESTION MAY BE ANSWERED FOR A PORTION OF THE WASTE STREAM IF THE WASTE STREAM IS BEST REPRESENTED BY SO DOING

- 4.9 Please check which of Questions 4.10 and 4.11 (rate of generation questions) are answered:
- Only Question 4.10___ (Annual rate of generation).
Only Question 4.11___ (Isolated instance or intermittent generation).
Both Questions 4.10 and 4.11

4.10 FUTURE WASTE--ANNUAL RATE OF GENERATION THROUGH 2020: (see Appendixes A and B for guidelines on determining waste volume and mass.)

- a. For future waste, please use the table below to provide estimates of the year the expected waste generation will start and the number of years (duration) this waste will be generated through 2020. If the duration is unknown, it will be assumed that this waste will be generated indefinitely, until possible termination or other change of business at some unknown future date. Average annual waste volume, mass, and activity should also be given for the duration of the waste generation. Your estimates should start with 1986 unless the waste generation will not begin until a later year. (See Appendixes A and B for guidelines on determining waste volume and mass.)

First Year	Duration (Years)	Average Annual Waste Volume (meters ³)	Average Annual Waste Mass (kg)	Average Annual Activity (Curies)
_____	_____	_____	_____	_____

- b. Is the volume given above for packaged or unpackaged waste? Packaged____
Unpackaged____
- c. Please indicate if the annual rate of future waste generation through 2020 is expected to change over time (e.g. increase 5% annually)
Yes____ No____ Don't know____. If yes, please describe as best you can the expected percentage changes in average annual volume, the approximate years over which these changes are expected to occur, whether the changes are increases or decreases: _____

4.11 FUTURE WASTE--INTERMITTENT RATE OF GENERATION THROUGH 2020:

For waste generation that cannot easily be represented by an annual rate of generation, please use the table below to provide estimates of the total volume, mass, and activity of waste projected to be generated in the period 1986 through 2020. (See Appendixes A and B for guidelines on determining waste volume and mass.) Estimate the principle generation periods as the first and last year of the period and give the approximate percentage of the volume that is projected to be generated in each such period. Do not include decommissioning waste (see Section 6 for decommissioning waste estimates.)

Total Volume ₃ (meters ³)	Total Mass (kg)	Total Activity (Curies)	Principal Generation Periods		
			First year	Last year	Percentage of Total
_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____

If you cannot be specific in the above table, please use the space below to provide any information that would assist DOE in making an initial estimate of disposal capacity needed for future waste. Approximate volumes and some estimate of the periods in which the waste would be generated are needed: _____

4.12 FUTURE WASTE--AVERAGE RADIONUCLIDE DISTRIBUTION:

With respect to your answers in 4.10 and 4.11, please check each of the following radionuclides that are expected to be present in future waste and use the table below to provide an estimate of the average radionuclide concentrations in the waste.

<u>Long-Lived Radionuclides</u>		<u>Short-Lived Radionuclides</u>	
Nuclide	Concentration (Ci/meter ³)	Nuclide	Concentration (Ci/meter ³)
C-14	_____	H-3	_____
Ni-59	_____	Co-60	_____
Nb-94	_____	Ni-63	_____
Tc-99	_____	Sr-90	_____
I-129	_____	Cs-137	_____
Am-241*	_____	Nuclides with half	
Pu-241*	_____	life less than	
Cm-242*	_____	5 years (total)	
Other TRU*, **	_____		
Ra-226*	_____		

* Please give concentration in nanocuries per gram for these radionuclides.

** TRU: Alpha-emitting transuranic radionuclides having half-lives exceeding 5 years.

What is the estimated accuracy of the activity projections?

1. Within 10% _____ 2. Within 100% _____ 3. Within a factor of 10 _____
4. Other _____. If other, please specify _____
5. Unknown _____

ADDITIONAL COMMENTS AND CLARIFICATIONS:

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QUESTIONS 4.13 TO 4.16 APPLY ONLY TO ADDITIONAL WASTE SEALED SOURCES THAT YOU EXPECT TO POSSESS IN THE FUTURE (1986 AND BEYOND) AND WERE NOT INCLUDED IN SECTION 3. IF YOU DO NOT EXPECT TO POSSESS ADDITIONAL WASTE SEALED SOURCES IN 1986 AND BEYOND, DO NOT COMPLETE THE REMAINING QUESTIONS OF SECTION 4

4.13 SEALED SOURCES. In the future, do you expect to possess waste sealed sources that WILL exceed Class C limits or contain Ra-226 in concentrations exceeding 100 nCi/g? Yes ___ No ___

If yes, please use the tables on the next page to specify as best you can the number of sources, size (radioactivity content per source), use, and other characteristics indicated below for these waste sealed sources. (Examples of "use" are irradiator source, well logging source, gauge, neutron startup source, etc.) Do NOT include sealed sources for which you will have an arrangement to (a) recycle the source, (b) recycle the radioactive material in the source, (c) transfer the source to another party for their use, or (d) return the source to an appropriate manufacturer or distributor.

Number	Size (Curies)	Nuclide	Neutron Source? (Yes/No)	Volume per Source (cm ³)	Mass per Source (Grams)
a. _____	_____	_____	_____	_____	_____
b. _____	_____	_____	_____	_____	_____
c. _____	_____	_____	_____	_____	_____
d. _____	_____	_____	_____	_____	_____
e. _____	_____	_____	_____	_____	_____
f. _____	_____	_____	_____	_____	_____

For each type of source listed above on line "a," "b," etc., please use the same line "a," "b," etc., below to give any known data on manufacturer, model number (or other model identification if model number is not known), approximate year disposal will be required, and approximate year of manufacture or purchase:

	Manufacturer	Model Number	Disposal Year	Mfg. Year	Use
a. _____	_____	_____	_____	_____	_____
b. _____	_____	_____	_____	_____	_____
c. _____	_____	_____	_____	_____	_____
d. _____	_____	_____	_____	_____	_____
e. _____	_____	_____	_____	_____	_____
f. _____	_____	_____	_____	_____	_____

- 4.14 Will any of the waste sealed sources listed in Question 4.13 contain Cs-137? Yes No . If yes, please check the chemical form of the radioactive material below for each source type identified by the same line letter ("a," "b," etc.) as used in Question 4.13:

Source Type	Number of Units	Chemical Form of the Cs-137			
		Unknown	Cesium Chloride	Ceramic	Other (Please Specify)
_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____
_____	_____	_____	_____	_____	_____

- 4.15 Will any of the sealed sources listed in Question 4.13 be contained in instruments, equipment, or other container from which they will be extremely difficult to remove? Yes No . If yes, did the volume and mass in Question 4.13 above include such material? Yes No . If yes, please describe the instrument, equipment, or other container for each such source type, identified by the same line letter "a," "b," etc., as used for that source type in

Question 4.13: _____

- 4.16 If you believe that you will be in possession of a waste sealed source that will exceed Class C limits or contain Ra-226 in concentrations exceeding 100 nCi/g, but are unable to be specific regarding numbers, sizes, etc., please provide whatever information you can: _____

5.0 Storage of Waste

THIS SECTION IS TO BE COMPLETED BY ALL RESPONDENTS WHO ANSWERED YES TO QUESTION 2.4. A NUMBER OF YEARS MAY ELAPSE PRIOR TO AVAILABILITY OF DISPOSAL SERVICES FOR RADIUM-226 WASTES AND FOR WASTES EXCEEDING CLASS C LIMITS. SUCH WASTES MUST BE SAFELY STORED IN THE INTERIM. THE PURPOSE OF THIS SECTION IS TO DETERMINE WHETHER THERE IS A NEED FOR ADDITIONAL CAPACITY TO STORE THESE WASTES PRIOR TO DISPOSAL.

- 5.1 Please estimate when your inventories of such wastes will exceed your present storage capacity. Now? Yes ___ No ___. Future? Yes ___ No ___. If yes, please indicate approximate year _____ and check the following types of wastes for which storage capacity will be exceeded:

- a. Wastes that contain radium-226 in concentrations that exceed 100 nCi/g
b. Low-level radioactive wastes exceeding Class C limits.

- 5.2 In the absence of off-site storage capacity, how would you plan to resolve these difficulties? (Please check applicable answers below.)

- a. ☐ Apply for an increase in radioactive material possession limits
- b. ☐ Construct additional storage capacity
- c. ☐ Discontinue operation that generates such waste
- d. ☐ Discontinue receiving such radioactive material
- e. ☐ Other. If other, please describe:

- 5.3 Please estimate your costs in meeting the plans listed in Question 5.2

[illegible]

- 5.4 Additional comments _____

6.0 Decommissioning Waste

THIS SECTION IS TO BE COMPLETED BY RESPONDENTS WHO ANSWERED YES TO QUESTION 2.5.

6.1 Estimated years over which decommissioning waste will be generated: _____

6.2 Please describe the waste you expect to generate. If possible, provide potential waste forms, volumes, and principle radionuclide content and inventories: _____

6.3 Additional Comments and Clarifications: _____

APPENDIX A

DEFINITION OF GREATER-THAN-CLASS-C LOW-LEVEL RADIOACTIVE WASTES

Low-level radioactive waste may be broadly defined as any radioactive discard material other than high-level waste, spent nuclear fuel, or byproduct material as defined in Section 11e.(2) of the Atomic Energy Act (uranium or thorium tailings and waste). Low-level radioactive waste may also be grouped into three distinct classes of waste acceptable for disposal at existing and future low-level radioactive waste disposal facilities. These waste classes are termed Class A, Class B, and Class C wastes, where each class of waste contains radionuclides in progressively higher concentrations. There is currently no disposal capacity for Greater-than-Class-C wastes.

Wastes to be Excluded from This Survey

The following greater-than-Class-C low-level radioactive wastes are EXCLUDED from this survey:

- o low-level radioactive waste owned or generated by the Department of Energy;
- o low-level radioactive waste owned or generated by the United States Navy as a result of the decommissioning of vessels of the United States Navy;
- o low-level radioactive waste owned or generated by the Federal Government as a result of any research, development, testing, or production of any atomic weapon.
- o any nuclear utility spent fuel assembly waste that is to be tendered to the Department of Energy under the terms of Appendix E of the standard utility--Department of Energy contract for disposal of spent fuel. Information on such waste will be gathered via the yearly spent fuel survey of nuclear utilities (NE-859).

Wastes to be Included in This Survey

The following greater-than-Class C low-level radioactive wastes are INCLUDED in this survey:

- o any other low-level radioactive waste with concentrations of radionuclides that exceed the limits established by the Nuclear Regulatory Commission (NRC) for Class C radioactive waste, as defined by Section 61.55 of Title 10, Code of Federal Regulations, as in effect on January 26, 1983 ("Licensing Requirements for Land Disposal of Radioactive Waste").

Greater-than-Class-C Low-Level Radioactive Waste Classification

For this survey, waste classification is based on the waste classification system in 10 CFR Part 61. This classification system is based on the concentration of particular radionuclides considered important for disposal. According to this system, Class C wastes contain the highest allowable radionuclide concentrations normally acceptable for disposal at existing commercial disposal facilities. Wastes that have higher radionuclide concentrations than the Class C limits are required to be disposed in a form and by methods that

provide greater protection. The radionuclides for which concentrations are limited in Class C waste are listed in the following two tables excerpted from 10 CFR 61.55, one table for long-lived radionuclides and one for short-lived radionuclides:

TABLE 1. LONG-LIVED RADIONUCLIDES

Radionuclide	Class C Limit (Ci/meter ³)
C-14	8
C-14 in activated metal	80
Ni-59 in activated metal	220
Nb-94 in activated metal	0.2
Tc-99	3
I-129	0.08
	(nCi/g)
alpha-emitting TRU*	100
Pu-241	3500
Cm-242	20,000

TABLE 2. SHORT-LIVED RADIONUCLIDES

Radionuclide	Class C Limit (Ci/meter ³)
Ni-63	700
Ni-63 in activated metal	7000
Sr-90	7000
Cs-137	4600

*With half life greater than 5 years

To determine if a waste exceeds Class C limits, the radionuclide concentrations listed in Tables 1 and 2 are used in the following manner:

1. If the waste contains only radionuclides listed in Table 1, and the concentrations of one or more of these radionuclides exceed the Table 1 listed concentrations, the waste is greater than Class C. If the waste contains a mixture of radionuclides listed in Table 1 and none of these radionuclides alone exceeds the Table 1 listed concentrations, then the waste must be classified using the sum of fractions rule described below in item 4.
2. If the waste contains only radionuclides listed in Table 2, and the concentrations of one or more of these radionuclides exceed the listed Table 2 concentration, the waste is greater than Class C. If the waste contains a mixture of radionuclides listed in Table 2 and none of these radionuclides alone exceeds the Table 2 listed concentrations, then the waste must be classified using the sum of fractions rule described below in item 4.
3. If the waste contains radionuclides listed in both Table 1 and Table 2, classification shall be determined by considering each table independently.
4. To determine the classification of waste by the sum-of-fractions rule, the concentration of each radionuclide present in the waste and listed in Table 1, for example, is divided by the listed concentration limit for that radionuclide, using the same concentration units as those listed in the table. If the sum of the resulting fractions for Table 1 radionuclides is greater than 1, the waste exceeds Class C limits. The same sum-of-fraction rule applies to radionuclides in Table 2. For example: A waste contains Sr-90 in a concentration of 3000 Ci/meter³ and Cs-137 in a concentration of 3500 Ci/meters³. The sum of fractions for this waste is $3000/7000 + 3500/4600 = 0.43 + 0.76 = 1.2$. The waste thus exceeds Class C concentration limits.

APPENDIX B

DETERMINATION OF WASTE VOLUME, WASTE MASS, AND RADIONUCLIDE CONCENTRATIONS FOR WASTE CLASSIFICATION

In May 1983, NRC provided guidance on radioactive waste classification in "Low-Level Waste Licensing Branch Technical Position on Radioactive Waste Classification." Copies of this technical position is available from G. Roles, Mail stop SS623, U. S. Nuclear Regulatory Commission, Washington, D. C., 20555 (phone 301-427-4791).

The technical position describes basic methods that may be used to determine radionuclide concentrations for the purpose of waste classification. The concentration of a radionuclide may be determined by indirect methods such as the use of radionuclide material accountability, knowledge and control of process sources of the waste, or scaling factors that relate the inferred concentration of one radionuclide to another that is measured. Gross radioactivity measurements and direct measurement of individual radionuclides may also be used to determine concentrations.

The determination of volumes and masses for the purpose of waste classification is also discussed in the NRC technical position. Actual requirements at individual commercial disposal facilities may vary from the guidelines in the technical position. The procedures given below for determination of volume and mass will ensure a conservative estimate of waste which may be subject to the requirements of the Low-Level Radioactive Waste Policy Amendments Act of 1985.

The concentration of a radionuclide may be averaged over the volume of the waste (or weight of the waste if the units are expressed as nanocuries per gram). Radionuclide concentrations should be determined based upon the volume or weight of the final waste form except as noted below. For most wastes, procedures for determining waste volume or mass are straightforward.

For wastes where some ambiguity may exist in the proper method to determine the waste volume or mass, the following methods are recommended to ensure that the results for this survey have a consistent basis. Care needs to be taken to differentiate between the volume of the waste form and the volume of the waste container if the latter is significantly larger (e.g., greater than 10%). For example, for wastes such as ion-exchange resins or filter media contained within a disposable demineralizer or liner, the volume used for waste classification should be the volume of the contained waste rather than the gross volume of the container. The volume of cartridge filters stabilized by solidification or by emplacement within high integrity containers should be determined as calculated over the volume of the cartridge filter itself rather than the gross volume of the container. Similarly, the volume and mass considered for purposes of waste classification of dewatered ion-exchange resins and filter media placed into high integrity containers should be the volume and mass of the contained waste. Concentrations of radionuclides in absorbed liquids should be based on the volume or mass of the liquids prior to absorption, but concentrations in solidified liquids or other wet wastes such as ion-exchange resins should be based on the volume or mass of the solidified waste mass.

Classification of equipment and components is difficult due to the irregular shapes of the waste forms. Large, unpackaged pieces of equipment whose radionuclide content is primarily due to surface contamination may be classified based on the overall, or "envelope," volume of the waste. Neutron-activated metal components should be classified by the "full density volume" determined by dividing the component mass by its specific gravity.

For purposes of this survey ONLY, the classification of sealed sources that are not currently acceptable at commercial disposal facilities is to be based on the volume and mass of the sealed source itself. Source holders, containers, etc. should only be included in the calculation if the source cannot readily be removed from the holder or container. Sealed sources that are currently acceptable for disposal at commercial facilities are not to be included in this survey, even if they exceed the Class C limits using the above method of volume determination.

APPENDIX C

GLOSSARY OF TERMS

Chelating Agent

Amine polycarboxylic acids (e.g., EDTA, DTPA), hydroxy-carboxylic acids, and polycarboxylic acids (e.g., citric acid, carboic acid, and gluconic acid).

Curie (Ci)

That quantity of any radioactive isotope undergoing 3.7×10^{10} disintegrations per second. A millicurie (mCi) is one-thousandth of a curie (3.7×10^7 disintegrations per second). A microcurie (uCi) is one-millionth of a curie (3.7×10^4 disintegrations per second). A nanocurie (nCi) is one-billionth of a curie, or 37 disintegrations per second.

High-Level Radioactive Waste

According to the Nuclear Waste Policy Act, the current definition of high-level radioactive waste is "the highly radioactive material resulting from the reprocessing of spent nuclear fuel, including any liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations." and

"other highly radioactive material that the (Nuclear Regulatory) Commission, consistent with existing law, determines by rule requires permanent isolation."

The NRC has NOT, as yet designated any material as "high-level waste" under the second part of the above definition

Low-Level Radioactive Waste

The Low Level Radioactive Waste Policy Amendments Act defines low-level radioactive waste as radioactive material that--"(A) is not high-level radioactive waste, spent nuclear fuel, or byproduct material (as defined in section 11e.(2) of the Atomic Energy Act of 1954 (42 U.S.C. 2014(e)(2))); and (B) the Nuclear Regulatory Commission, consistent with existing law and in accordance with paragraph (A), classifies as low-level radioactive waste."

APPENDIX D

The primary document that provides formal designation of hazardous waste is 40 CFR 261 "Identification and Listing of Hazardous Waste." This regulation lists materials that the Environmental Protection Agency has designated as hazardous waste. However, experience has shown that few of these materials are actually ever found in combination with radioactive materials. A list of those materials that have been found in combination with radioactive materials is shown below in Table D-1. Characteristics of hazardous waste for use in determining if materials not listed in 40 CFR 261 are hazardous wastes are also listed in the regulation. These characteristics are summarized in Table D-2.

If your radioactive low-level or radium waste contains any of the hazardous wastes listed in Table D-1, please indicate in the appropriate place in the survey form. If the radioactive materials in your waste are not combined with material listed in Table D-1, please use the criteria in Table D-2 (or the complete lists in 40 CFR 261) to assist you in determining if these nonradioactive materials are hazardous wastes.

TABLE D-1. HAZARDOUS WASTES FOUND ASSOCIATED WITH RADIOACTIVE WASTE

Organic liquids (including still bottoms from recovery of these liquids)

Organic laboratory liquids such as:

- Acetone
- Aromatic hydrocarbons, including benzene, pseudocumene, toluene, and xylene
- n-Butyl alcohol
- Cyclohexanone
- Dioxane
- Ethyl acetate
- Ethyl benzene
- Ethyl ether
- Methanol
- Methyl isobutyl ketone
- Phenols

Halogenated cleaning, degreasing, and miscellaneous solvents such as:

- Carbon tetrachloride
- Chlorinated fluorocarbons
- Chlorobenzene
- Methylene chloride
- Orthodichlorobenzene
- Tetrachloroethylene
- 1,1,1-Trichloroethane
- Trichloroethylene

Waste oils or organic sludges (not listed in 40 CFR 261, but included for the purpose of this survey only)

Toxic Metals

- Barium
- Beryllium (only as dust, finely divided particulate)
- Chromium (as hexavalent chromium, not trivalent chromium; most often as Na_2CrO_4 , Na_2CrO_4 , Cr_2O_3)
- Lead (shielding or containers)
- Mercury

Other Inorganic Substances

- Asbestos (not listed in 40 CFR 261, but included for the purpose of this survey only)
- Hydrazine
- Cyanide

TABLE D-2. CHARACTERISTICS OF HAZARDOUS WASTE

Ignitability

A waste exhibits the characteristic of ignitability of a representative sample of the waste has any of the following properties:

1. It is a liquid, other than an aqueous solution containing less than 24 percent alcohol by volume and has flash point less than 60 C (140oF).
2. It is not a liquid and is capable, under standard temperature and pressure, of causing fire through friction, absorption of moisture, or spontaneous chemical changes and, when ignited, burns so vigorously and persistently that it creates a hazard.
3. It is an ignitable (flammable) compressed gas (defined in detail in 49 CFR 173.300).
4. It is an oxidizer such as a chlorate, permanganate, peroxide, or nitrate, that yields oxygen readily to stimulate the combustion of organic matter.

Corrosivity

A waste exhibits the characteristic of corrosivity of a representative sample of the waste has either of the following properties:

1. It is aqueous and has a pH less than or equal to 2 or greater than or equal to 12.5.
2. It is a liquid and corrodes steel (SAE 1020) at a rate greater than 6.35 mm (0.250 inch) per year at a test temperature of 55°C (130oF).

TABLE D-2, continued

Reactivity

A waste exhibits the characteristic of reactivity if a representative sample of the waste has any of the following properties:

1. It is normally unstable and readily undergoes violent change without detonating.
2. It reacts violently with water.
3. It forms potentially explosive mixtures with water.
4. When mixed with water, it generates toxic gases, vapors, or fumes in a quantity sufficient to present a danger to human health or the environment.
5. It is a cyanide or sulfide bearing waste that, when exposed to pH conditions between 2 and 12.5, can generate toxic gases, vapors or fumes in a quantity sufficient to present a danger to human health or the environment.
6. It is capable of detonation or explosive reaction if it is subjected to a strong initiating source or if heated under confinement.
7. It is readily capable of detonation or explosive decomposition or reaction at standard temperature and pressure.
8. It is a forbidden explosive as defined in 49 CFR 173.53 or a Class B explosive as defined in 49 CFR 173.88.

APPENDIX B
REACTOR LIST

#	REACTOR	STATE	MWt ^a	MWe ^b	MWe ^c	TYPE	START DATE	DECON DATE
1	A W VOGTLE 1	GA	3411	1069	1079	PWR	1987	2027
2	A W VOGTLE 2	GA	3411	1069	1079	PWR	1989	2029
3	AK NUCLEAR 1	AK	2568	850	836	PWR	1974	2014
4	AK NUCLEAR 2	AK	2815	912	858	PWR	1980	2020
5	BEAVER VALLEY 1	PA	2653	835	810	PWR	1977	2017
6	BEAVER VALLEY 2	PA	2652	857	833	PWR	1987	2027
7	BELLEFONTE 1	AL	3620		1177	PWR	*	
8	BELLEFONTE 2	AL	3620		1213	PWR	*	
9	BIG ROCK PT	MI	240	72	67	BWR	1962	2002
10	BRAIDWOOD 1	IL	3411	1175	1120	PWR	1988	2028
11	BRAIDWOOD 2	IL	3411	1175	1120	PWR	1988	2028
12	BROWNS FERRY 1	AL	3293	1065	1065	BWR	1974	2014
13	BROWNS FERRY 2	AL	3293	1065	1065	BWR	1975	2015
14	BROWNS FERRY 3	AL	3293	1065	1065	BWR	1977	2017
15	BRUNSWICK 1	NC	2416	821	790	BWR	1977	2017
16	BRUNSWICK 2	NC	2436	821	790	BWR	1975	2015
17	BYRON 1	IL	3411	1120	1105	PWR	1985	2025
18	BYRON 2	IL	3411	1120	1105	PWR	1987	2027
19	CALLAWAY	MO	3565	1171	1150	PWR	1985	2025
20	CALVERT CLIFFS 1	MD	2700	845	825	PWR	1975	2015
21	CALVERT CLIFFS 2	MD	2700	845	825	PWR	1977	2017
22	CARROLL CTY 2	IL			1120	PWR	*	
23	CARROLL CTY 1	IL			1120	PWR	*	
24	CATAWBA 1	SC	3411	1145	1129	PWR	1985	2025
25	CATAWBA 2	SC	3411	1145	1129	PWR	1986	2026
26	CLINTON 1	IL	2894	933	930	BWR	1987	2027
27	COMANCHE PEAK 1	TX	3411	1150	1150	PWR	1989	2029
28	COMANCHE PEAK 2	TX	3411	1150	1150	PWR	1992	2032
29	COOPER	NE	2381	778	764	BWR	1974	2014
30	CRYSTAL RIVER	FL	2544	825	821	PWR	1977	2017
31	DAVIS-BESSE 1	OH	2772	906	860	PWR	1977	2017
32	DIABLO CANYON 1	CA	3338	1086	1073	PWR	1985	2025
33	DIABLO CANYON 2	CA	3411	1119	1087	PWR	1986	2026
34	DONALD COOK 1	MI	3250	1030	1020	PWR	1975	2015
35	DONALD COOK 2	MI	3411	1100	1060	PWR	1978	2018
36	DRESDEN 1	IL	700	437		BWR		1978
37	DRESDEN 2	IL	2527	794	772	BWR	1970	2010
38	DRESDEN 3	IL	2527	794	773	BWR	1971	2011
39	DUANE ARNOLD	IA	1658	538	515	BWR	1975	2015
40	EDWIN HATCH 1	GA	2436	777	756	BWR	1975	2015
41	EDWIN HATCH 2	GA	2436	784	768	BWR	1979	2019
42	FERMI 2	MI	3293	1093	1093	BWR	1988	2028
43	FORT CALHOUN 1	NE	1500	486	478	PWR	1973	2013
44	FORT ST VRAIN	CO	842		330	HTGR	1979	2019
45	GRAND GULF 1	MS	3833	1250	1142	BWR	1985	2025
46	GRAND GULF 2	MS	3833		1142	BWR	*	

#	REACTOR	STATE	MWt ^a	MWe ^b	MWe ^c	TYPE	START DATE	DECON DATE
47	HADDAM NECK	CN	1825	582	565	PWR	1968	2008
48	HOPE CREEK 1	NJ	3293	1118	1067	BWR	1987	2027
49	HUMBOLT BAY	CA	210	65		BWR		1984
50	INDIAN PT 1	NY	615	265		PWR		1974
51	INDIAN PT 2	NY	2758	873	864	PWR	1974	2014
52	INDIAN PT 3	NY	3025	965	965	PWR	1976	2016
53	J M FARLEY 1	AL	2652	829	813	PWR	1977	2017
54	J M FARLEY 2	AL	2652	829	823	PWR	1981	2021
55	JAMES A FITZPATRICK	NY	2436	821	778	BWR	1975	2015
56	KEWAUNEE	WI	1650	535	503	PWR	1974	2014
57	LA SALLE 1	WI	3323	1122	1036	BWR	1982	2022
58	LA SALLE 2	IL	3323	1122	1036	BWR	1984	2024
59	LA CROSS	IL	165	627		BWR		1987
60	LIMERICK 1	PA	3293	1055	1055	BWR	1986	2026
61	LIMERICK 2	PA	3293	1055	1055	BWR	1990	2030
62	MAINE YANKEE	ME	2630	825	810	PWR	1972	2012
63	MILLSTONE 1	CN	2011	660	654	BWR	1970	2010
64	MILLSTONE 2	CN	2700	870	863	PWR	1975	2015
65	MILLSTONE 3	CN	3579	1150	1142	PWR	1986	2026
66	MONTICELLO	MN	1670	545	536	BWR	1971	2011
67	NINE MILE PT 1	NY	1850	620	610	BWR	1969	2009
68	NINE MILE PT 2	NY	3323	1080	1045	BWR	1988	2028
69	NORTH ANNA 1	VA	2893	907	915	PWR	1978	2018
70	NORTH ANNA 2	VA	2893	907	915	PWR	1980	2020
71	OCONEE 1	SC	2568	887	846	PWR	1973	2013
72	OCONEE 2	SC	2566	887	846	PWR	1974	2014
73	OCONEE 3	SC	2568	886	846	PWR	1974	2014
74	OYSTER CR 1	NJ	1930	650	620	BWR	1969	2009
75	PALISADES	MI	2530	805	780	PWR	1971	2011
76	PALO VERDE 1	AZ	3817	1270	1221	PWR	1986	2026
77	PALO VERDE 2	AZ	3817	1270	1221	PWR	1986	2026
78	PALO VERDE 3	AZ	3817	1270	1221	PWR	1988	2028
79	PEACH BOTTOM 3	PA	3293	1065	1035	BWR	1974	2014
80	PEACH BOTTON 2	PA	3293	1065	1051	BWR	1974	2014
81	PERRY 1	OH	3579	1265	1205	BWR	1987	2027
82	PERRY 2	OH	3679		1205	BWR	*	
83	PILGRIM 1	MA	1998	655	670	BWR	1972	2012
84	POINT BEACH 1	WI	1518	497	485	PWR	1970	2010
85	POINT BEACH 2	WI	1518	497	485	PWR	1972	2012
86	PRAIRIE ISLAND 1	MN	1650	530	503	PWR	1973	2013
87	PRAIRIE ISLAND 2	MN	1650	530	500	PWR	1974	2014
88	QUAD CITIES 1	IL	2511	789	769	BWR	1972	2012
89	QUAD CITIES 2	IL	2511	789	769	BWR	1972	2012
90	RANCHO SECO	CA	2772	918	873	PWR	1975	2015
91	RIVER BEND 1	LA	2894	936	936	BWR	1986	2026
92	ROBERT E GINNA	NY	1520	490	470	PWR	1970	2010
93	ROBINSON 2	SC	2300	700	665	PWR	1971	2011
94	S TEXAS 1	TX	3817	1250	1250	PWR	1988	2028
95	S TEXAS 2	TX	3817	1250	1250	PWR	1989	2029

#	REACTOR	STATE	MWt ^a	MWe ^b	MWe ^c	TYPE	START DATE	DECON DATE
96	SALEM 1	NJ	3411	1115	1106	PWR	1977	2017
97	SALEM 2	NJ	3411	1115	1106	PWR	1981	2021
98	SAN ONOFRE 1	CA	1347	436	436	PWR	1968	2008
99	SAN ONOFRE 2	CA	3390	1070	1070	PWR	1983	2023
100	SAN ONOFRE 3	CA	3390	1080	1080	PWR	1984	2024
101	SEABROOK	NH	3411	1150	1150	BWR	*	
102	SEQUOYAH 1	TN	3411	1148	1148	PWR	1981	2021
103	SEQUOYAH 2	TN	3411	1148	1148	PWR	1982	2022
104	SHEARON HARRIS	NC	2775	940	860	PWR	1987	2027
105	SHOREHAM	NY	2436		809	BWR	*	
106	ST LUCIE 1	FL	2700	830	839	PWR	1976	2016
107	ST LUCIE 2	FL	2700	804	839	PWR	1983	2023
108	SURRY 1	VA	2441	788	781	PWR	1972	2012
109	SURRY 2	VA	2441	788	781	PWR	1973	2013
110	SUSQUEHANNA 1	PA	3293	1065	1032	BWR	1983	2023
111	SUSQUEHANNA 2	PA	3293	1065	1038	BWR	1985	2025
112	TMI 1	PA	2535	819	808	PWR	1974	2014
113	TROJAN	OR	3411	1130	1095	PWR	1976	2016
114	TURKEY PT 3	FL	2200	693	666	PWR	1972	2012
115	TURKEY PT 4	FL	2200	693	666	PWR	1973	2013
116	VERMONT YANKEE	VT	1593	514	504	BWR	1972	2012
117	VIRGIL C. SUMMER	SC	2775	900	885	PWR	1984	2024
118	W B McGUIRE 1	NC	3411	1180	1129	PWR	1981	2021
119	W B McGUIRE 2	NC	3411	1180	1129	PWR	1984	2024
120	WATERFORD	LA	3410	1104	1075	PWR	1985	2025
121	WATTS BAR 1	TN	3411	1165	1177	PWR	1992	2032
122	WATTS BAR 2	TN	3411		1177	PWR	*	
123	WNP-1	WA	3780		1250	PWR	*	
124	WNP-2	WA	3323	1100	1095	BWR	1984	2024
125	WNP-3	WA	3800		1240	PWR	*	
126	WOLF CREEK	KA	3411	1150	1128	PWR	1985	2025
127	YANKEE-ROWE	MA	600	175	167	PWR	1961	2001
128	ZION 1	IL	3250	1085	1040	PWR	1973	2013
129	ZION 2	IL	3250	1085	1040	PWR	1974	2014

a. "Nuclear Engineering International, November 1989 Supplement," World Nuclear Industry Handbook, 1989, pp. 54-61.

b. DOE Spent Fuel Database.

c. "World List of Nuclear Power Plants," Nuclear News, Vol. 32, No. 2, February 1989, pp. 83-86.

APPENDIX C
MODELING STRUCTURE

Modeling Structure

The projected data are modeled and stored in a data base which allows access to all available GTCC LLW information. As the schematic shows, data are input into the model where three cases (low, base, and high) are developed. Three scenarios are generated in the model. Unpackaged volume data are generated first, with an annual incremental volume of waste being generated by each component listed in the model (some components are only generated once, but carried in the waste inventory). Packaging factors are applied to those data for packaged volumes, and concentration averaging coefficients are applied to the packaged data. Concentration averaging coefficients assume that a percentage of the packaged waste will not be GTCC LLW. The enclosed example shows how each of the scenarios are applied. Notice that the volume of waste increases as packaging factors are applied, and a portion of that waste volume is not considered as GTCC LLW when concentration averaging is applied.

The radionuclide activity is calculated by assuming an initial activity for each component listed, then the radionuclides are decayed at the end of the first year. The next step is to add an incremental nuclide activity, much the same as the waste volume data. At the end of that next year, the radionuclides are decayed again with a repeating cycle until the life time of the modeling period is complete.

Data listed in the data base are easily retrievable, both in the formats listed in this study and in an adhoc format.

Application

- Waste Characterization Model

Example

APPENDIX D

ABOVE CLASS C SOURCE/DEVICE INVENTORY SURVEY

ABSTRACT

The Commission directed the staff to better define the scope of the problem of surplus sources that exceed 10 CFR Part 61 Class C concentrations. This paper presents the findings of a survey and several recommendations that ultimately could reduce the likelihood of loss of sources that presently must be stored.

Based on the survey data, we estimated that there are about 27,000 greater than Class C sources are possessed by specific licensees in the United States, and about 4,600 sources that are no longer needed and should be disposed or transferred. About 2,000 of the 4,600 sources could be disposed of if a site were available. The minimum volume needed to store these sources less the transportation packaging and shielding, is about 2,800 cubic centimeters, or about the size of a one-gallon container.

Based on the staffs' knowledge of general licensees it was estimated that there are about 18,500 general licensees possessing about 65,500 GTC sources of which 9,200 sources that are no longer needed and should be disposed or transferred. About 2,700 of these 9,200 sources could be disposed of if a site was available.

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3 DATA COLLECTION.....	7
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APPENDICES

A	TYPICAL SOURCES AND USES
B	EXAMPLE SURVEY AND DEFINITION SHEET
C	LIST OF LICENSEES

1 PURPOSE

As a result of a March 15, 1989 briefing, the Commission directed the staff in a April 7, 1989 Staff Requirements Memorandum, to undertake the following:

- a. NMSS is to submit by December 29, 1989, a paper which better defines the scope of the problem of surplus sources.
- b. NMSS is to submit by May 26, 1989, for Commission review, the planned staff survey of licensees to obtain data on numbers of GTCO sources that licensees currently possess, and the numbers which are in excess of their needs.
- c. Office of the General Counsel is to submit by June 30, 1989, a legal analysis of the extent of NRC's jurisdiction and regulatory responsibilities for licensing material temporarily stored by DOE but for which NRC must ultimately license disposal.
- d. Continue to work with DOE staff and report, by June 30, 1989, on DOE readiness to accept such sources, as DOE has indicated it will.
- e. NMSS is to issue an Information Notice to inform well-logging licensees and current applicants for well-logging licenses of the potential cost of disposing of no longer needed americium-241 sources. Also, the staff should escalate its attention to sealed sources in the field to assure that licensees are responsible for ensuring safe disposition of sources.

Items (a) and (b) are the subject of this paper. Items c, d, & e, were completed by other staff papers to the commission. Subsequent to the March 1989 briefing, the staff briefed the Commission on September 21, 1989, on a study of adequacy of regulatory oversight of materials under a general license. As a result of the briefing the Commission directed the staff in the October 3, 1989, Staff Requirements Memorandum to undertake the following:

- a. Submit, by March 30, 1989, an analysis of the potential health and safety impact of devices or sources under a general license.
- b. Submit, by December 31, 1989, the status of activities related to the disposal of greater than Class-C waste.

Item (b) is included in this paper because the problems of disposal is also pertinent to about 9,200 general licensees.

2 BACKGROUND

On March 15, 1989, the staff briefed the Commission on the need to establish a procedure whereby sealed sources that exceed 10 CFR Part 61 Class C concentrations, and that cannot be disposed of at commercial low-level waste disposal facilities are transferred to the Department of Energy (DOE) for storage. The GTCC source sizes and activities were based on data obtained from one of the leading source manufacturers. Typically, source sizes within each category do not vary significantly. See Appendix A for typical sources and diagrams of use.

Several thousand NRC and Agreement State licensees possess sealed sources which when no longer needed, must be stored indefinitely, since disposal at a commercial low-level waste disposal facility is unavailable and transfer of the sources to other licensees is usually too expensive.

The Low-Level Radioactive Waste Policy Amendments Act of 1985 (Amendments Act) stipulates that the low-level waste exceeding Class C concentrations is the responsibility of the Federal government, specifically the DOE.

As part of the staffs' ongoing effort to reduce the problems associated with loss of control of sealed sources, we formally transmitted our concerns to the Commission about the economically depressed well-logging licensees possessing large quantities of americium-241/beryllium neutron sources. The staff issued an Information Notice to well-logging licensees on March 30, 1989. NRC regulations force the licensee to store unwanted GTCC sources; therefore, the staff is emphasizing licensee responsibility for ensuring safe disposition of all sealed sources. The staff is also in the contract negotiation phase of soliciting a contract to survey 3,000 general licensees. The experience gained by performing the survey and the findings will be used to establish a permanent mail survey program.

Furthermore, there are five Agreement States that have taken possession of 15 abandoned sealed sources, in the interest of public health and safety. Also, the State of Oklahoma, a non-Agreement State has taken possession of two sources at the request of the NRC. These States cannot dispose of the materials and are forced to store the sealed sources or sell the usable sources to licensees at a public auction.

In estimating the numbers of NRC licensees affected, the staff used data from the Licensing Tracking System (LTS) and the General License Data Base (GLDB). The LTS system is used by NRC Headquarters and the regional offices to manage licensing and inspection of about 8200 byproduct material licensees. The GLDB system is used by NRC headquarters to manage vendors quarterly reports of transfers. Figures 1 through 4 show the approximate number of NRC specific licensees in each category that are expected to have GTCC C sources. For the purposes of this paper, GTCC sources and GTCC devices or gauges are used interchangeably because a device typically contains a sealed source. The numbers inside the boxes are the approximate number of licensees; the number on top of the boxes are the percentages. For example, Figure 1 shows 120 well-logging licensees according to the program codes listed for well logging licensees in the LTS system. Of the 120 licensees 102 (85%) are expected to have GTCC sources or devices.

PERCENTAGE OF NRC LICENSEES HAVING GREATER THAN CLASS C SEALED SOURCES

Note: Numbers in smaller boxes represent affected licensees.

Figure 1

PERCENTAGE OF NRC LICENSEES HAVING GREATER THAN CLASS C SEALED SOURCES

Note: Numbers in smaller boxes represent affected licensees.

Figure 2

PERCENTAGE OF NRC LICENSEES HAVING GREATER THAN CLASS C SEALED SOURCES

Note: Numbers in smaller boxes represent affected licensees.

Figure 3

PERCENTAGE OF NRC LICENSEES HAVING GREATER THAN CLASS C SEALED SOURCES

Note: Numbers in smaller boxes represent affected licensees

Figure 4

A query of the GLDB system showed about 9,200 out of about 41,000 (or 22%) general licensees are expected to have GTCC sources contained in devices.

3 DATA COLLECTION

Licenses authorized to possess radioactive materials are issued under 10 CFR Parts 30, 40, and 70, for periods of five years, after which renewals are required. License amendments are also required anytime a licensee makes a change to its program. In any one year, at least 20 percent of all licenses authorized to possess GTCC sources will be amended or renewed. The staff designed a survey that is being conducted over at least the next year, during the normal course of reviewing these pending licensing cases. Because license types that come up for renewal or that submit amendments are randomly spread over all license categories, the survey should yield a statistically valid population sample. NRC Regional and Agreement State licensing personnel administered the survey which started in July 1989, and obtained the information delineated on the attached Policy and Guidance Directive sent to the regions.

The staff chose this method because the existing Office of Management and Budget (OMB) Clearance control No. 31500017, as set forth in 10 CFR Part 30, Subsection 30.8(a), could be used to obtain the information. A larger survey of all licensees would require a special OMB Clearance and require the use of a contractor, which would take several months. Using the approach chosen, the staff was able to obtain good results in a short time.

On March 27, 1989, regional comments were requested on the GTCC source/device inventory survey. Comments from the regions were taken in consideration and a new survey form was drafted. See Appendix B for an example survey form and definition sheet. It was decided by the IMNS staff that the request for survey data would go out as a Policy and Guidance Directive.

On June 30, 1989, by memorandum from the Office of the Secretary, the Commission approved the proposed policy and Guidance Directive and Survey contingent upon inclusion of their modifications. Some of the Commissions modifications could not be included in the inventory sheet because they deviated from an earlier Staff Requirements memorandum to scope the problem of GTCC sources.

On July 17, 1989, the revised Policy and Guidance Directive FC 89-1: "Authorizing the conduct of Greater than Class C Sealed Source Survey," was sent out to all the regional offices. At the same time a memorandum was sent to State Local and Indian Tribe Program (SLITP), requesting that the Agreement States perform a similar survey of their licensees. SLITP sent out an All Agreement States Letter dated July 26, 1989, requesting the Agreement States to send the questionnaire to applicable licensees in their States.

General licensees also possess devices containing GTCC sources. A previous Commission paper #89-289 discusses general licenses in more detail than will be discussed here. Several years ago, the staff established the General License Data Base (GLDB) to allow NRC to manage information about general licensees. The GLDB contains about 41,000 general licensees, excluding those licensees that use beta backscatter gauges, static eliminators or devices containing microcurie quantities of byproduct material. The GLDB system was queried to collect the number of licensees that have GTCC sources. No CNB approval was needed for this type of data collection effort.

4 DATA INPUT

Two options were considered for data input. The first option was to manually go through each survey and count the important pieces of information. This option was ruled out due to the amount of time it would take to analyze all of the surveys and the associated human errors. Also, if an additional data analysis were needed, that meant that all the surveys would have to be evaluated again.

Since we only had a short period of time to get some initial results, and initially were unaware of the type of information we needed to analyze, we chose to put the surveys on a computer-based program. Once the data were entered into the database we could extract any information that we wanted to analyze. The software we chose was "dBASE III." A user friendly computer program was written in the dBASE language since the technical background of the persons entering the data was unknown. The computer program was menu-driven for ease of operation (see Figures 5 and 6). Under the "ADD NEW SURVEY" screen the following screens were set up to look similar to the actual survey (see Figures 7 through 9).

Any data that were to be analyzed in the database were screen edited before they could be appended to the database. For example, on the "ADD LICENSE PROGRAM" screen (Figure 10) if the licensees' license number, name, or State were not entered, the following error message would appear "*** INVALID DATA-RETYPE ***," and the cursor would highlight the invalid data. The user would have to retype correct information, or else the user could not continue to add the data. These screen edit checks were also incorporated into the other screens as well. For the "ANSWER THE FOLLOWING QUESTIONS" screen (Figure 8) you had to enter at least one disposal method, and answer "Y" for yes or "N" for no to the questions. For the "ADD LICENSEES' SOURCES/DEVICES" screen (Figure 9) you had to enter a "D" for device or "S" for source under type; answer with a valid isotope (Am-241, Cm-244, Cs-137, Pu-238, Pu-239, or Sr-90); answer with a valid unit (mCi, or Ci); answer with a valid typical use, according to the definitions on the inventory sheet (A-I, or O); answer to the active or the inactive field--if the inactive field were chosen you had to answer with a valid inactive code according to the inventory sheet (D, L, T, DT, X, or O); and answer "Y" for yes or "N" for no to the comments question. These edit checks were used to minimize errors, keep data consistent, and were necessary to be able to perform certain indexes or sorts (i.e., required State field).

Figure 5

Figure 6

ADD LICENSEE PROGRAM

LICENSE #: 01-00000-02G

LICENSEE'S NAME: EXAMPLE SURVEY

CONTACT NAME: TOM RICH

TITLE: PROGRAMMER

DEPARTMENT: NMSS

STREET:

CITY: WASHINGTON

STATE: DC ZIP CODE: 20555

PHONE NUMBER: (301) 492-0511 EXT.:

TYPE A TO ADD, C TO CHANGE, OR Q TO QUIT A

TYPE IN DATA AND HIT RETURN

Figure 7

ANSWER THE FOLLOWING QUESTIONS

2) How do you dispose of your sources and/or devices?

Manufacturer: X

Transfer to another licensee: (TYPE ANY LETTER IN THE APPROPRIATE BOX)

Have not disposed any:

Other:

If other, please elaborate:

3)a. Are you able to find and use an authorized recipient to purchase, dispose, or store any sources and/or devices that you no longer want (Y/N)? Y

3)b. Are there any difficulties in using this authorized recipient (Y/N)? N

4) Additional comments (Y/N)? N

TYPE A TO ADD, C TO CHANGE, OR Q TO QUIT A

TYPE IN DATA AND HIT RETURN

Figure 8

ADD LICENSEE'S SOURCES/DEVICES

LICENSE #: 01-00000-02G

MANUFACTURER: TROXLER INC.

MODEL #: 3411B

TYPE (D OR S): D

ISOTOPE (I.E. AM-241 OR AM241): AM-241

ACTIVITY: 50.000 UNIT (MCI OR CI): MCI

TYPICAL USE (A-I, OR O): E

ACTIVE (Type X in the box): X INACTIVE (D,L,T,DT,X OR O):

COMMENTS (Y/N): N SOURCE COUNT = 1

TYPE A TO ADD, C TO CHANGE, OR Q TO QUIT A

TYPE IN DATA AND HIT RETURN

Figure 9

ADD LICENSEE PROGRAM

LICENSE #: 01-00000-02G

LICENSEE'S NAME: -

CONTACT NAME: TOM RICH

TITLE: PROGRAMMER

DEPARTMENT: NMSS

STREET:

CITY:

STATE: MD ZIP CODE:

PHONE NUMBER: (301)492-0511 EXT.: 20511

*** INVALID DATA - RETYPE ***

TYPE IN DATA AND HIT RETURN

Figure 10

Besides the edit checks to keep the data consistent, the program generated a list of vendors and the model numbers they manufactured. This list is obtainable from the "GTCC SUMMARY PRINTOUT" menu Option 3 (see Figure 6). This list was periodically updated and given to the data entry people during data input.

There were additional quality control and assurance checks incorporated into the system. The data entry people were given only the option to add data; they were not allowed to modify the data once they were added. Each data entry person was given their own disk to add data to. This allowed multiple data-entry people to input surveys at the same time. Periodically, the person in charge of the data entry would take the data-entry persons' disk, edit their data for any obvious errors, and append the data to the main database. Therefore, data entry people could not accidentally damage the main database. In addition, the main database was backed up daily on two separate systems, in case one system failed or was tampered with.

5 FINDINGS

The findings are summarized in Tables 1 to 3. The number of specific licensees surveyed was 2,202. Of these 2,202 licensees, 1,332 licensees had 7,340 GTCC sources or devices.

Of the 1,332 specific licensees having GTCC sources or devices, 54 (4%) were having difficulty in disposing of their sources or devices. The licensees have 715 sources in storage (9.7%). The licensees are having difficulties in disposing due to cost, unable to locate a manufacturer to accept them, current manufacturer refusal to accept them back, and so on. Of the remaining licensees, 581 (43.6%) have not tried to dispose of their GTCC sources or devices.

Of the 7,340 GTCC sources or devices, licensees want to dispose or transfer 482 (6.6%) GTCC sources or devices.

The bulk of these sources or devices was americium-241 (6,728 or 92%). These americium-241 sources or devices account for an approximate total activity of 18,741 curies, of which 5,309 curies are surplus (28.3%). Licensees want to dispose of or transfer approximately 804 curies (4.3%) of americium-241 GTCC sources or devices.

There were 122 GTCC cesium-137 sources or devices that had an approximate total activity of 233,402 curies, of which 21,673 curies (9%) were surplus. Licensees want to transfer or dispose of 67,496 curies (28.9%) of cesium-137 GTCC sources or devices.

Totals of approximately 420 curies (23.6%) of plutonium-238 and 37 curies of plutonium-239 (5.4%) were surplus, and the licensees want to dispose of approximately 722 curies (40.9%) of plutonium-238 and 85 curies (13.5%) of plutonium-239.

There was an approximate total of 2 curies of curium-244.

GTCC SURVEY SUMMARY

THE NUMBER OF LICENSEES SURVEYED HAVING GTCC SOURCES/DEVICES IS 1692 ←

NUMBER OF SEALED SOURCES/DEVICES ABOVE CLASS C

	<u>DEVICES</u>	<u>SOURCES</u>	<u>SUBTOTAL</u>
AMERICIUM-241	5152	2714	7866
CESIUM-137	65	44	109
CURIUM-244	6	33	39
PLUTONIUM-238	64	222	286
PLUTONIUM-239	36	247	283
STRONTIUM-90	<u>1</u>	<u>0</u>	<u>1</u>
	5324	3260	TOTAL 8584 ←

HOW LICENSEE'S DISPOSE OF THEIR SOURCES/DEVICES

<u>TRANSFER BACK TO THE MANUFACTURER</u>	<u>TRANSFER TO ANOTHER SPECIFIC LICENSEE</u>	<u>LICENSEES THAT HAVE NOT DISPOSED ANY</u>
825	203	717

(NOTE: LICENSEES MAY USE SEVERAL METHODS OF DISPOSAL)

NUMBER OF LICENSEES HAVING DIFFICULTIES IN DISPOSING THEIR
SEALED SOURCES AND/OR DEVICES TO AN AUTHORIZED RECIPIENT IS 70 ←

NUMBER OF INACTIVE SOURCES/DEVICES

	<u>DEVICES</u>	<u>SOURCES</u>	<u>SUBTOTAL</u>
DAMAGED	10	12	22
LOST	3	1	4
OTHER	39	33	72
STORED FOR FUTURE USE	207	577	784
WANTS TO DISPOSE OR TRANSFER	129	372	501
DAMAGED AND WANTS TO DISPOSE OR TRANSFER	<u>1</u>	<u>10</u>	<u>11</u>
	389	1005	TOTAL 1394 ←

GTCC SURVEY SUMMARY

THE NUMBER OF LICENSEES SURVEYED HAVING GTCC SOURCES/DEVICES IS 1692 ←

NUMBER OF GTCC SEALED SOURCES/DEVICES IN EACH CATEGORY

	<u>DEVICES</u>	<u>SOURCES</u>	<u>SUBTOTAL</u>
WELL LOGGING	51	1763	1814
IRRADIATORS	67	87	154
TELETHERAPY	3	1	4
X-RAY FLUORESCENCE	51	91	142
PORTABLE GAUGES	4636	238	4874
FIXED GAUGES	440	192	632
PACEMAKERS	29	0	29
BROAD LICENSES	11	128	139
WASTE BROKERS	0	92	92
OTHERS	<u>36</u>	<u>668</u>	<u>704</u>
	5324	3260	TOTAL 8584 ←

ACTIVITY OF GTCC SEALED SOURCES/DEVICES IN EACH CATEGORY

	<u>ACTIVITY (CURIES)</u>					
	<u>Am-241</u>	<u>Cs-137</u>	<u>Cm-244</u>	<u>Pu-238</u>	<u>Pu-239</u>	<u>Sr-90</u>
WELL LOGGING	15231.202	5010.385	0.000	541.635	0.000	0.000
IRRADIATORS	81.390	267193.610	0.000	163.340	155.113	0.000
TELETHERAPY	0.000	7542.000	0.000	0.000	0.000	0.000
X-RAY FLUORESCENCE	47.272	0.000	1.520	2.090	0.030	0.000
PORTABLE GAUGES	335.517	0.134	0.030	0.150	5.500	0.000
FIXED GAUGES	515.572	24.400	0.400	228.460	0.000	0.010
PACEMAKERS	55.150	0.000	0.082	49.087	127.956	0.000
BROAD LICENSES	0.000	0.000	0.000	130.110	0.000	0.000
WASTE BROKERS	1642.184	0.000	0.000	375.000	0.000	0.000
OTHERS	<u>1315.036</u>	<u>1220.000</u>	<u>0.200</u>	<u>507.078</u>	<u>388.743</u>	<u>0.000</u>
TOTALS	19223.324	280990.529	2.232	1996.950	677.342	0.010

GTCC SURVEY SUMMARY

THE NUMBER OF LICENSEES SURVEYED HAVING GTCC SOURCES/DEVICES IS 1692 ←

ACTIVITY OF INACTIVE GTCC SEALED SOURCES/DEVICES

	ACTIVITY (CURIES)					
	<u>Am-241</u>	<u>Cs-137</u>	<u>Cm-244</u>	<u>Pu-238</u>	<u>Pu-239</u>	<u>Sr-90</u>
DAMAGED	12.514	3212.000	0.030	0.350	0.000	0.000
LOST	3.410	0.000	0.000	0.000	0.000	0.000
DISPOSE OR TRANSFER	812.013	64284.000	0.000	721.992	86.503	0.000
DAMAGED - TRANSFER	2.000	3212.000	0.030	0.350	0.000	0.000
SURPLUS	5339.216	21673.000	0.000	433.630	42.267	0.000
OTHER	<u>34.717</u>	<u>0.000</u>	<u>0.000</u>	<u>6.230</u>	<u>0.432</u>	<u>0.000</u>
TOTALS	<u>6203.870</u>	<u>92381.000</u>	<u>0.060</u>	<u>1162.552</u>	<u>129.202</u>	<u>0.000</u>

LICENSEES THAT WANT TO DISPOSE OR TRANSFER THEIR GTCC SOURCES/DEVICES

LICENSE #	MANUFACTURER	MODEL NUMBER	TYPE	USE	ISOTOPE	ACTIVITY	CMTS
ORE-902881	TROXLER INC.	2401	S	E	AM241	50.000 MCI	Y
ORE-0295-1	OHMART CORP.	A-5799	S	F	AM241	1000.000 MCI	Y
ORE-0026-1	TEXAS NUCLEAR	9256	D	O	PU238	30.000 MCI	Y
	TEXAS NUCLEAR	9256	D	O	PU238	30.000 MCI	N
ORE-901181	AMERSHAM CORP.	AMC.36	S	F	AM241	100.000 MCI	Y
SNM 204	MONSANTO RESEARCH CORP.	M-402-403-0-0	S	O	PU239	5.000 CI	N
0014-2-85AAB	TROXLER INC.	1207 (OLD 217-105A)	D	E	AM241	100.000 MCI	N
	TROXLER INC.	1255	D	E	AM241	100.000 MCI	N
SNM-255	MONSANTO RESEARCH CORP.	M17-1	S	O	PU239	0.930 CI	N
	MONSANTO RESEARCH CORP.	M17-2	S	O	PU239	0.930 CI	N
	MONSANTO RESEARCH CORP.	M17-3	S	O	PU239	0.930 CI	N
	MONSANTO RESEARCH CORP.	M17-4	S	O	PU239	0.930 CI	N
	MONSANTO RESEARCH CORP.	M17-5	S	O	PU239	0.930 CI	N
	MONSANTO RESEARCH CORP.	M17-6	S	O	PU239	0.930 CI	N
	MONSANTO RESEARCH CORP.	M17-7	S	O	PU239	0.930 CI	N
	MONSANTO RESEARCH CORP.	M17-8	S	O	PU239	0.930 CI	N
	MONSANTO RESEARCH CORP.	M17-9	S	O	PU239	0.930 CI	N
	MONSANTO RESEARCH CORP.	M17-10	S	O	PU239	0.930 CI	N
	MONSANTO RESEARCH CORP.	M-14-38	S	O	PU239	47.000 MCI	N
	MONSANTO RESEARCH CORP.	M-14-39	S	O	PU239	46.000 MCI	N
190R	MONSANTO RESEARCH CORP.	UNKNOWN	D	G	AM241	50.000 MCI	N
	MONSANTO RESEARCH CORP.	UNKNOWN	D	G	AM241	4500.000 MCI	N
03-14-0014-1	TROXLER INC.	2401	D	E	AM241	50.000 MCI	N
	TROXLER INC.	2401	D	E	AM241	50.000 MCI	N
	TROXLER INC.	2401	D	E	AM241	50.000 MCI	N

Table 2

'TWR

Table 2 (continued)

LICENSEES THAT WANT TO DISPOSE OR TRANSFER THEIR GTCC SOURCES/DEVICES

[illegible]

Table 2 (continued)

LICENSEES THAT WANT TO DISPOSE OR TRANSFER THEIR GTCC SOURCES/DEVICES

[illegible]

Table 2 (continued)

'I'WR

Table 2 (continued)

2M1,

Table 2 (continued)

LICENSEES THAT WANT TO DISPOSE OR TRANSFER THEIR GTCC SOURCES/DEVICES

LICENSE #	MANUFACTURER	MODEL NUMBER	TYPE USE	ISOTOPE	ACTIVITY	CMTS
L00446	GULF NUCLEAR INC.	71-2A	S	A AM241	20.000 CI	Y
	GULF NUCLEAR INC.	71-2A	S	A AM241	20.000 CI	Y
	GULF NUCLEAR INC.	71-2A	S	A AM241	20.000 CI	Y
	GULF NUCLEAR INC.	71-2A	S	A AM241	20.000 CI	Y
	GULF NUCLEAR INC.	71-2A	S	A AM241	20.000 CI	Y
	GULF NUCLEAR INC.	71-2A	S	A AM241	20.000 CI	Y
	GULF NUCLEAR INC.	71-2A	S	A AM241	20.000 CI	Y
	GULF NUCLEAR INC.	71-2A	S	A AM241	20.000 CI	Y
	GULF NUCLEAR INC.	71-2A	S	A AM241	20.000 CI	Y
	GULF NUCLEAR INC.	71-2A	S	A AM241	20.000 CI	Y
	GULF NUCLEAR INC.	71-2A	S	A AM241	20.000 CI	Y
	GULF NUCLEAR INC.	71-2A	S	A AM241	20.000 CI	Y
	GULF NUCLEAR INC.	71-2A	S	A AM241	20.000 CI	Y
	GULF NUCLEAR INC.	71-2A	S	A AM241	20.000 CI	Y
SNM-561	UNKNOWN	930 AM 238	S	O PU-238	41.760 CI	Y
	UNKNOWN	AMBE 58	S	O PU-238	41.760 CI	Y
SNM-362	UNKNOWN	AMBE 59	S	A AM-241	40.000 MCI	Y
	UNKNOWN	UNKNOWN	S	G PU-239	599.000 MCI	N
205-002-83	MONSANTO RESEARCH CORP.	M825	S	G PU238	1.000 CI	N
	MONSANTO RESEARCH CORP.	M-826	S	G PU238	1.000 CI	N
	MONSANTO RESEARCH CORP.	M827	S	G PU238	1.000 CI	N
	MONSANTO RESEARCH CORP.	M828	S	G PU238	1.000 CI	N
	MONSANTO RESEARCH CORP.	M829	S	G PU238	1.000 CI	N
R-47100-F89	TROXLER INC.	3411B	D	E AM-241	44.000 MCI	N
R-7108-J8	TEXAS NUCLEAR	9256	D	D PU-238	30.000 MCI	Y
SNM-1405	UNKNOWN	UNKNOWN	S	O PU-239	310.000 MCI	N
	OAK RIDGE NATIONAL LAB	UNKNOWN	S	O PU-239	60.000 MCI	N
	MONSANTO RESEARCH CORP.	MRC-N-SS-W-PU8BE-XX	S	O PU-239	17.400 CI	N

Table 2 (continued)

LICENSEES THAT WANT TO DISPOSE OR TRANSFER THEIR GTCC SOURCES/DEVICES

LICENSE #	MANUFACTURER	MODEL NUMBER	TYPE	USE	ISOTOPE	ACTIVITY	CMTS
LA0001-1.01	UNKNOWN	UNKNOWN	S	G	AM241	98.000 MCI	N
	UNKNOWN	UNKNOWN	S	G	AM241	98.000 MCI	N
	UNKNOWN	UNKNOWN	S	G	PU239	999.000 MCI	N
	UNKNOWN	UNKNOWN	S	G	PU239	999.000 MCI	N
	UNKNOWN	UNKNOWN	S	G	PU239	999.000 MCI	N
	UNKNOWN	UNKNOWN	S	G	PU239	999.000 MCI	N
	UNKNOWN	UNKNOWN	S	G	PU239	999.000 MCI	N
	UNKNOWN	UNKNOWN	S	G	PU239	999.000 MCI	N
	UNKNOWN	UNKNOWN	S	G	PU239	999.000 MCI	N
	UNKNOWN	UNKNOWN	S	G	PU239	999.000 MCI	N
ND3301376-01	TROXLER INC.	3216	D	E	AM241	40.000 MCI	Y
	TROXLER INC.	2401	D	E	AM241	50.000 MCI	Y
	TROXLER INC.	3241	D	E	AM241	100.000 MCI	Y
ND3306847-03	TROXLER INC.	3411B	D	E	AM241	50.000 MCI	Y
	TROXLER INC.	3411B	D	E	AM241	50.000 MCI	Y
	TROXLER INC.	3411B	D	E	AM241	50.000 MCI	Y
	TROXLER INC.	3411B	D	E	AM241	50.000 MCI	Y
	TROXLER INC.	3411B	D	E	AM241	50.000 MCI	Y
	TROXLER INC.	3411B	D	E	AM241	50.000 MCI	Y
	TROXLER INC.	3411B	D	E	AM241	50.000 MCI	Y
	TROXLER INC.	3411B	D	E	AM241	50.000 MCI	Y
	TROXLER INC.	3411B	D	E	AM241	50.000 MCI	Y
	TROXLER INC.	3411B	D	E	AM241	50.000 MCI	Y
	TROXLER INC.	3411B	D	E	AM241	50.000 MCI	Y
	TROXLER INC.	3411B	D	E	AM241	50.000 MCI	Y
	TROXLER INC.	3411B	D	E	AM241	50.000 MCI	Y
	TROXLER INC.	3411B	D	E	AM241	50.000 MCI	Y
	TROXLER INC.	3411B	D	E	AM241	50.000 MCI	Y
	TROXLER INC.	3411B	D	E	AM241	50.000 MCI	Y
	TROXLER INC.	3411B	D	E	AM241	50.000 MCI	Y
MS-273-01	FRIESEKE & HOEPFRIER	FH46T	D	F	AM-241	250.000 MCI	Y

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Table 2 (continued)

LICENSEES THAT WANT TO DISPOSE OR TRANSFER THEIR GTCC SOURCES/DEVICES

LICENSE #	MANUFACTURER	MODEL NUMBER	TYPE	USE	ISOTOPE	ACTIVITY	CMTS
0095157RD2	GENERAL ELECTRIC	DWG. 135B9598	S	O	AM241	5.880 CI	Y
L01095	GAMMATRON	AN-HP	S	E	AM241	200.000 MCI	N
L01381	AMERSHAM CORP.	PPC	S	D	PU238	30.000 MCI	N
	AMERSHAM CORP.	PPC	S	D	PU238	30.000 MCI	N
	AMERSHAM CORP.	PPC	S	D	PU238	30.000 MCI	N
	AMERSHAM CORP.	PPC	S	D	PU238	30.000 MCI	N
	AMERSHAM CORP.	PPC	S	D	PU238	30.000 MCI	N
	AMERSHAM CORP.	PPC	S	D	PU238	30.000 MCI	N
	AMERSHAM CORP.	PPC	S	D	PU238	30.000 MCI	N
	MONSANTO RESEARCH CORP.	MRC-314	S	O	PU238	37.000 CI	N
	MONSANTO RESEARCH CORP.	MRC-N-SS-W	S	O	PU238	23.000 CI	N
	AMERSHAM CORP.	PPC	S	D	PU238	75.000 MCI	N
	AMERSHAM CORP.	PPC	S	D	PU238	43.000 MCI	N
	AMERSHAM CORP.	PPC	S	D	PU238	43.000 MCI	N
	AMERSHAM CORP.	PPC	S	D	PU238	43.000 MCI	N
	AMERSHAM CORP.	PPC	S	D	PU238	43.000 MCI	N
	AMERSHAM CORP.	PPC	S	D	PU238	43.000 MCI	N
	AMERSHAM CORP.	PPC	S	D	PU238	43.000 MCI	N
	AMERSHAM CORP.	PPC	S	D	PU238	43.000 MCI	N
	AMERSHAM CORP.	PPC	S	D	PU238	43.000 MCI	N
L04221	GAMMA INDUSTRIES	NB-HP	S	A	AM241	1.000 CI	N
37-185-4	AMERSHAM CORP.	AMN. PE1	S	E	AM241	98.500 MCI	N
	AMERSHAM CORP.	AMN. PE1	S	E	AM241	98.500 MCI	N
	AMERSHAM CORP.	AMN. PE1	S	E	AM241	98.500 MCI	N
	AMERSHAM CORP.	AMN. PE1	S	E	AM241	98.500 MCI	N
SNM-142	TROXLER INC.	105-217	D	E	AM241	100.000 MCI	N
	TROXLER INC.	105A/S6-AS	D	E	AM241	100.000 MCI	N

LICENSEES THAT WANT TO DISPOSE OR TRANSFER THEIR GTCC SOURCES/DEVICES

LICENSE #	MANUFACTURER	MODEL NUMBER	TYPE	USE	ISOTOPE	ACTIVITY	QMTS
SNM-142	TROXLER INC. TROXLER INC.	104 1255	D D	E E	AM241 AM241	100.000 MCI 100.000 MCI	N N
SNM-910	MONSANTO RESEARCH CORP. MONSANTO RESEARCH CORP. MONSANTO RESEARCH CORP. MONSANTO RESEARCH CORP. MONSANTO RESEARCH CORP. UNKNOWN UNKNOWN UNKNOWN	M-1149 M-91 M-96 M-97 MRC-N-SS-W-330 W320C29 W320C30 W320C31	S S S S S S S S	O O O O O O O O	PU239 PU239 PU239 PU239 PU239 PU239 PU239 PU239	5.910 CI 0.900 CI 0.130 CI 0.130 CI 2.900 CI 1.820 CI 1.820 CI 1.820 CI	N N N N N N N N
50-18441-01	NSSI NSSI NSSI	AN-HP AN-HP AN-HP	S S S	A A A	AM241 AM241 AM241	4.500 CI 4.500 CI 4.500 CI	N N N
05-26899-01	CPN	CPN-131	S	E	AM241	50.000 MCI	N
MS-119-01	MONSANTO RESEARCH CORP.	MRC-G-SS-WAM	S	O	AM241	300.000 MCI	N
MS-637-01	TROXLER INC.	3440	D	E	AM241	44.000 MCI	Y
WN-M0151-01	PICKER CORP.	3800B-1	S	C	CS137	1240.000 CI	N
UT-1800289	TROXLER INC. TROXLER INC.	A-102451 1-102451	D D	E E	AM-241 AM241	50.000 MCI 50.000 MCI	N N
LO-1886	CPN ATOMIC LABS	MC-1 UNKNOWN	D S	E O	AM241 PU238	50.000 MCI 3.000 CI	Y Y
LO1373	SOILTEST INC. SOILTEST INC. SOILTEST INC.	NIC-5 SERIES NIC-5 SERIES NIC-5 SERIES	D D D	E E E	AM241 AM241 AM241	60.000 MCI 60.000 MCI 60.000 MCI	N N N
LO2116	AMERSHAM CORP.	AMN.PE1	S	O	AM241	594.000 MCI	N

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Table 2 (continued)

LICENSEES THAT WANT TO DISPOSE OR TRANSFER THEIR GTCC SOURCES/DEVICES

LICENSE #	MANUFACTURER	MODEL NUMBER	TYPE	USE	ISOTOPE	ACTIVITY	CMTS
L02116	AMERSHAM CORP. AMERSHAM CORP. TEXAS NUCLEAR	UNKNOWN UNKNOWN 9254	S S D	O O O	AM241 AM241 PU-238	543.400 MCI 543.400 MCI 28.200 MCI	N N N
14-17	CPN	503	D	E	AM241	50.000 MCI	N
L00197	TROXLER INC. TROXLER INC. TROXLER INC.	3411B 3411B 3411B	D D D	E E E	AM241 AM241 AM241	40.000 MCI 40.000 MCI 40.000 MCI	N N N
3-22	SOILTEST INC.	NIC-5DT	D	E	AM-241	55.000 MCI	N
7-205	CPN	503	D	E	AM-241	50.000 MCI	Y
7-3	CPN CPN	503 503	D D	E E	AM-241 AM-241	50.000 MCI 50.000 MCI	N N
L01667	NEEI GULF NUCLEAR INC.	71-1 71-1	S S	A A	AM-241 AM-241	3.000 CI 3.000 CI	N N
203-21-72	SOILTEST INC.	UNKNOWN	D	E	AM-241	50.000 MCI	N
4-1584	MONSANTO RESEARCH CORP.	2-259F	S	O	PU-239	2.000 CI	N
L02148	GAMMA INDUSTRIES GAMMA INDUSTRIES GAMMA INDUSTRIES GAMMA INDUSTRIES MONSANTO RESEARCH CORP. MONSANTO RESEARCH CORP. MONSANTO RESEARCH CORP. MONSANTO RESEARCH CORP. MONSANTO RESEARCH CORP.	NB-HP NB-HP NB-HP NB-HP NS-WELEX NS-WELEX UNKNOWN UNKNOWN UNKNOWN	S S S S S S S S S	O O O O A A O O O	AM-241 AM-241 AM-241 AM-241 AM-241 AM-241 AM-241 AM-241 AM-241	5.000 CI 5.000 CI 5.000 CI 5.000 CI 4.000 CI 4.000 CI 4.000 CI 4.000 CI 4.000 CI	N N N N N N N N N

Table 2 (continued)

LICENSEES THAT WANT TO DISPOSE OR TRANSFER THEIR GTCC SOURCES/DEVICES

LICENSE #	MANUFACTURER	MODEL NUMBER	TYPE	USE	ISOTOPE	ACTIVITY	CMTS
L02148	MONSANTO RESEARCH CORP. MONSANTO RESEARCH CORP.	UNKNOWN UNKNOWN	S	O	AM-241 AM-241	4.000 CI 4.000 CI	N N
L03354	GULF NUCLEAR INC.	71-1	S	A	AM-241	3.000 CI	N
L02148	GAMMATRON	AN-HP	S	O	AM241	18.000 CI	N
R-43001-C90	OHMART CORP.	SR-2	D	F	AM-241	2.500 CI	N
R-19017-J89	TROXLER INC. TROXLER INC. TROXLER INC. TROXLER INC. TROXLER INC. TROXLER INC. TROXLER INC. TROXLER INC. TROXLER INC. TROXLER INC. TROXLER INC. TROXLER INC. TROXLER INC. TROXLER INC. TROXLER INC. TROXLER INC.	2401 2401 2401 2401 2401 2401 2401 2401 2401 2401 2401 2401 2401 2401 2401 2401	D D D D D D D D D D D D D D D D	E E E E E E E E E E E E E E E E	AM-241 AM-241 AM-241 AM-241 AM-241 AM-241 AM-241 AM-241 AM-241 AM-241 AM-241 AM-241 AM-241 AM-241 AM-241 AM-241	50.000 MCI 50.000 MCI 50.000 MCI 50.000 MCI 50.000 MCI 50.000 MCI 50.000 MCI 50.000 MCI 50.000 MCI 50.000 MCI 50.000 MCI 50.000 MCI 50.000 MCI 50.000 MCI 50.000 MCI 50.000 MCI	N N N N N N N N N N N N N N N N N
L03121	ATOMIC ACCESSORIES	ND-217	S	B	PU239	2.000 CI	N
L02113	GAMMATRON GAMMATRON GAMMATRON GAMMA INDUSTRIES GAMMA INDUSTRIES GAMMATRON GAMMATRON	AN-HP AN-HP AN-HP NB-HP NB-HP AN-HP AN-HP	S S S S S S S	A A A A A A A	AM241 AM241 AM241 AM241 AM241 AM241 AM241	33.000 MCI 33.000 MCI 33.000 MCI 5.000 CI 5.000 CI 5.000 CI 5.000 CI	N N N N N N N

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Table 2 (continued)

LICENSEES THAT WANT TO DISPOSE OR TRANSFER THEIR GTCC SOURCES/DEVICES

LICENSE #	MANUFACTURER	MODEL NUMBER	TYPE	USE	ISOTOPE	ACTIVITY	CMTS
L02113	GAMMATRON	AN-HP	S	A	AM241	5.000 CI	N
	GAMMATRON	AN-HP	S	A	AM241	5.000 CI	N
	GAMMATRON	AN-HP	S	A	AM241	5.000 CI	N
	GAMMATRON	AN-HP	S	A	AM241	5.000 CI	N
	GAMMATRON	AN-HP	S	A	AM241	5.000 CI	N
COLO 285-01E	GAMMA INDUSTRIES	NB-HP	S	A	AM-241	1.000 CI	N
	GAMMA INDUSTRIES	NB-HP	S	A	AM-241	1.000 CI	N
	GAMMA INDUSTRIES	NB-HP	S	A	AM-241	1.000 CI	N
	GAMMA INDUSTRIES	NB-HP	S	A	AM-241	1.000 CI	N
	GAMMA INDUSTRIES	NB-HP	S	A	AM-241	1.000 CI	N
	GAMMA INDUSTRIES	NB-HP	S	A	AM-241	1.000 CI	N
	GAMMA INDUSTRIES	VD-HP	S	A	AM-241	0.840 CI	N
COLO 225-01	TROXLER INC.	3401	D	E	AM-241	40.000 MCI	Y
	TROXLER INC.	3401	D	E	AM-241	40.000 MCI	Y
	TROXLER INC.	3401	D	E	AM-241	40.000 MCI	Y
	TROXLER INC.	3401B	D	E	AM-241	40.000 MCI	Y
	TROXLER INC.	3401B	D	E	AM-241	40.000 MCI	Y
COLO 365-01	TROXLER INC.	3400	D	E	AM-241	44.000 MCI	N
	TROXLER INC.	3400	D	E	AM-241	44.000 MCI	N
96-01	TROXLER INC.	2401	D	E	AM-241	50.000 MCI	N
522-01	TROXLER INC.	3411B	D	E	AM241	44.000 MCI	N
	TROXLER INC.	3411B	D	E	AM241	44.000 MCI	N
	TROXLER INC.	3411B	D	E	AM241	44.000 MCI	N
495-2	TEXAS NUCLEAR	9256	S	O	PU-238	30.000 MCI	N
9-3985	TROXLER INC.	3400	D	E	AM-241	44.000 MCI	N
IL 01005-01	AMERSHAM CORP.	AMC.16	S	F	AM-241	25.000 MCI	N

LICENSEES THAT WANT TO DISPOSE OR TRANSFER THEIR GTCC SOURCES/DEVICES

LICENSE #	MANUFACTURER	MODEL NUMBER	TYPE USE ISOTOPE	ACTIVITY	CMTS
129-2	OHMART CORP.	NMG-1	D F AM-241	500.000 MCI	Y
1447-1	AMERSHAM CORP.	X-102	D F AM-241	47.000 MCI	N
	AMERSHAM CORP.	X-102	D F AM-241	47.000 MCI	N
	AMERSHAM CORP.	X-102	D F AM-241	47.000 MCI	N
	AMERSHAM CORP.	X-102	D F AM-241	47.000 MCI	N
1270-5	CPN	BRC	D E AM-241	50.000 MCI	N
	CPN	BRC	D E AM-241	50.000 MCI	N
	CPN	BRC	D E AM-241	50.000 MCI	N
	CPN	BRC	D E AM-241	50.000 MCI	N
	CPN	BRC	D E AM-241	50.000 MCI	N
IL-01370-01	NEW ENGLAND NUCLEAR	NER-478B	S D AM-241	300.000 MCI	N
284-10612-01	CPN	B(R) MARK II	D E AM-241	50.000 MCI	N
	CPN	B(R) MARK II	D E AM-241	50.000 MCI	N
	CPN	B(R) MARK II	D E AM-241	50.000 MCI	N
L00656	UNKNOWN	UNKNOWN	D B PU-239	1.962 CI	Y
286R	CPN	CPN-131	S E AM241	50.000 MCI	Y
L02870	CPN	CPN-131	S E AM241	50.000 MCI	N
972-6	CPN	MC-1	D E AM241	50.000 MCI	N
120038203	MONSANTO RESEARCH CORP.	UNKNOWN	S G PU-239	1.000 CI	Y
	MONSANTO RESEARCH CORP.	UNKNOWN	S G PU-239	1.000 CI	Y
	MONSANTO RESEARCH CORP.	UNKNOWN	S G PU-239	1.000 CI	Y
	MONSANTO RESEARCH CORP.	UNKNOWN	S G PU-239	1.000 CI	Y
	MONSANTO RESEARCH CORP.	UNKNOWN	S G PU-239	1.000 CI	Y
12-00010-05	KEVEX	3000T	D D AM-241	5.000 CI	Y

Table 2 (continued)

LICENSEES THAT WANT TO DISPOSE OR TRANSFER THEIR GTCC SOURCES/DEVICES

LICENSE #	MANUFACTURER	MODEL NUMBER	TYPE USE	ISOTOPE	ACTIVITY	CMTS
12-12836-01	AMERSHAM CORP.	AMC.19	S	F AM-241	1.000 CI	N
	AMERSHAM CORP.	UNKNOWN	S	F AM-241	100.000 MCI	N
	AMERSHAM CORP.	UNKNOWN	S	F AM-241	100.000 MCI	N
	AMERSHAM CORP.	AMC.16	S	F AM-241	100.000 MCI	N
	AMERSHAM CORP.	AMC.16	S	F AM-241	100.000 MCI	N
	AMERSHAM CORP.	AMC.16	S	F AM-241	100.000 MCI	N
	AMERSHAM CORP.	AMC.17	S	F AM-241	300.000 MCI	N
109-1	TROXLER INC.	2401	D	E AM241	50.000 MCI	Y
	TROXLER INC.	2401	D	E AM241	50.000 MCI	Y
	TROXLER INC.	2401	D	E AM241	50.000 MCI	Y
	TROXLER INC.	2401	D	E AM241	50.000 MCI	Y
	TROXLER INC.	2401	D	E AM241	50.000 MCI	Y
	TROXLER INC.	2401	D	E AM241	50.000 MCI	Y
	TROXLER INC.	2401	D	E AM241	50.000 MCI	Y
	CPN	B-2	D	E AM241	50.000 MCI	Y
1319-2,76-13	NUCLEAR MATERIALS & EQUIP.	4B6-211-50-60	D	B PU-239	5.000 CI	N
SNM-56	MOUND LABS	M39	S	O PU239	1.000 CI	N
	MOUND LABS	M40	S	O PU239	1.000 CI	N
	MOUND LABS	M41	S	O PU239	1.000 CI	N
	MONSANTO RESEARCH CORP.	MRC-N-SS-PU8BE-65	S	O PU238	27.970 CI	N
SNM-918	MONSANTO RESEARCH CORP.	MRC-N-SS-W-PUBE	S	O PU239	4.900 CI	N
AR363BP2	OHMART CORP.	WEIGHTART 36	D	F AM241	500.000 MCI	N
SNM-203	CSI	7002	S	O PU238	30.000 MCI	N
47-14862-01	AMERSHAM CORP.	AM.15	S	F AM241	30.000 MCI	N
	AMERSHAM CORP.	AM.15	S	F AM241	30.000 MCI	N
45-09475-30	E. I. DUPONT	1-XRF	D	D AM241	100.000 MCI	N

Table 2 (continued)

LICENSEES THAT WANT TO DISPOSE OR TRANSFER THEIR GTCC SOURCES/DEVICES

LICENSE #	MANUFACTURER	MODEL NUMBER	TYPE	USE	ISOTOPE	ACTIVITY	CMTS
45-09475-30	MONSANTO RESEARCH CORP.	M181	S	O	PU239	902.000 MCI	N
	MONSANTO RESEARCH CORP.	M-182	S	O	PU239	909.000 MCI	N
	MONSANTO RESEARCH CORP.	M-179	S	O	PU239	896.000 MCI	N
	MONSANTO RESEARCH CORP.	M-180	S	O	PU239	903.000 MCI	N
	MONSANTO RESEARCH CORP.	M-567	S	O	PU239	933.000 MCI	N
	PARKWELL LAB	33PL	S	O	PU239	913.000 MCI	N

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Table 2 (continued)

LICENSEES THAT WANT TO DISPOSE OF THEIR GTCC DAMAGED SOURCES AND/OR DEVICES

LICENSE #	MANUFACTURER	MODEL NUMBER	TYPE	USE	ISOTOPE	ACTIVITY	CMTS
LO7288	AMERSHAM CORP.	0634	S	D	CM244	30.000 MCI	Y
SNM-362	ATOMLAB	MODEL 1026/1036	D	B	CS-137	3212.000 CI	N
L01715	AMERSHAM CORP.	PPCQ357	S	D	PU-238	50.000 MCI	Y
L01715	AMERSHAM CORP.	PPCQ357	S	D	PU-238	50.000 MCI	Y
L01715	AMERSHAM CORP.	PPCQ357	S	D	PU-238	50.000 MCI	Y
L01715	AMERSHAM CORP.	PPCQ357	S	D	PU-238	50.000 MCI	Y
L01715	AMERSHAM CORP.	PPCQ357	S	D	PU-238	50.000 MCI	Y
L01715	AMERSHAM CORP.	PPCQ357	S	D	PU-238	50.000 MCI	Y
L01715	AMERSHAM CORP.	PPCQ357	S	D	PU-238	50.000 MCI	Y
L01715	AMERSHAM CORP.	PPCQ357	S	D	PU-238	50.000 MCI	Y

Although 54.6% of the GTCC sources or devices were portable gauges, mainly moisture density gauges, the americium-241 activity totaled about 272 curies (1.5% of the total americium-241 activity). This signifies that most of the activity is contained in well-logging sources or devices.

Well-logging accounted for 24.2% of the total number of GTCC sources or devices, 80.4% of the total americium-241 activity, and 30.4% of the total plutonium-238 activity.

Of the approximately 41,000 NRC general licensees about 19% or 7,090 general licensees possessed devices containing a total of 2,270 curies of americium-241 and about 3.5% or 1425 general licensees possess 68 curies of curium-244.

Although 22.5% or about 8,500 general licensees are expected to possess GTCC sources the GLDB does identify individual licensees wishing to transfer or dispose of GTCC sources or devices.

6 DATA ANALYSIS

Based on the findings from Tables 1 to 3, and the numbers given to us from the Agreement States, we estimate that 37% of the licensees are expected to have GTCC sources or devices. The survey shows that 56% of the 37% (or 21%) actually have GTCC sources or devices. Since there are approximately 15,783 Agreement State licensees and 8,204 (from the LTS system) NRC specific licensees, we estimate that there are 4,928 licensees having GTCC sources and devices.

From the survey, we approximate that each licensee has 5.51 (7340/1332) sources. Therefore, it is estimated that there are 27,156 GTCC sources or devices. Based on a similar analysis we estimate that 200 licensees will have trouble disposing of their GTCC sources or devices, and a portion of the estimated 2,150 licensees that have not tried to transfer or dispose their sources. We estimated that there will be 1,784 sources or devices that will need to be disposed of or transferred (see Tables 2 and 3 for survey breakdown) and 2,645 sources or devices in storage.

The approximate volume, assuming all the sources to be well-logging sources (typical volume = 0.603 cm^3) the minimum amount of space to hold the GTCC sources that need to be disposed of is $1,076 \text{ cm}^3$. If all the 2,645 stored or surplus sources were to be disposed of, then those sources would account for an approximate volume of $1,595 \text{ cm}^3$.

The 2,645 GTCC sources or devices in storage that will possibly need to be disposed of accounts for the following estimated activities (survey values x 27,156/7340); 19,643 curies of americium-241, 80,184 curies of cesium-137, 1,555 curies of plutonium-238, and 137 curies of plutonium-239.

The 1,784 GTCC sources or devices that the licensees would like to dispose or transfer account for the following estimated activities; 2,975 curies of americium-241, 249,717 curies of cesium-137, 0.1 curie of curium-244, 2,672 curies of plutonium-238, and 313 curies of plutonium-239.

The data analyzed from the GLDB represent the bulk of general licensees NRC believes to be under its jurisdiction. Based on this analysis, 22.5% of the general licensee are expected to have GTCC sources. This is very close to that expected of specific licensees, or 21%. This close agreement occurs because certain types of devices can be used either under a specific license or under a general license. If one assumes that the NRC and Agreement State ratio of general licensees is 1 to 1, then 22.5 percent of the general licensees in the States would also have GTCC sources. The estimated universe could be 18,200 licensees possessing about 67,500 Class C sources. Support for these assumptions comes from past studies and from recognition that some Agreement States do not recognize the general license concept.

If we assume that general licensees' problems with disposal are identical to specific licensees, then we would expect 4% or 730 of the licensees to have difficulty in finding disposal and 9.7% or 1,765 licensees having surplus sources in storage. Based on data from past studies of general licensees, each licensee possesses on the average, 3.7 gauges. This projects to 2,700 devices that could be disposed if a site were available and about 6,530 surplus devices in storage.

The possible errors associated with the above estimates are due to: inaccurate information in the LTS and GLDB systems; inaccurate data submitted on survey form by licensees; duplicate surveys; rounding errors; and inaccurate data input. The errors associated with data entry are expected to be low since the data were edited before they were added to the database. In addition, the program code kept the data consistent where possible, and the computer code was thoroughly tested before data were entered. The errors associated with the LTS, and the GLDB, management systems would be due to inaccurate data in them. The errors in the LTS system tracking systems are expected to be low, since these systems are constantly being updated. The errors in the GLDB systems are expected to be higher than the LTS system due mainly to lack of staff time to perform continued quality control of the data. The data extracted from these systems covers all license categories and regions, and, therefore, any errors would tend to balance themselves out. These databases are constantly changing, but were assumed to remain constant over the survey period. Duplicate surveys were checked for, after all the data were put in the system. A list was printed of all the licensees in the database, and any licensees that had the same license name were checked to see if the information were the same. If it was, then the duplicate licensee and their GTCC sources or devices were permanently removed from the database.

We note that the conclusions from the survey data are within a factor of two of the estimate made by the staff in early 1988 of 17,000 GTCC source possessed by specific licensees used to stimulate the staff to inform the Commission about the GTCC problem. This early estimate was refined by NRC staff to 25,000 to 30,000 and was used by the Office of Technology Assessment (OTA) report on "An Evaluation of Options for Managing Greater-Than-Class-C Low-Level Radioactive Waste" dated October 1988. The refined staff estimates were within 10% of the survey findings, and were within an order of magnitude for those sources that need to be disposed of or transferred.

X-ray fluorescence

Technique

Primary radiation from the radioisotope source excites atoms of the elements present in the sample, removing electrons from the sub-shells round the nucleus. X-rays characteristic of each element are emitted as electrons from the outer shells and move to fill the gaps created in inner shells. The shell from which the electron is removed determines the series of X-rays produced. The intensity of the X-ray is indicative of the concentration of the particular element in the sample. Since radioisotopes emit specific radiations, a limitation results on the range of elements whose characteristic X-rays can be excited. Thus a series of nuclides is employed in order that excitation of all elements from silicon to uranium can be achieved.

Geometry

Applications

- Alloy analysis for checking stock, scrap sorting and checking components.
- In mining, analysis of material excavated from pits, and cores, chippings and slimes from drilling operations.
- Analysis of electroplating solutions.
- General chemical analysis.

Source

Capsule dimensions and Safety performance testing

Capsule	Overall diam. 'A' mm	Active diam. 'B' mm	Window thickness 'C' mm	Safety performance testing ANSI/ISO Classification	IAEA special form	NRC model no.
1102	3	4.2	0.2-0.25	77C64545	GB/3-S	AMC D2
111	10.8	7.2	0.2-0.25	77C64444	GB/4-S	AMC D3
111/1	10.9	8.0	0.2-0.25	77C64444	GB/4-S	AMC D3

Thickness gauging

Transmission thickness technique

The source and the detector are placed on opposite sides of the material to be measured. Gamma or beta radiation transmitted through the sample is then directly related to the sample thickness, provided the density of the material is constant.

Geometry

Applications

Gamma gauging

- Thickness gauging of sheet metal, glass, plastic, and rubber at thickness too large for beta sources, e.g., greater than 500mg/cm²
- Belt weighing, giving mass (kg, mt) flowing on conveyor belt.

Beta Gauging

- Thickness gauging of thinner plastics, thin sheet metal, rubbers, textiles and paper, e.g., 1 – 1000mg/cm²
- The weighing of cigarettes
- Measurement of dust and pollutant levels on filter paper samples e.g., 0.1 – 200 mg/m² dust.

Source

Capsule dimensions and Safety performance testing

Capsule	Overall diam. 'A'	Active diam. 'B'	Window thickness 'C'	Safety performance testing ANSI/ISO classification	IAEA special form	NRC model no.
K 24	76	31	0.05-0.3	TE6444	GB 121 S	AMC 30
K 25	45	40	0.05-0.3	TE6444	GB 121 S	AMC 30

Thickness gauging

Beta backscatter thickness technique

The intensity of beta radiation which is scattered back from thin samples is related to sample thickness and atomic number

Geometry

Beta backscatter thickness gauging

Applications

- The thickness gauging of paper, plastic and rubber on steel rolls.
- The measurement of a coating thickness on a substrate. Virtually any coating/substrate combination providing there is sufficient difference in density or atomic number. Coating range < 1–100μm depending on source and materials.

Source

Capsule dimensions and Safety performance testing

Capsule	Overall diam. "A" mm	Active diam. "B" mm	Window thickness "C" mm	Safety performance testing ANSI/ISO classification	IAEA speed test model no.	HRC model no.
K 91	19	15	0.2-0.25	11C64444	GB-38/S	AMC 15
K 92	15	12	0.2-0.25	11C64444	GB-39/S	AMC 11
K 93	20	16	0.2-0.25	11C64444	GB-40/S	AMC 13
K 97	22	18	0.2-0.25	11C64444	GB-41/S	AMC 18

Gamma backscatter thickness technique

The intensity of back scattered radiation from the sample is measured to give sample thickness or mean atomic number. Used for the measurement of substances of low Z for which transmission measurements are not sufficiently sensitive.

Geometry

Gamma backscatter thickness gauging

Mean atomic number (Z) gauging (ie where thickness is known).

Applications

Thickness gauging

- Measurement of light alloys, glass, plastics, rubbers for which beta sources are not suitable e.g., greater than 500 mg/cm², and access only available from one side. e.g., tube wall thickness gauging

Level gauging

Gamma switching technique

The transmission of gamma radiation through a container is affected by the level of the contents. The intensity of the transmitted radiation is measured and used to activate switches when pre-set intensity levels are reached.

Geometry

Applications

Storage hopper level control

- Switch can be used to operate high level and low level alarms or pump switch control.

oil well logging

Neutron porosity logging technique

Fast neutrons emitted by a neutron source are slowed down by the formation and may undergo three interactions: 1) inelastic scatter, 2) elastic scatter, 3) absorption. Therefore, by collision with hydrogen atoms in the formation the neutron will be moderated to thermal or epithermal energies where it is soon captured by hydrogen nuclei and emits a secondary gamma ray. The detection of these three interactions by using different types of neutron detectors (BF_3 (thermal), ^3He (epithermal)) can be used to determine the hydrogen content of the formation. Since the majority of hydrogen in a formation generally exists within the pore space, the neutron flux will then be related to the porosity.

Geometry

Applications

- Determination of formation hydrogen content
- Formation porosity for oil and mineral logging

Source:

Moisture gauging

Technique

Fast neutrons emitted by the source are moderated by collision with hydrogen atoms in moisture contained in the material. These moderated or thermal neutrons are detected by a neutron detector (usually a boron trifluoride (BF_3) proportional counter) to give a measure of the concentration of hydrogen atoms.

Geometry

Applications

- Soil moisture content for agricultural and construction use.
- Moisture content of materials in silos.
- Continuous moisture content gauging in raw material supplies e.g. gravel, wood chips etc.

Source

Safety performance testing

Capsule	ANSI/ISO classification	IAEA special form	NRC model no.
X.2	11E56544	DB-3-S	AMN-PES
X.21	11E56545	DB-4-S	AMN-PES

SURVEYOR'S INFORMATION SHEET

- License #** - This number must be placed on all survey sheets. The license number will be used as the unique identifier for computer tracking purposes.
- Manufacturer** - This will denote the vendor of the source or device (source housing), i.e. Ohmart, Kay-Ray, Amersham, Gulf Nuclear, DOE, etc.
- Model #** - The vendor's NRC approved model number for the source or source and device combination. Care should be exercised not to include system numbers.
- Type** - Denotes if the model is a source by placing a "S" in the box or a device by placing a "D" in the box.
- Active** - Only check this box if the source or device is used on a routine basis by the licensee and not considered inactive as defined below.
- Inactive** - If the source or device is not used on a routine basis by the licensee, the codes on the survey sheet are to be used to indicate the status of the source.
- The codes are self explanatory. However, for the purposes of this paper the following codes are defined.
- Damaged** - This code denotes that the source or device has been damaged in any way. This also includes any leaking sources. If you use this code then provide a brief explanation of the damage. You may also need to bring this to the attention of your supervisor to determine the need for follow-up action.
- Surplus** - Those sources or devices that are being held as a spare for later use by the licensee.
- Logistics** - Please print legibly.
- Make a copy of the survey sheet, retain a copy for the license file and forward the original to our office for analysis.
 - Thank you for your cooperation.

ABOVE CLASS C SOURCE/DEVICE INVENTORY SURVEY

Licensee's (name and address)

License #: _____

Licensee Name: _____
Contact Name: _____
Title: _____
Department: _____
Street: _____
City: _____ State: _____ Zip Code: _____
Phone Number: () - Ext.: _____

Provide accurate and complete responses to each question below:

1) How many sealed sources and/or devices do you have that are above Class C (i.e. Am-241 > 27 mCi, Pu-238 or -239 > 27 mCi, Cm-244 > 27 mCi, Cs-137 > 910 Ci, or any other transuranic > 27 mCi with a half-life greater than five years)? Identify each source or device on the attached inventory sheet.

2) How do you dispose of your sources and/or devices?
(check appropriate box)

Manufacturer: _____
Transfer to another licensee: _____
Other: _____

If other, please elaborate: _____

3)a. Are you able to find and use an authorized recipient to purchase, dispose, or store any sources and/or devices that you no longer want? (check one) Yes _____ No _____

If no, please elaborate: _____

b. Are there any difficulties in using this authorized recipient? (check one) Yes _____ No _____

If yes, please elaborate: _____

4) Additional comments - check here _____ and use back of this sheet.

Surveyor: _____ Date: _____

Note: Activity levels described in question 1 were derived from limits established in 10 CFR 61 section 61.55. The levels were based on typical size sources.

License #:

ABOVE CLASS C SOURCE/DEVICE INVENTORY SHEET

[illegible]

NOTE: List each source/device separately. If inactive or use is other, please explain.

CODES:

Isotope:	Am-241	Inactive:	D - Damaged	X - Surplus	Use:	A - Well logging	F - Fixed gauges
Cm-244		L - Lost	O - Other		B - Irradiator		G - Broad licenses
Cs-137		T - Wants to dispose or transfer			C - Teletherapy		H - Pacemakers
Pu-238		DT - Damaged and wants to dispose or transfer			D - X-ray fluorescence		I - Waste brokers
Pu-239					E - Portable gauges		O - Other

APPENDIX E
DATA TABLES

Light Water Reactors utility waste generation

--◆-- Process PWR -●- Decommissio PWR -▲- Process BWR -+--- Decommissio BWR

REA P COMPONENT	Greater Than Class C PACKED VOLUME REPORT				
	1985	1990	1995	2000	2005
D COMPACTIBLE TRASH-DECOMMISSION	4.20E+01	4.20E+01	4.20E+01	4.20E+01	4.20E+01
* *****	4.20E+01	4.20E+01	4.20E+01	4.20E+01	4.20E+01
S CARTRIDGE FILTER-OPERATIONAL	2.60E+01	2.60E+01	3.60E+01	4.60E+01	4.60E+01
O COMPACTIBLE TRASH-OPERATIONS	2.58E+01	4.94E+01	7.09E+01	9.24E+01	9.39E+01
CONTAMINATED EQUIPMENT-OPERATIONS	1.50E+00	1.65E+01	1.75E+01	1.85E+01	1.95E+01
CONTROL ROD BLADES - NOT CONTROL CELL	8.50E-02	1.10E-01	1.30E-01	1.30E-01	1.30E-01
DEWATERED ION EXCHANGE RESIN	5.00E+00	5.00E+00	5.00E+00	5.00E+00	5.00E+00
FILTER MEDIA	1.50E+00	2.50E+00	3.50E+00	4.50E+00	5.50E+00
METAL SHAVINGS	5.00E+00	5.00E+00	5.00E+00	5.00E+00	5.00E+00
METALS-OPERATIONS	3.40E-01	6.75E-01	1.01E+00	1.35E+00	1.68E+00
MIXED-OXIDE FUEL PELLETS/RODS	4.15E+00	4.15E+00	4.15E+00	4.15E+00	4.15E+00
NONCOMPACTIBLE TRASH-OPERATIONS	3.75E+00	9.82E+00	9.88E+00	9.95E+00	1.00E+01
ORGANIC LIQUID	1.38E+00	2.00E+00	2.62E+00	3.24E+00	3.86E+00
SOLIDIFIED RESIN	6.00E+00	6.00E+00	1.10E+01	1.60E+01	1.60E+01
* *****	4.36E+01	1.27E+02	1.67E+02	2.06E+02	2.11E+02
S					

REA P COMPONENT		Greater Than Class C PACKED VOLUME REPORT			
		2015	2020	2025	2030
D COMPACTIBLE TRASH-DECOMMISSION		4.20E+01	4.20E+01	4.20E+01	4.20E+01

A *****					
S *****					
O CARTRIDGE FILTER-OPERATIONAL					
		4.20E+01	4.20E+01	4.20E+01	4.20E+01
		4.60E+01	4.60E+01	4.60E+01	4.60E+01
COMPACTIBLE TRASH-OPERATIONS					
		9.69E+01	9.84E+01	9.96E+01	9.96E+01
CONTAMINATED EQUIPMENT-OPERATIONS					
		2.15E+01	2.25E+01	2.33E+01	2.33E+01
CONTROL ROD BLADES - NOT CONTROL CELL					
		1.30E-01	1.30E-01	1.30E-01	1.30E-01
DEWATERED ION EXCHANGE RESIN					
		5.00E+00	5.00E+00	5.00E+00	5.00E+00
FILTER MEDIA					
		7.50E+00	8.50E+00	9.30E+00	9.30E+00
METAL SHAVINGS					
		5.00E+00	5.00E+00	5.00E+00	5.00E+00
METALS-OPERATIONS					
		2.35E+00	2.68E+00	2.95E+00	2.95E+00
MIXED-OXIDE FUEL PELLETS/RODS					
		4.15E+00	4.15E+00	4.15E+00	4.15E+00
NONCOMPACTIBLE TRASH-OPERATIONS					
		1.02E+01	1.02E+01	1.03E+01	1.03E+01
ORGANIC LIQUID					
		5.10E+00	5.72E+00	6.22E+00	6.22E+00
SOLIDIFIED RESIN					
		1.60E+01	1.60E+01	1.60E+01	1.60E+01

A *****					
S *****					
O *****					
		2.20E+02	2.24E+02	2.28E+02	2.28E+02

AREA P COMPONENT		Greater Than Class C PACKED VOLUME REPORT				
		1985	1990	1995	2000	2005
PWR L PWR CORE SHROUD AND FORMER PLATES		2.54E+00	2.54E+00	2.54E+00	2.54E+00	5.08E+00
PWR FULL SYSTEM FUEL-IN RESINS		4.24E+00	4.24E+00	4.24E+00	4.24E+00	8.48E+00
*****		*****	*****	*****	*****	*****
S CARTRIDGE FILTER-OPERATIONAL		6.78E+00	6.78E+00	6.78E+00	6.78E+00	1.36E+01
		4.44E-01	4.44E-01	4.44E-01	4.44E-01	4.44E-01
CONTROL ROD DRIVE		1.80E+01	1.80E+01	1.80E+01	1.80E+01	1.80E+01
FLUX WIRE		4.60E-01	4.60E-01	4.60E-01	4.60E-01	4.60E-01
METALS-OPERATIONS		1.29E+00	1.59E+00	1.59E+00	1.59E+00	1.59E+00
PRIMARY/SECONDARY SOURCE		1.05E+00	1.08E+00	1.10E+00	1.13E+00	1.16E+00
PWR CRUD TANK FILTERS		1.21E+00	1.80E+00	2.40E+00	2.99E+00	3.59E+00
PWR IN-CORE INSTRUMENTS		2.01E+01	2.95E+01	3.98E+01	5.01E+01	6.04E+01
PWR MISC. CARTRIDGE FILTERS		4.09E+01	6.77E+01	9.81E+01	1.29E+02	1.60E+02
THIMBLE PLUGS OR ASSEMBLY		5.34E+01	5.66E+01	6.03E+01	6.41E+01	6.79E+01
*****		*****	*****	*****	*****	*****
S		1.37E+02	1.77E+02	2.22E+02	2.68E+02	3.13E+02

Greater Than Class C PACKED VOLUME REPORT		2015		2020		2025		2030		2035	
REA P COMPONENT											
PWR D PWR CORE SHROUD AND FORMER PLATES		3.17E+01	1.24E+02	1.55E+02	2.33E+02	2.65E+02					
PWR FULL SYSTEM FUEL-IN RESINS		4.66E+01	1.61E+02	2.04E+02	2.80E+02	3.22E+02					

CARTRIDGE FILTER-OPERATIONAL		7.83E+01	2.85E+02	3.58E+02	5.21E+02	5.80E+02					
CONTROL ROD DRIVE		4.44E-01	4.44E-01	4.44E-01	4.44E-01	4.44E-01					
FIUX WIRE		1.80E+01	1.80E+01	1.80E+01	1.80E+01	1.80E+01					
METALS-OPERATIONS		4.60E-01	4.60E-01	4.60E-01	4.60E-01	4.60E-01					
PRIMARY/SECONDARY SOURCE		1.59E+00	1.59E+00	1.59E+00	1.59E+00	1.59E+00					
PWR CRUD TANK FILTERS		1.21E+00	1.23E+00	1.24E+00	1.24E+00	1.24E+00					
PWR IN-CORE INSTRUMENTS		4.59E+00	4.64E+00	4.64E+00	4.64E+00	4.64E+00					
PWR MISC. CARTRIDGE FILTERS		7.89E+01	8.35E+01	8.64E+01	8.69E+01	8.69E+01					
THIMBLE PLUGS OR ASSEMBLY		2.13E+02	2.29E+02	2.39E+02	2.42E+02	2.42E+02					

		7.45E+01	7.66E+01	7.81E+01	7.86E+01	7.86E+01					

		3.93E+02	4.15E+02	4.30E+02	4.34E+02	4.34E+02					

S

REA F COMPONENT	Greater Than Class C PACKED VOLUME REPORT				
	1985	1990	1995	2000	2005
BWR D BWR CORE SHROUD	5.78E+00	5.78E+00	5.78E+00	1.16E+01	1.16E+01
BWR FULL SYSTEM FUEL-IN RESINS	2.98E+00	2.98E+00	2.98E+00	2.98E+00	5.96E+00

BWR INNER CONTROL ROD DRIVE FILTERS	5.78E+00	8.76E+00	8.76E+00	1.45E+01	1.75E+01
BWR OUTER CONTROL ROD DRIVE FILTERS	1.07E-01	1.66E-01	2.31E-01	2.95E-01	3.59E-01
BWR UNDERWATER VACUUM FILTERS	4.76E+00	7.33E+00	1.01E+01	1.30E+01	1.57E+01
CARTRIDGE FILTER-OPERATIONAL	3.53E+00	5.47E+00	7.59E+00	9.72E+00	1.18E+01
CONTROL ROD BLADES - NOT CONTROL CELL	5.80E-01	1.16E+00	1.16E+00	1.16E+00	1.16E+00
CONTROL ROD ROLLER BEARINGS	8.58E+01	1.36E+02	1.92E+02	2.48E+02	3.03E+02
LPRM - LOCAL POWER RANGE MONITORS	8.49E-05	1.42E-04	1.42E-04	1.42E-04	1.42E-04
POISON CURTAIN	2.13E+01	3.23E+01	4.46E+01	5.66E+01	6.85E+01
SRM DRY TUBES - SOURCE RANGE MONITORS	6.78E-03	6.78E-03	6.78E-03	6.78E-03	6.78E-03
	5.40E+00	7.67E+00	1.02E+01	1.27E+01	1.52E+01

	1.21E+02	1.90E+02	2.66E+02	3.42E+02	4.16E+02

					4.87E+02

Greater Than Class C PACKED		VOLUME REPORT			
		2015	2020	2025	2030
REA F COMPONENT		2035			
BWR D BWR COKE SHROUD		7.51E+01	1.47E+02	1.53E+02	2.42E+02
BWR FULL SYSTEM FUEL-IN RESINS		3.28E+01	6.85E+01	7.15E+01	1.04E+02
* *****		-----			
O BWR INNER CONTROL ROD DRIVE FILTERS		1.08E+02	2.16E+02	2.24E+02	3.46E+02
BWR OUTER CONTROL ROD DRIVE FILTERS		4.61E-01	4.86E-01	5.06E-01	5.09E-01
BWR UNDERWATER VACUUM FILTERS		2.01E+01	2.12E+01	2.20E+01	2.22E+01
CARTRIDGE FILTER-OPERATIONAL		1.52E+01	1.60E+01	1.67E+01	1.68E+01
CONTROL ROD BLADES - NOT CONTROL CELL		1.16E+00	1.16E+00	1.16E+00	1.16E+00
CONTROL ROD ROLLER BEARINGS		3.94E+02	4.19E+02	4.39E+02	4.41E+02
LPRM - LOCAL POWER RANGE MONITORS		1.42E-04	1.42E-04	1.42E-04	1.42E-04
POISON CURTAIN		8.77E+01	9.24E+01	9.60E+01	9.67E+01
SKM DRY TUBES - SOURCE RANGE MONITORS		6.78E-03	6.78E-03	6.78E-03	6.78E-03
*****		1.93E+01	2.04E+01	2.12E+01	2.13E+01
*****		5.38E+02	5.70E+02	5.96E+02	6.00E+02
*****		-----			

Greater Than Class		C 5 YEAR ACTIVITY REPORT						
REA P COMPONENT	NUCLID	1985	1990	1995	2000	2005	2010	

BWR D BWR CORE SHROUD	C-14a	9.22E+01	9.21E+01	9.21E+01	1.84E+02	1.84E+02	2.76E+02	
	Co-60	1.10E+05	5.71E+04	2.96E+04	1.59E+05	8.22E+04	1.86E+05	
	Fe-55	5.41E+04	1.50E+04	4.15E+03	9.15E+04	2.53E+04	9.74E+04	
	Mn-54	8.31E+00	1.41E-01	2.39E-03	4.25E+01	7.20E-01	4.25E+01	
	Nb-94a	9.22E-01	9.22E-01	9.21E-01	1.84E+00	1.84E+00	2.76E+00	
	Ni-59a	4.81E+02	4.81E+02	4.81E+02	9.62E+02	9.62E+02	1.44E+03	
	Ni-63a	6.40E+04	6.18E+04	5.97E+04	1.23E+05	1.18E+05	1.79E+05	

BWR FULL SYSTEM FUEL-IN RESINS	Am-241	4.57E-02	3.26E-02	3.26E-02	2.56E-02	6.58E-02	4.84E-02	
	C-14a	1.00E-03	9.99E-04	9.99E-04	9.98E-04	2.00E-03	2.00E-03	
	Ce-141	1.48E-02	2.84E-03	2.84E-03	5.43E-04	1.49E-02	2.86E-03	
	Ce-144	4.16E-02	4.86E-04	4.86E-04	5.68E-06	4.16E-02	4.86E-04	
	Cm-242	1.77E-04	6.73E-08	6.73E-08	2.55E-11	1.77E-04	6.73E-08	
	Co-58	3.34E-05	5.83E-13	5.83E-13	1.02E-20	3.34E-05	5.83E-13	
	Co-60	1.42E+02	7.33E+01	7.33E+01	3.80E+01	1.61E+02	8.35E+01	
	Cs-134	2.19E-02	4.07E-03	4.07E-03	7.56E-04	2.20E-02	4.09E-03	
	Cs-137	6.53E-02	5.82E-02	5.82E-02	5.19E-02	1.12E-01	9.95E-02	
	Fe-55	2.09E+01	5.78E+00	5.78E+00	1.60E+00	2.13E+01	5.91E+00	
	Fe-59	4.75E-09	2.19E-21	2.19E-21	1.01E-33	4.75E-09	2.19E-21	
	Mn-54	3.70E-01	6.27E-03	6.27E-03	1.06E-04	3.70E-01	6.27E-03	
	Nb-95	9.40E-11	1.97E-26	1.97E-26	.00E+00	9.40E-11	1.97E-26	
	Ni-63a	5.42E+00	5.23E+00	5.23E+00	5.05E+00	1.03E+01	9.95E+00	
	Pu-238	7.91E-05	1.29E-09	1.29E-09	4.88E-13	7.91E-05	1.29E-09	
	Pu-241	2.53E+01	1.98E+01	1.98E+01	1.56E+01	3.75E+01	2.95E+01	
	Sr-90	1.86E-02	1.65E-02	1.65E-02	1.46E-02	3.15E-02	2.79E-02	
	TRU5	4.90E-01	4.90E-01	4.90E-01	4.90E-01	9.80E-01	9.80E-01	
	Zn-65	9.96E-02	5.60E-04	5.60E-04	3.15E-06	9.96E-02	5.60E-04	

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S		2.29E+05	1.35E+05	9.41E+04	3.74E+05	2.27E+05	4.64E+05	

O BWR INNER CONTROL ROD DRIVE FILTERS	Am-241	6.39E-04	9.22E-04	1.18E-03	1.39E-03	1.54E-03	1.64E-03	
	C-14a	1.08E-04	1.67E-04	2.32E-04	2.97E-04	3.61E-04	4.22E-04	
	Co-60	7.32E+01	9.89E+01	1.17E+02	1.27E+02	1.29E+02	1.29E+02	
	Cs-137	1.42E-01	2.11E-01	2.82E-01	3.45E-01	3.99E-01	4.43E-01	
	Fe-55	2.55E+02	3.35E+02	3.77E+02	3.89E+02	3.87E+02	3.69E+02	
	H-3	2.92E-02	4.18E-02	5.31E-02	6.17E-02	6.78E-02	7.13E-02	
	Mn-54	6.76E-01	8.93E-01	9.34E-01	9.35E-01	9.12E-01	8.37E-01	
	Pu-241	3.89E-01	5.62E-01	7.21E-01	8.47E-01	9.40E-01	1.00E+00	
	Sr-90	3.47E-03	5.17E-03	6.88E-03	8.39E-03	9.70E-03	1.07E-02	
	TRU5	4.25E-02	6.59E-02	9.15E-02	1.17E-01	1.42E-01	1.66E-01	

BWR OUTER CONTROL ROD DRIVE FILTERS	Am-241	6.12E-04	9.01E-04	1.17E-03	1.38E-03	1.53E-03	1.63E-03	
	C-14a	1.03E-04	1.62E-04	2.27E-04	2.92E-04	3.56E-04	4.17E-04	
	Co-60	7.06E+01	9.76E+01	1.17E+02	1.27E+02	1.31E+02	1.29E+02	
	Cs-137	1.35E-01	2.06E-01	2.77E-01	3.40E-01	3.95E-01	4.40E-01	
	Fe-55	2.49E+02	3.34E+02	3.77E+02	3.89E+02	3.87E+02	3.69E+02	
	H-3	2.80E-02	4.09E-02	5.25E-02	6.12E-02	6.74E-02	7.10E-02	
	Mn-54	6.71E-01	8.93E-01	9.34E-01	9.35E-01	9.12E-01	8.37E-01	
	Pu-241	3.73E-01	5.49E-01	7.11E-01	8.39E-01	9.34E-01	9.95E-01	
	Sr-90	3.32E-03	5.04E-03	6.76E-03	8.29E-03	9.60E-03	1.07E-02	
	TRU5	4.06E-02	6.40E-02	8.96E-02	1.15E-01	1.41E-01	1.65E-01	

BWR UNDERWATER VACUUM FILTERS	Am-241	7.71E-04	1.11E-03	1.43E-03	1.68E-03	1.86E-03	1.98E-03	
	C-14a	3.46E-01	5.36E-01	7.45E-01	9.53E-01	1.16E+00	1.35E+00	
	Co-60	1.64E+02	2.22E+02	2.64E+02	2.85E+02	2.94E+02	2.90E+02	
	Cs-137	4.85E+00	7.23E+00	9.64E+00	1.18E+01	1.37E+01	1.52E+01	
	Fe-55	1.93E+02	2.54E+02	2.85E+02	2.94E+02	2.92E+02	2.79E+02	
	H-3	5.56E-01	7.96E-01	1.01E+00	1.17E+00	1.29E+00	1.36E+00	
	Mn-54	2.94E+00	3.89E+00	4.07E+00	4.07E+00	3.97E+00	3.65E+00	
	Ni-63a	3.48E+01	5.33E+01	7.31E+01	9.22E+01	1.10E+02	1.27E+02	
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Greater Than Class C 5 YEAR ACTIVITY REPORT

REA P COMPONENT	NUCLID	2015	2020	2025	2030	2035
3WR D BWR CORE SHROUD	C-14a	1.29E+03	2.29E+03	2.38E+03	3.34E+03	3.52E+03
	Co-60	2.39E+06	3.26E+06	1.85E+06	3.06E+06	1.89E+06
	Fe-55	2.14E+06	2.21E+06	7.29E+05	1.91E+06	7.26E+05
	Mn-54	7.19E+03	3.16E+03	1.50E+02	4.46E+03	2.17E+02
	Nb-94a	1.29E+01	2.31E+01	2.40E+01	3.42E+01	3.60E+01
	Ni-59a	6.73E+03	1.20E+04	1.25E+04	1.77E+04	1.86E+04
	Ni-63a	9.00E+05	1.58E+06	1.59E+06	2.24E+06	2.29E+06
BWR FULL SYSTEM FUEL-IN RESINS	Am-241	4.37E-01	7.82E-01	6.47E-01	9.41E-01	8.36E-01
	C-14a	1.20E-02	2.40E-02	2.50E-02	3.59E-02	3.89E-02
	Ce-141	1.07E-01	1.03E-01	2.53E-02	1.09E-01	3.43E-02
	Ce-144	2.51E-01	9.35E-02	3.97E-03	1.74E-01	7.28E-03
	Cm-242	1.33E-03	2.23E-04	1.66E-06	5.21E-04	2.42E-06
	Co-58	1.26E-03	3.35E-05	7.40E-10	7.06E-05	7.82E-10
	Co-60	1.25E+03	1.88E+03	1.07E+03	1.84E+03	1.22E+03
	Cs-134	1.57E-01	1.50E-01	3.59E-02	1.59E-01	4.89E-02
	Cs-137	7.23E-01	1.39E+00	1.30E+00	1.85E+00	1.83E+00
	Fe-55	1.60E+02	1.81E+02	5.99E+01	1.77E+02	7.36E+01
	Fe-59	1.40E-06	4.75E-09	1.88E-16	9.57E-09	1.89E-16
	Mn-54	2.23E+00	9.26E-01	4.77E-02	1.64E+00	8.81E-02
	Nb-95	1.29E-07	9.40E-11	3.68E-11	1.88E-10	3.69E-20
	Ni-63a	6.33E+01	1.25E+02	1.26E+02	1.81E+02	1.90E+02
	Pu-238	1.77E-04	8.00E-05	3.16E-08	1.61E-04	4.62E-08
	Pu-241	2.61E+02	4.74E+02	3.94E+02	5.68E+02	5.09E+02
	Si-90	2.05E-01	3.92E-01	3.64E-01	5.20E-01	5.12E-01
	TRU5	5.88E+00	1.18E+01	1.22E+01	1.76E+01	1.91E+01
	Zn-65	6.12E-01	1.87E-01	5.50E-03	3.79E-01	9.74E-03

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O BWR INNER CONTROL ROD DRIVE FILTERS

Am-241	5.44E+06	7.07E+06	4.19E+06	7.23E+06	4.92E+06
C-14a	1.58E-03	1.42E-03	1.26E-03	1.01E-03	7.92E-04
Co-60	4.64E-04	4.89E-04	5.09E-04	5.12E-04	5.11E-04
Cs-137	1.08E+02	8.17E+01	6.22E+01	3.48E+01	1.80E+01
Fe-55	4.54E-01	4.42E-01	4.23E-01	3.81E-01	3.40E-01
H-3	2.72E+02	1.85E+02	1.33E+02	4.58E+01	1.27E+01
Mn-54	6.75E-02	5.94E-02	5.15E-02	3.98E-02	3.00E-02
Pu-241	4.94E-01	3.52E-01	2.35E-01	1.29E-02	2.18E-04
Si-90	9.63E-01	8.67E-01	7.66E-01	6.14E-01	4.83E-01
TRU5	1.10E-02	1.06E-02	1.01E-02	9.08E-03	8.04E-03
	1.83E-01	1.93E-01	2.01E-01	2.02E-01	2.02E-01

BWR OUTER CONTROL ROD DRIVE FILTERS

Am-241	1.59E-03	1.44E-03	1.28E-03	1.03E-03	8.06E-04
C-14a	4.61E-04	4.88E-04	5.09E-04	5.12E-04	5.11E-04
Co-60	1.10E+02	8.51E+01	6.44E+01	3.59E+01	1.86E+01
Cs-137	4.54E-01	4.45E-01	4.26E-01	3.84E-01	3.43E-01
Fe-55	2.82E+02	1.97E+02	1.39E+02	4.73E+01	1.31E+01
H-3	6.79E-02	6.05E-02	5.25E-02	4.06E-02	3.06E-02
Mn-54	5.36E-01	3.79E-01	2.37E-01	1.29E-02	2.19E-04
Pu-241	9.67E-01	8.80E-01	7.80E-01	6.25E-01	4.91E-01
Si-90	1.10E-02	1.07E-02	1.02E-02	9.16E-03	8.11E-03
TRU5	1.82E-01	1.93E-01	2.01E-01	2.02E-01	2.02E-01

BWR UNDERWATER VACUUM FILTERS

Am-241	1.91E-03	1.72E-03	1.52E-03	1.22E-03	9.55E-04
C-14a	1.49E+00	1.57E+00	1.63E+00	1.64E+00	1.64E+00
Co-60	2.42E+02	1.83E+02	1.39E+02	7.80E+01	4.04E+01
Cs-137	1.56E+01	1.51E+01	1.45E+01	1.30E+01	1.16E+01
Fe-55	2.06E+02	1.40E+02	1.01E+02	3.46E+01	9.59E+00
H-3	1.28E+00	1.13E+00	9.79E-01	7.57E-01	5.72E-01
Mn-54	2.15E+00	1.53E+00	1.02E+00	5.60E-02	9.50E-04
Ni-63a	1.36E+02	1.40E+02	1.42E+02	1.38E+02	1.34E+02

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Greater Than Class C 5 YEAR ACTIVITY REPORT

REA P COMPONENT	NUCLID	1985	1990	1995	2000	2005	2010	
BWR O BWR UNDERWATER VACUUM FILTERS	Pu-241	4.69E-01	6.77E-01	8.70E-01	1.02E+00	1.13E+00	1.21E+00	
	Sr-90	3.26E-01	4.85E-01	6.46E-01	7.88E-01	9.10E-01	1.01E+00	
	TRU5	1.10E-01	1.70E-01	2.36E-01	3.02E-01	3.67E-01	4.29E-01	
	Zn-65	4.36E+00	5.74E+00	5.96E+00	5.96E+00	5.61E+00	5.29E+00	
	Am-241	1.07E-03	2.94E-03	2.31E-03	1.81E-03	1.43E-03	1.12E-03	
	C-14a	9.48E-04	1.90E-03	1.90E-03	1.89E-03	1.89E-03	1.89E-03	
CARTRIDGE FILTER-OPERATIONAL	Cm-242	1.08E-15	4.08E-06	1.55E-09	5.87E-13	2.23E-16	8.45E-20	
	Co-60	3.29E+01	2.25E+02	1.16E+02	6.03E+01	3.12E+01	1.62E+01	
	Cs-137	8.59E-01	1.95E+00	1.74E+00	1.55E+00	1.38E+00	1.23E+00	
	LT5	2.15E-01	1.05E+01	2.63E+00	6.56E-01	1.64E-01	4.10E-02	
	Nb-94a	3.80E-04	7.60E-04	7.60E-04	7.59E-04	7.59E-04	7.59E-04	
	Ni-59a	6.60E-02	1.32E-01	1.32E-01	1.32E-01	1.32E-01	1.32E-01	
	Ni-63a	3.09E+00	6.40E+00	6.18E+00	5.97E+00	5.77E+00	5.57E+00	
	Pu-238	2.06E-17	7.79E-08	2.96E-11	1.12E-14	4.26E-18	1.62E-21	
	Pu-241	6.51E-01	1.79E+00	1.41E+00	1.11E+00	8.68E-01	6.82E-01	
	Sr-90	2.12E-01	4.85E-01	4.30E-01	3.81E-01	3.37E-01	2.99E-01	
	TRU5	6.70E-02	1.34E-01	1.34E-01	1.34E-01	1.34E-01	1.34E-01	
	Tc-99	1.90E-03	3.80E-03	3.80E-03	3.80E-03	3.80E-03	3.80E-03	
	CONTROL ROD BLADES - NOT CONTROL CELL	C-14a	3.45E+01	5.35E+01	7.44E+01	9.52E+01	1.16E+02	1.35E+02
		Co-60	2.14E+05	2.90E+05	3.44E+05	3.73E+05	3.84E+05	3.79E+05
Fe-55		1.50E+05	1.97E+05	2.22E+05	2.28E+05	2.27E+05	2.17E+05	
H-3		5.00E+03	7.17E+03	9.11E+03	1.06E+04	1.16E+04	1.22E+04	
Mn-54		9.81E+02	1.30E+03	1.36E+03	1.36E+03	1.32E+03	1.22E+03	
Nb-94a		1.10E+00	1.70E+00	2.37E+00	3.03E+00	3.69E+00	4.31E+00	
Ni-59a		1.83E+02	2.78E+02	3.83E+02	4.87E+02	5.90E+02	6.88E+02	
Ni-63a		2.60E+04	3.90E+04	5.30E+04	6.64E+04	7.92E+04	9.08E+04	
C-14a		3.27E-03	5.44E-03	5.44E-03	5.44E-03	5.43E-03	5.43E-03	
Co-60		3.49E+02	7.11E+02	3.68E+02	1.91E+02	9.89E+01	5.12E+01	
CONTROL ROD ROLLER BEARINGS	Fe-55	9.78E-01	3.48E+00	9.65E-01	2.67E-01	7.41E-02	2.05E-02	
	Mn-54	1.93E-05	1.63E-03	2.77E-05	4.70E-07	7.96E-09	1.35E-10	
	Nb-94a	9.99E-05	1.66E-04	1.66E-04	1.66E-04	1.66E-04	1.66E-04	
	Ni-59a	1.44E-02	2.40E-02	2.40E-02	2.40E-02	2.40E-02	2.40E-02	
	Ni-63a	2.07E+00	3.45E+00	3.33E+00	3.22E+00	3.11E+00	3.00E+00	
	Am-241	3.06E-01	2.40E-01	1.89E-01	1.48E-01	1.17E-01	9.15E-02	
	C-14a	1.43E+01	2.02E+01	2.66E+01	3.29E+01	3.91E+01	4.50E+01	
	Cm-242	4.38E-08	5.58E-05	2.12E-08	8.04E-12	3.05E-15	1.16E-18	
	Co-60	5.25E+04	7.29E+04	8.45E+04	8.94E+04	9.12E+04	8.96E+04	
	Cs-137	2.62E+04	2.34E+04	2.08E+04	1.86E+04	1.65E+04	1.48E+04	
	Fe-55	4.47E+04	5.75E+04	6.55E+04	6.66E+04	6.59E+04	6.28E+04	
	H-3	6.39E+01	4.82E+01	3.64E+01	2.75E+01	2.08E+01	1.57E+01	
	LT5	8.04E+01	3.92E+03	9.80E+02	2.45E+02	6.13E+01	1.53E+01	
	Mn-54	1.92E+03	2.47E+03	2.65E+03	2.58E+03	2.52E+03	2.31E+03	
Nb-94a	2.55E-01	3.79E-01	4.97E-01	6.12E-01	7.26E-01	8.34E-01		
LPRM - LOCAL POWER RANGE MONITORS	Ni-59a	8.72E+01	1.27E+02	1.61E+02	1.94E+02	2.27E+02	2.58E+02	
	Ni-63a	1.17E+04	1.62E+04	2.06E+04	2.48E+04	2.88E+04	3.24E+04	
	Pu-238	8.37E-10	1.07E-06	4.05E-10	1.54E-13	5.83E-17	2.21E-20	
	Pu-241	1.86E+02	1.46E+02	1.15E+02	9.03E+01	7.10E+01	5.57E+01	
	Sr-90	2.47E+04	2.19E+04	1.94E+04	1.72E+04	1.52E+04	1.35E+04	
	TRU5	2.37E+03	2.37E+03	2.37E+03	2.37E+03	2.37E+03	2.37E+03	
	Tc-99	4.38E+00	4.38E+00	4.38E+00	4.38E+00	4.38E+00	4.38E+00	
	Am-241	1.99E-01	1.99E-01	1.99E-01	1.99E-01	1.98E-01	1.98E-01	
	C-14a	5.13E+01	2.66E+01	1.38E+01	7.13E+00	3.69E+00	1.91E+00	
	Co-60	2.76E+00	7.65E-01	2.12E-01	5.87E-02	1.63E-02	4.50E-03	
	H-3	3.88E+00	2.93E+00	2.21E+00	1.67E+00	1.26E+00	9.51E-01	
	Mn-54	3.53E-08	5.99E-10	1.02E-11	1.72E-13	2.92E-15	4.95E-17	
	Nb-94a	1.89E-03	1.89E-03	1.89E-03	1.89E-03	1.88E-03	1.88E-03	
	POISON CURTAIN	C-14a	1.99E-01	1.99E-01	1.99E-01	1.99E-01	1.98E-01	1.98E-01
Co-60		5.13E+01	2.66E+01	1.38E+01	7.13E+00	3.69E+00	1.91E+00	
Fe-55		2.76E+00	7.65E-01	2.12E-01	5.87E-02	1.63E-02	4.50E-03	
H-3		3.88E+00	2.93E+00	2.21E+00	1.67E+00	1.26E+00	9.51E-01	
Nb-94a		3.53E-08	5.99E-10	1.02E-11	1.72E-13	2.92E-15	4.95E-17	

REA P COMPONENT	Greater Than Class NUCLID	C 5 YEAR ACTIVITY REPORT			
		2015	2020	2025	2030
BWR O BWR UNDERWATER VACUUM FILTERS	Pu-241	1.16E+00	1.04E+00	9.24E-01	7.41E-01
	Sr-90	1.03E+00	9.98E-01	9.50E-01	8.52E-01
	TRU5	4.71E-01	4.98E-01	5.18E-01	5.21E-01
	Zn-65	3.01E+00	2.22E+00	1.43E+00	4.24E-02
CARTRIDGE FILTER-OPERATIONAL	Am-241	8.79E-04	6.91E-04	5.42E-04	4.26E-04
	C-14a	1.89E-03	1.89E-03	1.89E-03	1.89E-03
	Cm-242	3.21E-23	1.22E-26	4.62E-30	1.75E-33
	Co-60	8.38E+00	4.34E+00	2.25E+00	1.17E+00
	Cs-137	1.10E+00	9.80E-01	8.73E-01	7.79E-01
	LT5	1.03E-02	2.56E-03	6.41E-04	1.60E-04
	Nb-94a	7.59E-04	7.59E-04	7.59E-04	7.59E-04
	Ni-59a	1.32E-01	1.32E-01	1.32E-01	1.32E-01
	Ni-63a	5.38E+00	5.20E+00	5.02E+00	4.85E+00
	Pu-238	6.14E-25	2.33E-28	8.84E-32	3.35E-35
	Pu-241	5.36E-01	4.21E-01	3.30E-01	2.60E-01
	Sr-90	2.65E-01	2.35E-01	2.08E-01	1.84E-01
	TRU5	1.34E-01	1.34E-01	1.34E-01	1.34E-01
	Tc-99	3.80E-03	3.80E-03	3.80E-03	3.80E-03
CONTROL ROD BLADES - NOT CONTROL CELL	C-14a	1.49E+02	1.57E+02	1.63E+02	1.64E+02
	Co-60	3.18E+05	2.40E+05	1.83E+05	1.02E+05
	Fe-55	1.61E+05	1.09E+05	7.84E+04	2.69E+04
	H-3	1.16E+04	1.02E+04	8.84E+03	6.84E+03
	Mn-54	7.25E+02	5.12E+02	3.42E+02	1.87E+01
	Nb-94a	4.74E+00	5.00E+00	5.21E+00	5.24E+00
	Ni-59a	7.56E+02	7.97E+02	8.29E+02	8.34E+02
	Ni-63a	9.75E+04	1.00E+05	1.02E+05	9.88E+04
	C-14a	5.43E-03	5.42E-03	5.42E-03	5.42E-03
	Co-60	2.65E+01	1.38E+01	7.12E+00	3.69E+00
	Fe-55	5.69E-03	1.58E-03	4.36E-04	1.21E-04
	Mn-54	2.29E-12	3.88E-14	6.58E-16	1.12E-17
	Nb-94a	1.66E-04	1.66E-04	1.66E-04	1.66E-04
	Ni-59a	2.40E-02	2.40E-02	2.40E-02	2.40E-02
Ni-63a	2.90E+00	2.80E+00	2.71E+00	2.61E+00	
LPRM - LOCAL POWER RANGE MONITORS	Am-241	7.19E-02	5.65E-02	4.43E-02	3.48E-02
	C-14a	4.90E+01	5.14E+01	5.32E+01	5.34E+01
	Cm-242	4.39E-22	1.67E-25	6.33E-29	2.40E-32
	Co-60	7.42E+04	5.55E+04	4.16E+04	2.36E+04
	Cs-137	1.31E+04	1.17E+04	1.05E+04	9.32E+03
	Fe-55	4.58E+04	3.06E+04	2.15E+04	7.83E+03
	H-3	1.18E+01	8.93E+00	6.74E+00	5.09E+00
	LT5	3.83E+00	9.57E-01	2.39E-01	5.98E-02
	Mn-54	1.33E+03	9.30E+02	5.98E+02	5.26E+01
	Nb-94a	9.07E-01	9.50E-01	9.84E-01	9.90E-01
	Ni-59a	2.79E+02	2.92E+02	3.02E+02	3.03E+02
	Ni-63a	3.44E+04	3.51E+04	3.53E+04	3.44E+04
	Pu-238	8.40E-24	3.19E-27	1.21E-30	4.59E-34
	Pu-241	4.38E+01	3.44E+01	2.70E+01	2.12E+01
	Sr-90	1.19E+04	1.06E+04	9.36E+03	8.29E+03
TRU5	2.37E+03	2.37E+03	2.37E+03	2.37E+03	
Tc-99	4.38E+00	4.38E+00	4.38E+00	4.38E+00	
POISON CURTAIN	C-14a	1.98E-01	1.98E-01	1.98E-01	1.98E-01
	Co-60	9.91E-01	5.13E-01	2.66E-01	1.38E-01
	Fe-55	1.25E-03	3.46E-04	9.58E-05	2.65E-05
	H-3	7.18E-01	5.42E-01	4.09E-01	3.09E-01
	Mn-54	8.38E-19	1.42E-20	2.41E-22	4.08E-24
	Nb-94a	1.88E-03	1.88E-03	1.88E-03	1.88E-03
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Greater Than Class C 5 YEAR ACTIVITY REPORT

REA P COMPONENT
PWR D PWR CORE SHROUD AND FORMER PLATES

NUCLID	1985	1990	1995	2000	2005	2010
C-14a	5.94E+02	5.94E+02	5.93E+02	5.93E+02	1.19E+03	2.38E+03
Co-60	3.44E+05	1.78E+05	9.22E+04	4.78E+04	8.87E+05	2.43E+06
Fe-55	1.83E+05	5.08E+04	1.41E+04	3.90E+03	1.11E+06	3.17E+06
Mn-54	3.93E+01	6.66E-01	1.13E-02	1.91E-04	1.18E+04	5.37E+04
Nb-94a	8.74E+00	8.74E+00	8.73E-00	8.73E-00	1.75E+01	3.79E+01
Ni-59a	1.47E+03	1.47E+03	1.47E+03	1.47E+03	2.94E+03	5.88E+03
Ni-63a	3.09E+03	2.98E+03	2.88E-03	2.78E+03	5.93E+03	1.22E+04
Am-241	3.18E-03	2.50E-03	1.96E-03	1.54E-03	5.68E-03	4.46E-03
C-14a	3.28E+00	3.27E+00	3.27E+00	3.27E+00	6.55E+00	6.54E+00
Ce-141	7.79E-01	1.49E-01	2.85E-02	5.46E-03	7.89E+00	1.51E+00
Ce-144	3.69E-03	4.31E-05	5.04E-07	5.89E-09	1.87E+00	2.19E-02
Cm-242	4.77E-08	1.81E-11	6.87E-15	2.61E-18	2.93E-03	1.11E-06
Co-58	3.50E-15	6.10E-23	1.06E-30	1.86E-38	2.55E-04	4.44E-12
Co-60	2.75E+02	1.43E+02	7.39E+01	3.83E+01	7.11E+02	3.68E+02
Cs-134	1.62E-01	3.02E-02	5.61E-03	1.04E-03	1.71E+00	3.18E-01
Cs-137	7.65E+00	6.82E+00	6.08E+00	5.42E+00	1.38E+01	1.23E+01
Fe-55	3.49E+01	9.67E+00	2.68E+00	7.42E-01	2.11E+02	5.84E+01
Fe-59	1.90E-26	8.72E-39	.00E+00	.00E+00	3.55E-09	1.63E-21
Mn-54	7.51E-03	1.27E-04	2.16E-06	3.66E-08	2.26E+00	3.84E-02
Nb-95	1.58E-33	.00E+00	.00E+00	.00E+00	1.41E-11	2.96E-27
Ni-63a	6.69E+01	6.46E+01	6.24E+01	6.03E+01	1.28E+02	1.24E+02
Pu-238	9.12E-10	3.46E-13	1.31E-16	4.98E-20	5.61E-05	2.13E-08
Pu-241	1.94E+00	1.52E+00	1.20E+00	9.40E-01	3.46E+00	2.72E+00
Sr-90	5.03E+00	4.46E+00	3.95E+00	3.50E+00	9.06E+00	8.03E+00
TRUS	1.50E+00	1.50E+00	1.50E+00	1.50E+00	3.00E+00	3.00E+00
Zn-65	1.84E-04	1.04E-06	5.83E-09	3.28E-11	2.60E-01	1.46E-03
C-14a	3.14E+01	3.13E+01	3.13E+01	3.13E+01	6.27E+01	1.25E+02
Co-60	2.78E+04	1.44E+04	7.45E+03	3.86E+03	7.17E+04	1.96E+05
Fe-55	1.00E+04	2.78E+03	7.70E+02	2.13E+02	6.06E+04	1.73E+05
Mn-54	1.54E+00	2.61E-02	4.42E-04	7.50E-06	4.64E+02	2.10E+03
Nb-94a	5.42E-01	5.42E-01	5.42E-01	5.42E-01	1.08E+00	2.17E+00
Ni-59a	1.84E+02	1.84E+02	1.84E+02	1.84E+02	3.68E+02	7.36E+02
Ni-63a	2.16E+04	2.09E+04	2.01E+04	1.95E+04	4.15E+04	8.57E+04

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O CARTRIDGE FILTER-OPERATIONAL	5.92E+05	2.72E+05	1.40E+05	8.04E+04	2.19E+06	6.13E+06
Am-241	9.32E-03	1.14E-02	1.29E-02	1.42E-02	1.52E-02	1.60E-02
C-14a	1.57E-01	1.57E-01	1.57E-01	1.57E-01	1.57E-01	1.57E-01
Cm-242	1.20E-06	4.56E-10	1.73E-13	6.56E-17	2.49E-20	9.45E-24
Co-60	3.14E+01	1.63E+01	8.42E+00	4.36E+00	2.26E+00	1.17E+00
Cs-137	8.64E-01	7.71E-01	6.87E-01	6.12E-01	5.46E-01	4.87E-01
H-3	1.35E-01	1.02E-01	7.70E-02	5.81E-02	4.39E-02	3.31E-02
LT5	2.59E+01	6.47E+00	1.62E+00	4.04E-01	1.01E-01	2.53E-02
Ni-63a	1.27E+02	1.68E+02	2.07E+02	2.45E+02	2.81E+02	3.17E+02
Pu-238	2.30E-08	8.72E-12	3.31E-15	1.26E-18	4.76E-22	1.81E-25
Pu-241	5.68E+00	6.91E+00	7.89E+00	8.65E+00	9.25E+00	9.72E+00
Sr-90	1.46E+00	1.66E+00	1.83E+00	1.99E+00	2.13E+00	2.25E+00
TRUS	1.20E-04	1.20E-04	1.20E-04	1.20E-04	1.20E-04	1.20E-04
TC-99	5.77E-01	7.83E-01	9.90E-01	1.20E+00	1.40E+00	1.61E+00
C-14a	7.75E+00	8.30E+00	8.84E+00	9.38E+00	9.93E+00	1.05E+01
Co-60	1.34E+05	7.47E+04	4.38E+04	2.78E+04	1.95E+04	1.52E+04
Cs-137	7.16E-03	6.38E-03	5.69E-03	5.07E-03	4.52E-03	4.03E-03
Fe-55	6.17E+03	6.11E+03	6.09E+03	6.08E+03	6.08E+03	6.08E+03
H-3	3.04E+02	3.27E+02	3.44E+02	3.57E+02	3.67E+02	3.75E+02
LT5	3.99E+05	9.97E+04	2.49E+04	6.23E+03	1.56E+03	3.90E+02
Mn-54	3.57E+01	3.57E+01	3.57E+01	3.57E+01	3.57E+01	3.57E+01
Nb-94a	1.69E+00	1.71E+00	1.72E+00	1.74E+00	1.76E+00	1.78E+00

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CONTROL ROD DRIVE

PWR D PWR CORE SHROUD AND FORMER PLATES

REA P COMPONENT	Greater Than Class C 5 YEAR ACTIVITY REPORT	2020					2025					2030					2035					
		NUCLID	2015					2015					2015					2015				
PWR D PWR CORE SHROUD AND FORMER PLATES	C-14a		6.45E+03		1.98E+04		2.37E+04		3.26E+04		3.56E+04											
	Co-60		7.78E+06		2.36E+07		1.80E+07		2.46E+07		1.71E+07											
	Fe-55		9.04E+06		2.35E+07		1.26E+07		1.95E+07		8.91E+06											
	Mn-54		1.28E+05		1.97E+05		6.29E+04		1.68E+05		1.60E+04											
	Nb-94a		1.01E+02		3.24E+02		3.87E+02		5.51E+02		6.06E+02											
	Ni-59a		1.75E+04		5.99E+04		7.23E+04		1.05E+05		1.17E+05											
Ni-63a		8.26E+05		5.29E+06		6.38E+06		1.09E+07		1.25E+07												

PWR FULL SYSTEM FUEL-IN RESINS

Am-241	4.42E-02	1.49E-01	1.59E-01	2.12E-01	1.97E-01
C-14a	3.60E+01	1.25E+02	1.57E+02	2.23E+02	2.49E+02
Ce-141	6.62E+01	1.69E+02	9.50E+01	1.51E+02	6.01E+01
Ce-144	1.92E+01	3.62E+01	1.44E+01	3.24E+01	3.06E+00
Cm-242	4.97E-02	8.57E-02	2.72E-02	7.31E-02	1.60E-03
Co-58	2.77E-02	4.64E-02	1.01E-02	3.74E-02	1.47E-05
Co-60	6.08E+03	1.92E+04	1.61E+04	2.10E+04	1.50E+04
Cs-134	1.44E+01	3.65E+01	2.03E+01	3.26E+01	1.27E+01
Cs-137	9.09E+01	3.17E+02	3.70E+02	5.07E+02	5.20E+02
Fe-55	1.77E+03	4.87E+03	3.08E+03	4.50E+03	2.21E+03
Fe-59	3.13E-06	5.22E-06	1.06E-06	4.18E-06	2.43E-11
Mn-54	2.23E+01	4.30E+01	1.75E+01	3.86E+01	4.33E+00
Nb-95	5.79E-08	9.66E-08	1.94E-08	7.73E-08	2.07E-14
Ni-63a	7.50E+02	2.60E+03	3.21E+03	4.50E+03	4.90E+03
Pu-238	1.91E-02	3.19E-02	6.58E-03	2.56E-02	3.05E-05
Pu-241	2.60E+01	8.96E+01	9.64E+01	1.28E+02	1.20E+02
Sr-90	6.01E+01	2.09E+02	2.44E+02	3.33E+02	3.40E+02
TRU5	1.65E+01	5.70E+01	7.20E+01	1.02E+02	1.14E+02
Zn-65	2.93E+00	5.34E+00	2.02E+00	4.75E+00	3.23E-01

PWR LOWER CORE BARREL

[illegible]

Q CARTRIDGE FILTER-OPERATIONAL

O CARTRIDGE FILTER-OPERATIONAL									
Am-241	1.48E-02	1.16E-02	9.15E-03	7.19E-03	5.64E-03				
C-14a	1.57E-01	1.57E-01	1.57E-01	1.56E-01	1.56E-01				
Cm-242	3.59E-27	1.36E-30	5.17E-34	1.94E-37	0.00E+00				
Co-60	6.07E-01	3.14E-01	1.63E-01	8.44E-02	4.37E-02				
Cs-137	4.34E-01	3.87E-01	3.45E-01	3.07E-01	2.74E-01				
H-3	2.50E-02	1.89E-02	1.43E-02	1.08E-02	8.12E-03				
LTS	6.31E-03	1.58E-03	3.95E-04	9.87E-05	2.47E-05				
Ni-63a	3.33E+02	3.21E+02	3.11E+02	3.00E+02	2.90E+02				
Pu-238	6.86E-29	2.60E-32	9.88E-36	3.72E-39	0.00E+00				
Pu-241	9.03E+00	7.09E+00	5.57E+00	4.38E+00	3.44E+00				
Sr-90	2.21E+00	1.96E+00	1.73E+00	1.54E+00	1.36E+00				
TRU5	1.20E-04	1.20E-04	1.20E-04	1.20E-04	1.20E-04				
TC-99	1.73E+00	1.73E+00	1.73E+00	1.73E+00	1.73E+00				

CONTROL ROD DRIVE

[illegible]

ND-548 1.78E+00 1.77E+00 1.
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Greater Than Class C 5 YEAR ACTIVITY REPORT

REA P COMPONENT

PWR O CONTROL ROD DRIVE

NUCLID	1985	1990	1995	2000	2005	2010
Ni-59a	2.78E+02	2.81E+02	2.84E+02	2.87E+02	2.89E+02	2.92E+02
Ni-63a	3.97E+04	3.85E+04	3.73E+04	3.62E+04	3.51E+04	3.41E+04
Sr-90	7.08E-03	6.27E-03	5.55E-03	4.90E-03	4.36E-03	3.86E-03
Tc-99	9.00E-04	9.00E-04	9.00E-04	9.00E-04	9.00E-04	9.00E-04
Co-60	2.59E+03	1.34E+03	6.95E+02	3.60E+02	1.87E+02	9.67E+01
LT5	7.00E-02	1.75E-02	4.38E-03	1.09E-03	2.74E-04	6.84E-05
Ni-59a	1.52E+02	1.52E+02	1.52E+02	1.52E+02	1.52E+02	1.52E+02
Ni-63a	2.16E+04	2.09E+04	2.02E+04	1.95E+04	1.88E+04	1.82E+04
TRU5	3.34E-02	3.34E-02	3.34E-02	3.34E-02	3.34E-02	3.34E-02
TRU5	2.34E+06	4.16E+06	6.20E+06	8.26E+06	1.03E+07	1.23E+07

PRIMARY/SECONDARY SOURCE

PWR CRUD TANK FILTERS

Am-241	9.48E-04	1.27E-03	1.53E-03	1.73E-03	1.89E-03	2.01E-03
C-14a	8.73E-01	1.30E+00	1.73E+00	2.16E+00	2.59E+00	3.02E+00
Co-58	1.38E-01	1.38E-01	1.38E-01	1.38E-01	1.38E-01	1.38E-01
Co-60	2.39E+01	2.81E+01	3.03E+01	3.15E+01	3.21E+01	3.24E+01
Cs-137	2.84E-02	4.02E-02	5.07E-02	6.01E-02	6.84E-02	7.59E-02
Fe-55	1.24E+01	1.32E+01	1.35E+01	1.35E+01	1.35E+01	1.35E+01
H-3	1.05E-01	1.38E-01	1.64E-01	1.83E-01	1.97E-01	2.08E-01
Mn-54	1.14E-01	1.14E-01	1.14E-01	1.14E-01	1.14E-01	1.14E-01
Ni-63a	9.42E+00	1.38E+01	1.81E+01	2.22E+01	2.62E+01	3.00E+01
Pu-241	5.77E-01	7.75E-01	9.31E-01	1.05E+00	1.15E+00	1.22E+00
Sr-90	1.41E-02	1.99E-02	2.51E-02	2.97E-02	3.37E-02	3.73E-02
TRU5	5.10E-02	7.62E-02	1.01E-01	1.27E-01	1.52E-01	1.77E-01
Zn-65	2.62E-02	2.62E-02	2.62E-02	2.62E-02	2.62E-02	2.62E-02

PWR IN-CORE INSTRUMENTS

C-14a	2.07E+01	3.43E+01	5.03E+01	6.66E+01	8.27E+01	9.83E+01
Co-60	1.03E+05	1.51E+05	1.90E+05	2.13E+05	2.22E+05	2.24E+05
Cs-137	2.85E-03	2.54E-03	2.26E-03	2.02E-03	1.80E-03	1.60E-03
Fe-55	4.33E+04	6.44E+04	7.90E+04	8.41E+04	8.43E+04	8.18E+04
H-3	2.83E-02	2.14E-02	1.62E-02	1.22E-02	9.21E-03	6.95E-03
LT5	4.37E+03	1.09E+03	2.73E+02	6.82E+01	1.71E+01	4.26E+00
Mn-54	1.95E+02	3.03E+02	3.40E+02	3.43E+02	3.37E+02	3.17E+02
Nb-94a	3.91E-02	6.43E-02	9.25E-02	1.21E-01	1.49E-01	1.77E-01
Ni-59a	2.62E+02	3.84E+02	5.24E+02	6.67E+02	8.08E+02	9.46E+02
Ni-63a	3.29E+04	4.63E+04	6.14E+04	7.64E+04	9.06E+04	1.04E+05
Sr-90	2.81E-03	2.49E-03	2.21E-03	1.96E-03	1.73E-03	1.54E-03
TRU5	1.67E-02	1.67E-02	1.67E-02	1.67E-02	1.67E-02	1.67E-02
Tc-99	3.50E-04	3.50E-04	3.50E-04	3.50E-04	3.50E-04	3.50E-04

PWR MISC. CARTRIDGE FILTERS

Am-241	6.50E-03	1.01E-02	1.35E-02	1.64E-02	1.85E-02	2.01E-02
C-14a	6.17E+00	1.02E+01	1.48E+01	1.94E+01	2.40E+01	2.85E+01
Co-58	1.02E+00	1.42E+00	1.48E+00	1.50E+00	1.48E+00	1.40E+00
Co-60	1.60E+02	2.33E+02	2.89E+02	3.21E+02	3.35E+02	3.38E+02
Cs-137	1.97E-01	3.16E-01	4.40E-01	5.54E-01	6.53E-01	7.38E-01
Fe-55	8.38E+01	1.18E+02	1.37E+02	1.44E+02	1.45E+02	1.42E+02
H-3	7.15E-01	1.10E+00	1.46E+00	1.75E+00	1.95E+00	2.10E+00
Mn-54	8.13E-01	1.13E+00	1.22E+00	1.24E+00	1.22E+00	1.17E+00
Ni-63a	6.62E+01	1.08E+02	1.55E+02	2.01E+02	2.45E+02	2.86E+02
Pu-241	3.96E+00	6.15E+00	8.25E+00	9.97E+00	1.13E+01	1.22E+01
Sr-90	9.82E-02	1.57E-01	2.18E-01	2.74E-01	3.22E-01	3.63E-01
TRU5	3.61E-01	5.97E-01	8.65E-01	1.14E+00	1.41E+00	1.67E+00
Zn-65	1.89E-01	2.64E-01	2.80E-01	2.84E-01	2.81E-01	2.69E-01

THIMBLE PLUGS OR ASSEMBLY

C-14a	7.66E+00	9.43E+00	1.15E+01	1.36E+01	1.57E+01	1.78E+01
Co-60	2.13E+04	1.97E+04	2.02E+04	2.07E+04	2.08E+04	2.05E+04
Fe-55	8.20E+03	1.21E+04	1.46E+04	1.56E+04	1.57E+04	1.53E+04
LT5	6.25E+02	1.56E+02	3.91E+01	9.77E+00	2.44E+00	6.10E-01
Mn-54	4.04E+01	6.15E+01	6.77E+01	6.94E+01	6.81E+01	6.42E+01
Nb-94a	1.82E-01	1.96E-01	2.13E-01	2.30E-01	2.47E-01	2.63E-01

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Greater Than Class C 5 YEAR ACTIVITY REPORT

REA P COMPONENT	NUCLID	2015	2020	2025	2030	2035
PWR C CONTROL ROD DRIVE	Ni-59a	2.92E+02	2.92E+02	2.92E+02	2.92E+02	2.92E+02
	Ni-63a	3.29E+04	3.18E+04	3.07E+04	2.97E+04	2.87E+04
	Sr-90	3.42E-03	3.03E-03	2.68E-03	2.38E-03	2.11E-03
	Tc-99	9.00E-04	9.00E-04	9.00E-04	9.00E-04	9.00E-04
FLUX WIRE	Co-60	5.01E+01	2.60E+01	1.34E+01	6.97E+00	3.61E+00
	LT5	1.71E-05	4.28E-06	1.07E-06	2.67E-07	6.68E-08
	Ni-59a	1.52E+02	1.52E+02	1.52E+02	1.52E+02	1.52E+02
	Ni-63a	1.76E+04	1.70E+04	1.54E+04	1.58E+04	1.53E+04
	TRU5	3.34E-02	3.34E-02	3.34E-02	3.34E-02	3.34E-02
PRIMARY/SECONDARY SOURCE	TRU5	1.40E+07	1.51E+07	1.59E+07	1.60E+07	1.60E+07
PWR CRUD TANK FILTERS	Am-241	1.93E-03	1.56E-03	1.22E-03	9.61E-04	7.54E-04
	C-14a	3.31E+00	3.35E+00	3.35E+00	3.34E+00	3.34E+00
	Co-58	3.95E-02	4.49E-07	7.83E-15	1.36E-22	2.38E-30
	Co-60	2.67E+01	1.49E+01	7.72E+00	4.00E+00	2.07E+00
	Cs-137	7.77E-02	7.05E-02	6.28E-02	5.60E-02	4.99E-02
	Fe-55	9.39E+00	3.12E+00	8.64E-01	2.39E-01	6.63E-02
	H-3	1.97E-01	1.53E-01	1.16E-01	8.72E-02	6.58E-02
	Mn-54	4.67E-02	2.27E-03	3.85E-05	6.53E-07	1.11E-08
	Ni-63a	3.22E+01	3.15E+01	3.05E+01	2.94E+01	2.84E+01
	Pu-241	1.18E+00	9.48E-01	7.45E-01	5.85E-01	4.60E-01
	Sr-90	3.80E-02	3.43E-02	3.04E-02	2.69E-02	2.38E-02
	TRU5	1.94E-01	1.96E-01	1.96E-01	1.96E-01	1.96E-01
	Zn-65	9.53E-03	2.38E-04	1.34E-06	7.53E-09	4.24E-11
PWR IN-CORE INSTRUMENTS	C-14a	1.11E+02	1.19E+02	1.26E+02	1.28E+02	1.28E+02
	Co-60	2.01E+05	1.62E+05	1.26E+05	7.63E+04	4.00E+04
	Cs-137	1.43E-03	1.27E-03	1.14E-03	1.01E-03	9.02E-04
	Fe-55	6.74E+04	5.18E+04	3.92E+04	1.75E+04	5.12E+03
	H-3	5.25E-03	3.96E-03	2.99E-03	2.26E-03	1.70E-03
	LT5	1.07E+00	2.66E-01	6.66E-02	1.67E-02	4.16E-03
	Mn-54	2.30E+02	1.79E+02	1.25E+02	2.53E+01	7.13E-01
	Nb-94a	2.00E-01	2.14E-01	2.23E-01	2.26E-01	2.26E-01
	Ni-59a	1.06E+03	1.13E+03	1.18E+03	1.20E+03	1.20E+03
	Ni-63a	1.14E+05	1.18E+05	1.21E+05	1.18E+05	1.14E+05
	Sr-90	1.36E-03	1.20E-03	1.07E-03	9.46E-04	8.38E-04
	TRU5	1.67E-02	1.67E-02	1.67E-02	1.67E-02	1.67E-02
	Tc-99	3.50E-04	3.50E-04	3.50E-04	3.50E-04	3.50E-04
PWR MISC. CARTRIDGE FILTERS	Am-241	2.01E-02	1.86E-02	1.65E-02	1.35E-02	1.07E-02
	C-14a	3.21E+01	3.44E+01	3.59E+01	3.64E+01	3.64E+01
	Co-58	9.30E-01	6.72E-01	3.39E-01	4.06E-02	5.54E-04
	Co-60	3.03E+02	2.41E+02	1.80E+02	1.07E+02	5.80E+01
	Cs-137	7.81E-01	7.76E-01	7.46E-01	6.79E-01	6.08E-01
	Fe-55	1.17E+02	8.35E+01	5.64E+01	2.35E+01	7.83E+00
	H-3	2.07E+00	1.88E+00	1.63E+00	1.29E+00	9.83E-01
	Mn-54	8.31E-01	5.76E-01	3.41E-01	6.30E-02	9.12E-03
	Ni-63a	3.16E+02	3.31E+02	3.36E+02	3.30E+02	3.19E+02
	Pu-241	1.23E+01	1.14E+01	1.01E+01	8.22E+00	6.51E+00
	Sr-90	3.83E-01	3.79E-01	3.63E-01	3.29E-01	2.92E-01
	TRU5	1.88E+00	2.02E+00	2.11E+00	2.13E+00	2.14E+00
	Zn-65	1.86E-01	1.31E-01	7.48E-02	1.21E-02	1.52E-03
THIMBLE PLUGS OR ASSEMBLY	C-14a	1.94E+01	2.06E+01	2.14E+01	2.16E+01	2.17E+01
	Co-60	1.84E+04	1.51E+04	1.17E+04	7.21E+03	3.86E+03
	Fe-55	1.27E+04	9.70E+03	7.15E+03	3.18E+03	1.00E+03
	LT5	1.53E-01	3.81E-02	9.54E-03	2.38E-03	5.96E-04
	Mn-54	4.74E+01	3.63E+01	2.45E+01	4.96E+00	3.46E-01
	Nb-94a	2.77E-01	2.86E-01	2.93E-01	2.95E-01	2.95E-01

AREA P COMPONENT		Greater Than Class C 5 YEAR ACTIVITY REPORT						
		NUCLID	1985	1990	1995	2000	2005	2010
PWR O THIMBLE PLUGS OR ASSEMBLY		NI-59a	4.16E+01	5.16E+01	6.33E+01	7.54E+01	8.73E+01	9.88E+01
		NI-63a	5.83E+03	6.81E+03	7.96E+03	9.12E+03	1.02E+04	1.12E+04
*****		*****	3.16E+06	4.71E+06	6.71E+06	8.78E+06	1.08E+07	1.28E+07

REA P COMPONENT		Greater Than Class C 5 YEAR ACTIVITY REPORT				
		NUCLID	2015	2020	2025	2030
PWR C THIMBLE PLUGS OR ASSEMBLY						
		Ni-59a	1.08E+02	1.15E+02	1.19E+02	1.21E+02
		Ni-63a	1.19E+04	1.23E+04	1.24E+04	1.22E+04
*****		*****	1.45E+07	1.56E+07	1.62E+07	1.63E+07
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Greater Than Class C 5 YEAR ACTIVITY REPORT

REA P COMPONENT

D COMPACTIBLE TRASH-DECOMMISSION

NUCLID	1985	1990	1995	2000	2005	2010
Am-241	8.01E+02	7.94E+02	7.88E+02	7.81E+02	7.74E+02	7.68E+02
Cm-242	1.66E+02	6.28E-02	2.38E-05	9.05E-09	3.43E-12	1.30E-15
Pu-238	3.17E+00	1.20E+03	4.56E-07	1.73E-10	6.57E-14	2.49E-17
Pu-241	1.52E+03	1.20E+03	9.41E-02	7.39E+02	5.80E+02	4.56E+02
TRU5	8.00E+02	8.00E+02	8.00E+02	8.00E+02	8.00E+02	8.00E+02

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C CARTRIDGE FILTER-OPERATIONAL

Am-241	3.29E+03	2.79E+03	2.53E+03	2.32E+03	2.15E+03	2.02E+03
Co-60	1.02E+00	1.01E+00	9.98E-01	9.98E-01	9.98E-01	9.98E-01
Cs-137	1.14E+01	5.91E+00	1.58E+00	3.06E+00	1.58E+00	8.21E-01
I-129	3.16E+01	2.81E+01	2.51E+01	2.51E+01	2.24E+01	1.99E+01
LT5	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00
Ni-59a	7.50E-01	1.88E-01	4.69E-02	1.17E-02	1.17E-02	2.93E-03
Ni-63a	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00
Pu-241	9.33E-01	9.01E-01	8.71E-01	8.71E-01	8.71E-01	8.41E-01
Sr-90	7.85E-01	6.17E-01	4.85E-01	3.81E-01	3.81E-01	2.99E-01
TRU5	2.66E+00	2.36E+00	2.09E+00	1.85E+00	1.85E+00	1.64E+00
Tc-99	4.00E+00	4.00E+00	4.00E+00	4.00E+00	4.00E+00	4.00E+00
	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00

COMPACTIBLE TRASH-OPERATIONS

Am-241	1.56E+02	1.64E+02	1.63E+02	1.62E+02	1.60E+02	1.59E+02
Cm-242	3.79E-03	1.44E-06	5.47E-10	5.47E-10	2.07E-13	7.87E-17
Co-60	5.18E-04	2.68E-04	1.39E-04	1.39E-04	7.20E-05	3.73E-05
Cs-137	1.93E+01	1.72E+01	1.53E+01	1.36E+01	1.22E+01	1.22E+01
Pu-238	7.26E-05	2.75E-08	1.05E-11	3.97E-15	3.97E-15	1.51E-18
Pu-241	6.52E+01	5.26E+01	4.13E+01	3.24E+01	3.24E+01	2.55E+01
Sr-90	8.86E-04	7.85E-04	6.16E-04	6.16E-04	6.16E-04	5.46E-04
TRU5	3.20E+00	1.32E+01	1.32E+01	1.32E+01	1.32E+01	1.32E+01

CONTAMINATED EQUIPMENT-OPERATIONS

Am-241	2.32E+01	2.22E+02	2.20E+02	2.18E+02	2.16E+02	2.15E+02
Cm-242	7.59E-02	2.88E-05	1.09E-08	4.15E-12	4.15E-12	1.57E-15
Pu-238	1.45E-03	5.51E-07	2.09E-10	7.93E-14	7.93E-14	3.01E-17
Pu-241	3.14E+02	2.47E+02	1.94E+02	1.52E+02	1.52E+02	1.20E+02
TRU5	2.00E+02	2.00E+02	2.00E+02	2.00E+02	2.00E+02	2.00E+02

CONTROL ROD BLADES - NOT CONTROL CELL

Ni-59a	7.40E-01	1.11E+00	1.41E+00	1.41E+00	1.41E+00	1.41E+00
Ni-63a	1.24E+00	6.24E+01	1.09E+02	1.05E+02	1.02E+02	9.83E+01

LEWATERED ION EXCHANGE RESIN

Am-241	1.00E+00	9.92E-01	9.84E-01	9.76E-01	9.68E-01	9.60E-01
Co-60	5.26E+00	2.73E+00	1.41E+00	7.31E-01	3.79E-01	1.96E-01
Cs-137	3.91E+00	3.48E+00	3.11E+00	2.77E+00	2.47E+00	2.20E+00
I-129	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00
LT5	4.55E+00	1.14E+00	2.84E-01	7.10E-02	1.78E-02	4.44E-03
Ni-59a	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00
Ni-63a	9.93E-01	9.59E-01	9.27E-01	8.95E-01	8.65E-01	8.35E-01
Pu-241	1.43E+00	1.12E+00	8.82E-01	6.93E-01	5.44E-01	4.27E-01
Sr-90	9.76E-01	8.65E-01	7.66E-01	6.79E-01	6.01E-01	5.33E-01
TRU5	1.50E-01	1.50E-01	1.50E-01	1.50E-01	1.50E-01	1.50E-01

FILTER MEDIA

Am-241	2.32E+01	2.30E+01	2.28E+01	2.26E+01	2.24E+01	2.23E+01
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METAL SHAVINGS

Am-241	9.94E-01	9.86E-01	9.86E-01	9.77E-01	9.69E-01	9.61E-01
C-14a	9.99E-01	9.99E-01	9.99E-01	9.99E-01	9.98E-01	9.97E-01
Co-60	4.66E+00	2.42E+00	1.25E+00	6.48E-01	3.36E-01	3.36E-01
Cs-137	3.57E+00	3.18E+00	2.83E+00	2.53E+00	2.25E+00	2.25E+00
I-129	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00
LT5	1.25E+00	3.13E-01	7.81E-02	1.95E-02	4.88E-03	1.00E+00
Ni-59a	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00
Ni-63a	9.66E-01	9.33E-01	8.71E-01	8.41E-01	8.41E-01	8.41E-01
Pu-241	1.18E+00	9.25E-01	7.27E-01	5.71E-01	4.48E-01	4.48E-01

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Greater Than Class C 5 YEAR ACTIVITY REPORT		2015			2020			2025			2030			2035		
REA P	COMPONENT	NUCLID														
D COMPACTIBLE TRASH-DECOMMISSION																
		Am-241	7.62E+02	7.56E+02	7.49E+02	7.43E+02	7.37E-02									
		Cm-242	4.94E-19	1.88E-22	7.12E-26	2.70E-29	1.03E-32									
		Pu-238	9.46E-21	3.59E-24	1.36E-27	5.17E-31	1.96E-34									
		Pu-241	3.58E+02	2.81E+02	2.21E+02	1.73E+02	1.36E+02									
		TRU5	8.00E+02	8.00E+02	8.00E+02	8.00E+02	8.00E-02									
O CARTRIDGE FILTER-OPERATIONAL																
		Am-241	9.82E-01	9.74E-01	9.66E-01	9.58E-01	9.51E-01									
		Co-60	4.25E-01	2.20E-01	1.14E-01	5.92E-02	3.06E-02									
		Cs-137	1.78E+01	1.58E+01	1.41E+01	1.26E+01	1.12E+01									
		I-129	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00									
		LT5	7.32E-04	1.83E-04	4.58E-05	1.14E-05	2.86E-06									
		Ni-59a	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00									
		Ni-63a	8.12E-01	7.85E-01	7.58E-01	7.32E-01	7.07E-01									
		Pu-241	2.35E-01	1.84E-01	1.45E-01	1.14E-01	8.94E-02									
		Sr-90	1.45E+00	1.28E+00	1.14E+00	1.01E+00	8.93E-01									
		TRU5	4.00E+00	4.00E+00	4.00E+00	4.00E+00	4.00E+00									
		Tc-99	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00									
COMPACTIBLE TRASH-OPERATIONS																
		Am-241	1.58E+02	1.57E+02	1.55E+02	1.54E+02	1.53E+02									
		Cm-242	2.99E-20	1.13E-23	4.30E-27	1.63E-30	6.20E-34									
		Co-60	1.93E-05	1.00E-05	5.19E-06	2.69E-06	1.39E-06									
		Cs-137	1.08E+01	9.67E+00	8.62E+00	7.68E+00	6.85E+00									
		Pu-238	5.71E-22	2.17E-25	8.23E-29	3.12E-32	1.18E-35									
		Pu-241	2.00E+01	1.57E+01	1.23E+01	9.69E+00	7.61E+00									
		Sr-90	4.83E-04	2.28E-04	3.79E-04	3.36E-04	2.98E-04									
		TRU5	1.32E+01	1.32E+01	1.32E+01	1.32E+01	1.32E+01									
CONTAMINATED EQUIPMENT-OPERATIONS																
		Am-241	2.13E+02	2.11E+02	2.09E+02	2.08E+02	2.06E+02									
		Cm-242	5.97E-19	2.27E-22	8.60E-26	3.26E-29	1.24E-32									
		Pu-238	1.14E-20	4.33E-24	1.65E-27	6.24E-31	2.37E-34									
		Pu-241	9.39E+01	7.38E+01	5.79E+01	4.55E+01	3.57E+01									
		TRU5	2.00E+02	2.00E+02	2.00E+02	2.00E+02	2.00E+02									
CONTROL ROD BLADES - NOT CONTROL CELL																
		Ni-59a	1.41E+00	1.41E+00	1.41E+00	1.41E+00	1.41E+00									
		Ni-63a	9.50E+01	9.18E+01	8.86E+01	8.56E+01	8.27E+01									
DEWATERED ION EXCHANGE RESIN																
		Am-241	9.52E-01	9.44E-01	9.37E-01	9.29E-01	9.22E-01									
		Co-60	1.02E-01	5.27E-02	2.73E-02	1.41E-02	7.33E-03									
		Cs-137	1.96E+00	1.75E+00	1.56E+00	1.39E+00	1.24E+00									
		I-129	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00									
		LT5	1.11E-03	2.78E-04	6.94E-05	1.73E-05	4.34E-06									
		Ni-59a	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00									
		Ni-63a	8.07E-01	7.79E-01	7.53E-01	7.27E-01	7.02E-01									
		Pu-241	3.36E-01	2.64E-01	2.07E-01	1.63E-01	1.28E-01									
		Sr-90	4.72E-01	4.18E-01	3.70E-01	3.28E-01	2.91E-01									
		TRU5	1.50E-01	1.50E-01	1.50E-01	1.50E-01	1.50E-01									
FILTER MEDIA																
		Am-241	2.21E+01	2.19E+01	2.17E+01	2.16E+01	2.14E+01									
METAL SHAVINGS																
		Am-241	9.54E-01	9.46E-01	9.38E-01	9.31E-01	9.23E-01									
		C-14a	9.96E-01	9.96E-01	9.95E-01	9.95E-01	9.94E-01									
		Co-60	1.74E-01	9.02E-02	4.67E-02	2.42E-02	1.25E-02									
		Cs-137	2.01E+00	1.79E+00	1.60E+00	1.42E+00	1.27E+00									
		I-129	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00									
		LT5	1.22E-03	3.05E-04	7.63E-05	1.91E-05	4.77E-06									
		Ni-59a	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00									
		Ni-63a	8.12E-01	7.85E-01	7.58E-01	7.32E-01	7.07E-01									
		Pu-241	3.52E-01	2.77E-01	2.17E-01	1.71E-01	1.34E-01									

REA P COMPONENT		Greater Than Class C 5 YEAR ACTIVITY REPORT					
	NUCLID	1985	1990	1995	2000	2005	2010
O METAL SHAVINGS							
	Sr-90		8.86E-01	7.85E-01	6.95E-01	6.16E-01	5.46E-01
	TRU5		1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00
	Tc-99		1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00
METALS-OPERATIONS							
	Am-241	5.02E+00	8.71E+00	1.24E+01	1.60E+01	1.96E+01	2.32E+01
	Sr-90	1.02E+00	9.13E-01	8.13E-01	7.25E-01	6.47E-01	5.78E-01
	TRU5	1.00E-03	1.50E-03	2.00E-03	2.50E-03	3.00E-03	3.50E-03
MIXED-OXIDE FUEL PELLETS/RODS							
	Am-241	3.42E+01	3.37E+01	3.33E+01	3.29E+01	3.25E+01	3.22E+01
	Pu-238	1.40E+01	1.34E+01	1.29E+01	1.24E+01	1.19E+01	1.15E+01
	Pu-239	2.05E+02	2.05E+02	2.05E+02	2.05E+02	2.05E+02	2.05E+02
	Pu-240	4.95E+00	4.95E+00	4.94E+00	4.94E+00	4.94E+00	4.94E+00
	Pu-241	5.98E+02	4.70E+02	3.69E+02	2.90E+02	2.28E+02	1.79E+02
	TRU5	3.88E+01	3.88E+01	3.88E+01	3.88E+01	3.88E+01	3.88E+01
NONCOMPACTIBLE TRASH-OPERATIONS							
	Am-241	3.53E+01	5.50E+01	5.45E+01	5.41E+01	5.36E+01	5.32E+01
	Cm-242		7.59E-03	2.88E-06	1.09E-09	4.15E-13	1.57E-16
	Pu-238		1.45E-04	5.51E-08	2.09E-11	7.93E-15	3.01E-18
	Pu-241		3.14E+01	2.47E+01	1.94E+01	1.52E+01	1.20E+01
	TRU5		2.00E+01	2.00E+01	2.00E+01	2.00E+01	2.00E+01
ORGANIC LIQUID							
	Am-241	9.98E-01	1.74E+00	2.47E+00	3.20E+00	3.92E+00	4.63E+00
	C-14a	1.20E+01	1.20E+01	1.20E+01	1.20E+01	1.20E+01	1.20E+01
	C-14n	1.02E+01	1.02E+01	1.02E+01	1.02E+01	1.02E+01	1.02E+01
	Sr-90	2.05E-01	2.05E-01	2.05E-01	2.05E-01	2.05E-01	2.04E-01
	TRU5	1.00E-03	1.50E-03	2.00E-03	2.50E-03	3.00E-03	3.50E-03
SOLIDIFIED RESIN							
	Am-241		2.98E-02	2.95E-02	2.93E-02	2.91E-02	2.88E-02
	Co-60		5.18E-04	2.68E-04	1.39E-04	7.20E-05	3.73E-05
	Cs-137		4.81E+00	4.29E+00	3.83E+00	3.41E+00	3.04E+00
	Sr-90		8.86E-04	7.85E-04	6.95E-04	6.16E-04	5.46E-04
	TRU5		1.00E-03	1.00E-03	1.00E-03	1.00E-03	1.00E-03
*****		1.25E+03	2.09E+03	1.92E+03	1.76E+03	1.63E+03	1.53E+03
*****		-----					

S

Greater Than Class C 5 YEAR ACTIVITY REPORT		NUCLID			
REA P COMPONENT		2015			
		2020			
		2025			
		2030			
		2035			
O METAL SHAVINGS					
Sr-90		4.83E-01	4.28E-01	3.79E-01	3.36E-01
TRU5		1.00E+00	1.00E+00	1.00E+00	1.00E+00
Tc-99		1.00E+00	1.00E+00	1.00E+00	1.00E+00
METALS-OPERATIONS					
Am-241		2.67E+01	3.03E+01	3.30E+01	3.27E+01
Sr-90		5.16E-01	4.62E-01	4.13E-01	3.66E-01
TRU5		4.00E-03	4.50E-03	4.90E-03	4.90E-03
MIXED-OXIDE FUEL PELLETS/RODS					
Am-241		3.19E+01	3.16E+01	3.13E+01	3.10E+01
Pu-238		1.10E+01	1.06E+01	1.02E+01	9.80E+00
Pu-239		2.05E+02	2.05E+02	2.05E+02	2.05E+02
Pu-240		4.93E+00	4.93E+00	4.93E+00	4.93E+00
Pu-241		1.41E+02	1.10E+02	8.67E+01	6.81E+01
TRU5		3.88E+01	3.88E+01	3.88E+01	3.88E+01
NONCOMPACTIBLE TRASH-OPERATIONS					
Am-241		5.28E+01	5.23E+01	5.19E+01	5.15E+01
Cm-242		5.97E-20	2.27E-23	8.60E-27	3.26E-30
Pu-238		1.14E-21	4.33E-25	1.65E-28	6.24E-32
Pu-241		9.39E+00	7.38E+00	5.79E+00	4.55E+00
TRU5		2.00E+01	2.00E+01	2.00E+01	2.00E+01
ORGANIC LIQUID					
Am-241		5.34E+00	6.05E+00	6.59E+00	6.54E+00
C-14a		1.20E+01	1.19E+01	1.19E+01	1.19E+01
C-14n		1.02E+01	1.02E+01	1.02E+01	1.02E+01
Sr-90		2.04E-01	2.04E-01	1.99E-01	1.77E-01
TRU5		4.00E-03	4.50E-03	4.90E-03	4.90E-03
SOLIDIFIED RESIN					
Am-241		2.86E-02	2.84E-02	2.81E-02	2.79E-02
Co-60		1.93E-05	1.00E-05	5.19E-06	2.69E-06
Cs-137		2.71E+00	2.42E+00	2.15E+00	1.92E+00
Sr-90		4.83E-04	4.28E-04	3.79E-04	3.36E-04
TRU5		1.00E-03	1.00E-03	1.00E-03	1.00E-03

		1.44E+03	1.38E+03	1.33E+03	1.28E+03

Greater-Than-Class C Low-Level Radioactive Waste Characterization

Appendix F: Greater-Than-Class C Low-Level Radioactive Waste Light Water Reactor Projections

*Greater-Than-Class C Low-Level Waste
Management Program*

August 1991

**GREATER-THAN-CLASS C
LOW-LEVEL WASTE CHARACTERIZATION**

**Appendix F: Greater-Than-Class C
Low-Level Radioactive Waste
Light Water Reactor Projections**

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ABSTRACT

This study characterizes potential greater-than-Class C low-level radioactive waste streams, estimates the amounts of waste generated, and estimates their radionuclide content and distribution. Several types of low-level radioactive wastes produced by light water reactors were identified in an earlier study as being potential greater-than-Class C low-level waste, including specific activated metal components and certain process wastes in the form of cartridge filters and decontamination resins. Light water reactor operating parameters and current management practices at operating plants were reviewed and used to estimate the amounts of potential greater-than-Class C low-level waste generated per fuel cycle. The amounts of routinely generated activated metal components and process waste were estimated as a function of fuel cycle. Component-specific radionuclide content and distribution was calculated for activated metals components. Empirical data from actual low-level radioactive waste streams were used to estimate radionuclide content and distribution for process wastes. The greater-than-Class C low-level waste volumes that could be generated through plant closure were also estimated, along with volumes and activities for potential greater-than-Class C activated metals generated at decommissioning.

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APPENDIX F: GREATER-THAN-CLASS C LOW-LEVEL RADIOACTIVE WASTE LIGHT WATER REACTOR PROJECTIONS

1. INTRODUCTION

The objective of this study is to characterize potential greater-than-Class C (GTCC) low-level radioactive waste streams identified in an earlier WMG, Inc., study,¹ estimate the amounts of waste generated, and estimate their radionuclide content and distribution. The resultant estimates will be used by EG&G Idaho, Inc., as input to the Department of Energy (DOE) computer model for GTCC waste projections.

The earlier WMG, Inc., study analyzed low-level radioactive waste (LLW) streams to identify those waste streams that could be GTCC LLW as a function of concentration averaging scenario. Several types of LLW wastes produced by light water reactors (LWR) were identified as potential GTCC LLW under the most restrictive concentration averaging scenario considered. These wastes included specific activated metal components and certain process wastes in the form of cartridge filters and decontamination resins. The waste streams, their physical characteristics, and that portion of them which could be GTCC LLW was described in the previous study.

The amounts of GTCC LLW generated by LWRs are plant type and fuel cycle dependent. To define LWR operating parameters, a master list of LWRs was initially prepared. This list included start up dates, plant type, reactor model, and the most recent information on plant-specific fuel cycles and refueling outage times. Current LLW management practices at operating plants were then reviewed. These practice reviews were used to estimate the amounts of potential GTCC LLW generated per fuel cycle and routine generation rates. The amounts of routinely generated activated metal components and process waste were then estimated as a function of fuel cycle. Component-specific radionuclide content and distribution was calculated for activated metals components. Empirical data from actual LLW waste streams were used to estimate radionuclide content and distribution for process wastes. The GTCC LLW volumes that could be generated through plant closure were also estimated by LWR plant. These estimates were based on the GTCC LLW generation rates and the number of fuel cycles remaining for each LWR from January 1993 through plant closure. January 1993 is the date when the commercial LLW disposal sites are currently scheduled to close.

Volumes and activities for potential GTCC activated metals generated at decommissioning were also estimated. The methods used were similar to those described in the Nuclear Regulatory Commission (NRC) reports on decommissioning waste. Six typical LWR plants, four pressurized water reactors (PWRs) and two boiling water reactors (BWRs), were selected for detailed analysis. Plant-specific drawings and data were used to model the reactor vessel internals and conditions for each plant. Surveillance capsule reports, which are periodically prepared to determine integrated neutron exposure to the reactor vessel, were requested from the six plants selected for analysis. These reports were obtained from four of the six plants. The information in the surveillance capsule reports was used to normalize the neutron fluxes calculated by the methods customarily used for activation analysis of reactor internals. The normalized fluxes were then used to perform the activation calculations for the vessel internal components present in each plant-specific reactor model. Component-specific radionuclide content and distributions were calculated. Decommissioning GTCC LLW volumes were determined from drawings and data for the six plant-specific reactor models.

analyzed. These values were then used to estimate the GTCC volumes that could be generated from the decommissioning of all LWRs based on 40-year operating cycles.

Section 2 presents the summary and conclusions for this study. The overall approach to the study and the LWR fuel cycle information used is presented in Section 3. Sections 4, 5, and 6 present the study results for routinely generated activated metals, process wastes, and decommissioning activated metal respectively. Section 7 presents the study references.

2. EVALUATION SUMMARY AND CONCLUSIONS

The generation of potential GTCC LLW from LWRs is both plant type and fuel cycle dependent. For this study, all LWRs, both operating and planned for startup, were considered. This included 113 LWRs composed of 36 BWRs and 77 PWRs.

Based on the results of a previous WMG study (Reference 1) three categories of potential GTCC LLW were considered:

- **Routinely Generated Activated Metal**--The nonfuel-bearing components that are activated through exposure to neutron flux and periodically replaced during routine operations
- **Process Wastes**--The wet wastes, in the form of cartridge filters and decontamination project resins, which are generated from cleanup of liquids containing soluble and insoluble radioactive constituents
- **Decommissioning Activated Metals**--The nonfuel-bearing structural components within the reactor vessel that are activated by neutron exposure and removed at decommissioning.

This evaluation confirmed the results of the earlier WMG study, which indicated that GTCC LLW volumes would be much lower than previously estimated. Routinely generated activated metals and process wastes, in the form of cartridge filters, were found to represent a relatively small portion of the LLW routinely generated at LWRs. The estimated generation rates at BWRs ranged from a low of 3.5 ft³ to a high of 6.5 ft³ per plant per fuel cycle. The estimated generation rates at PWRs, which varied with reactor vendor, ranged from a low of 2.4 ft³ to a high of 3.5 ft³ per plant per fuel cycle. A summary of these estimated generation rates organized by waste type and reactor vendor is presented in Table F-1, along with the estimated total volumes of GTCC LLW that could be generated through plant closure. These cumulative volume estimates are based on the generation rates shown and the number of fuel cycles remaining through plant closure assuming 40-year plant life cycles. As shown, under the "no concentration averaging" or highest GTCC LLW volume scenario, the cumulative volumes are relatively small, amounting to about 6000 ft³ or 170 m³.

All indications are that full reactor coolant system (RCS) decontamination projects with fuel assemblies remaining in the vessel will result in GTCC LLW in the form of deep bed resins. Full RCS decontamination projects have never been performed and will not be until the mid-1990s. Initially, these operations will be performed with fuel out at a frequency of about once every 10 years per plant. After these decontamination processes have been fully demonstrated with fuel assemblies removed, plants will consider full RCS decontaminations with fuel assemblies in the vessel. To project RCS decontamination resin volumes, it was assumed that plants with less than 20 years remaining through closure would not perform full RCS decontaminations with fuel assemblies in the vessel. This assumption eliminated 43 plants--17 BWRs, and 26 PWRs--as potential generators of GTCC resin waste. The remaining 70 plants--19 BWRs, and 51 PWRs--were assumed to perform one full RCS decontamination project with fuel assemblies in the vessel before shutdown. The estimated volume of GTCC decontamination resin waste for a typical BWR was 100 ft³, and that for a PWR was 150 ft³. These estimates lead to cumulative volumes of about 9500 ft³ or about 270 m³ of decontamination resin for all LWRs through shutdown.

Table F-1. Summary of routine GTCC LLW

SUMMARY OF ROUTINE GTCC WASTES								
	Number Of Plants	Unit Volume (ft ³)	Units Per Cycle	Percent GTCC	GTCC Units Cycle	GTCC (ft ³) Cycle	Cycles To Shutdown	Total GTCC (ft ³)
ROUTINE ACTIVATED METALS								
1.BWR Operations	36							
a. Control Rod Blades								
- Small Normal	10	0.6	14	10	1	0.6	127	76
- Small Controlled Cell	7	0.6	13/11	25/10	4	2.4	85	204
- Large Normal	12	0.6	19	10	2	1.2	220	264
- Large Controlled Cell	7	0.6	19/15	25/10	6	3.6	131	473
b. LPRM Strings	36	0.1	12	75	9	0.9	564	507
c. IRM/SRM Dry Tubes	36	0.08	2	100	2	0.16	564	90
							Subtotal	1,614
'XX/YY' - Controlled Cell/Normal								
2. West. PWR Operations	55							
a. Thimble Plug Assemblies		0.024	16	10	1.6	0.0384	978	38
b. Thimble Tubes		0.06	2	100	2	0.12	978	117
c. Primary Source Rods		0.007	0.4	100	0.4	0.0028	978	3
							Subtotal	158
3. CE PWR Operations	15							
a. Instrument Strings		0.07	24	50	12	0.84	249	209
b. Primary Sources		0.02	0.4	100	0.4	0.008	249	2
							Subtotal	211
4. B&W PWR Operations	7							
a. In-Core Detectors		0.1	16	50	8	0.8	100	80
b. Primary Sources		0.002	0.4	100	0.4	0.0008	100	<1
								80
							Subtotal	2,063
ROUTINE PROCESS WASTES								
1.BWR Cartridge Filters	36							
a. Control Rod Drive		0.07	30	50	15	1.05	564	592
b. Fuel Pool/Vacuum		0.22	12	25	3	0.66	564	372
							Subtotal	964
2.PWR Cartridge Filters	77							
a. Reactor Coolant System		0.45	8	25	2	0.9	1,327	1,194
b. Seal Water Injection		0.1	16	25	4	0.4	1,327	530
c. Cavity Drain		0.1	2	25	0.5	0.05	1,327	66
d. Spent Fuel Pool		0.85	4	25	1	0.85	1,327	1,128
							Subtotal	2,918
3.B&W Crud Tank Filters	7	0.95	0.5	100	0.5	0.475	100	48
							Subtotal	3,930
							Grand Total	5,993

Decommissioning activated metals represent the largest source of GTCC LLW in terms of volume and activity. For this study, a detailed analysis was performed for six actual plants typical of LWRs. Two BWRs and four PWRs were analyzed in depth to conservatively estimate volumes and to realistically determine activities of potential GTCC components. Table F-2 summarizes the results of this analysis extended to all LWRs in terms of volumes. As shown, the cumulative volume of decommissioning activated metals for all plants considered is about 21,400 ft³ or 610 m³. These values are conservative, since only larger plants of each LWR type were evaluated in detail.

The study results for decommissioning activated metals were somewhat different than those previously reported in the standard NRC reference reports. For BWR components, the results of this study indicate that when calculated fluxes are normalized to surveillance capsule data, the calculated integrated fluxes are about 35% lower than those previously reported, resulting in lower activities. Thus, while the core shroud exceeds Class C limits, it does so by factors of less than 2. These results indicate that further investigations to better define BWR flux levels are warranted.

For PWR components, the results of this evaluation confirm that the core baffle or shroud is well above Class C limits for all PWRs considered and that the thermal shield is not GTCC LLW. Mixed results were found for the core barrel, a major PWR component. For the two Westinghouse PWRs, the core barrel was found to barely exceed Class C limits. Since these plants were considered to be representative of 55 PWRs, further evaluations could eliminate the core baffle as GTCC LLW in this group of PWRs.

For larger LWRs, this study showed that the use of surveillance capsule data to normalize activation fluxes eliminates some of the conservatism in earlier studies. Some of the older, smaller LWRs, with relatively long operating histories, were not analyzed. These older units should be analyzed to determine if the use of surveillance capsule data for plants with the higher integrated fluxes associated with longer operation further reduce activation flux estimates.

The methodology employed for the analysis of decommissioning activated metals relied on NUREG 3474² for materials compositions. These data have not been applicable to other activated metals components for some time. Activated materials heat data, or composites thereof, are typically used for activation analysis performed on routinely generated activated metal components. Similar current materials composition data obtained from actual component heats should be compiled for decommissioning components. These updated materials composition data could be used to eliminate some of the uncertainties in decommissioning component activation analysis and better define potential GTCC LLW volumes.

The results from the reactor models evaluated also clearly demonstrate that plants should be modeled and analyzed individually due to variances in geometry and operating history, which can significantly influence the results.

Table F-2. LWR decommissioning GTCC LLW summary

LWR DECOMMISSIONING GTCC WASTE SUMMARY					
	Number of Units	Core Shroud (ft³)	Core Baffle (ft³)	Core Barrel (ft³)	Total Volume (ft³)
BWR Plants					
1. Models 2/3	9	102	NA	NA	918
2. Model 4	13	102	NA	NA	1,326
3. Model 4/5	2	176	NA	NA	352
4. Model 6	12	176	NA	NA	2,112
Totals	36				4,708
PWR Plants					
1. B&W Reactors	7	NA	52	90	994
2. CE Reactors	15	NA	88	290	5,670
3. West. Moderate	28	NA	45	125	4,760
4. Westinghouse Large	27	NA	71	125	5,292
Totals	77				16,716
Grand Totals	113				21,424

3. TECHNICAL APPROACH

The study objective is to provide quantitative volumes and radioactivity amounts for the GTCC LLW that could be generated by LWRs over the period 1995 through 2025.

To achieve this objective, parameters that describe LWR plant performance and the impact of this performance on GTCC LLW generation were considered. These parameters are discussed below.

3.1 LWR Operations Parameters

At present, there are 36 General Electric BWR plants ranging in size from the 504 MWe Vermont Yankee plant to the 1205 MWe Perry plant. Table F-3 summarizes the overall plant parameters relevant to this study. Table F-4 illustrates similar information for PWR stations broken down by reactor vendor. As shown, there are 7 Babcock & Wilcox plants, 15 Combustion Engineering plants, and 55 Westinghouse plants. Three of the Westinghouse plants have not yet initiated operations. Since there is no indication that they will not be started up on the dates shown, they were included in the study.

The parameters shown on these tables include:

- Reactor Type--For BWRs: 2, 3, 4, or 6; for Westinghouse PWRs: 2, 3, or 4.
- Plant Name, Unit, and MWe.
- Startup Date--Actual (or expected) date of commercial operation.
- Closure Date--Assumed to be 40 years after commercial operation.
- Cycle Months--The latest published information on the length of the plants' fuel cycle.
- Outage Start and Finish Dates--The latest published information on outages in 1990, and those planned for 1991.
- Days Shut Down--The elapsed time between start and finish dates of the 1990-1991 outages.
- Capacity Factors--The latest published information on the plants, values for capacity factors for the periods 1984-1986 and 1987-1989. These factors are the effective full power days (EFPD) generated by the plant divided by the EFPD that would be generated if the plant had run at 100% power for 100% of the time.
- Best Capacity Factor--The highest capacity factor reported for the plant averaged over either the 1984-1986 or the 1987-1989 time period.

These operations parameters provided the overall framework for defining LWR fuel cycles and the basis for GTCC LLW generation rates.

Table F-3. BWR stations

BWR STATIONS													
#	Type	Plant Name	Unit	MWE Net	Startup Date	Closure Date	Cycle Mos	Outage Start	Outage Finish	Days SD	CF 84-86	CF 87-89	Best Factor
1	2	Nine Mile Point	1	610	Dec-69	Nov-2009		03/03/91	04/13/91	41	71.8	28.3	71.8
2	2	Oyster Creek	1	620	Dec-69	Nov-2009	21	02/15/91	06/01/91	41	31.2	52.9	52.9
3	3	Dresden	2	772	Jun-70	May-2010	15	09/01/90	11/15/90	75	58.4	59.3	59.3
4	3	Millstone	1	654	Mar-71	Feb-2011	21	03/30/91	05/10/91	41	81.6	83.8	83.8
5	3	Monticello		536	Jun-71	May-2011	19	04/01/91	05/27/91	56	55.2	75.0	75.0
6	3	Dresden	3	773	Nov-71	Oct-2011	17	03/30/91	06/08/91	70	43.1	65.5	65.5
7	4	Vermont Yankee		504	Nov-72	Oct-2012	16	09/01/90	10/13/90	42	62.0	83.3	83.3
8	3	Pilgrim		670	Dec-72	Nov-2012	20	05/04/91	07/13/91	70	34.7	9.9	34.7
9	3	Quad Cities	1	769	Feb-73	Jan-2013	14	10/27/90	01/06/91	71	66.7	69.4	69.4
10	3	Quad Cities	2	769	Mar-73	Feb-2013	16	09/14/91	11/23/91	70	68.7	71.1	71.1
11	4	Cooper		764	Jul-74	Jun-2014	15	10/06/91	11/30/91	55	42.0	70.9	70.9
12	4	Peach Bottom	2	1051	Jul-74	Jun-2014	18	01/12/91	03/20/91	67	41.6	19.3	41.6
13	4	Peach Bottom	3	1035	Dec-74	Nov-2014	18	08/31/91	11/08/91	69	55.6	5.9	55.6
14	4	Duane Arnold		538	Feb-75	Jan-2015	18	06/28/90	09/04/90	68	54.2	62.4	62.4
15	4	J.A. FitzPatrick		757	Jul-75	Jun-2015	24	04/01/91	04/10/91	9	70.0	68.5	70.0
16	4	Brunswick	2	790	Nov-75	Oct-2015	16	04/15/91	05/07/91	22	43.2	63.9	63.9
17	4	Edwin I. Hatch	1	753	Dec-75	Nov-2015	18	09/18/91	11/24/91	67	58.7	76.7	76.7
18	4	Brunswick	1	790	Mar-77	Feb-2017	14	09/27/90	03/14/91	168	59.8	58.7	59.8
19	4	Edwin I. Hatch	2	766	Sep-79	Aug-2019	18	03/20/91	05/24/91	65	52.7	68.6	68.6
20	4/5	Susquehanna	1	1032	Jun-83	May-2023	18	09/08/90	11/10/90	63	61.3	75.5	75.5
21	6	LaSalle County	1	1036	Jan-84	Dec-2023	18	02/16/91	04/27/91	70	42.4	55.3	55.3
22	6	LaSalle County	2	1036	Oct-84	Sep-2024							
23	6	WNP-2	2	1095	Dec-84	Nov-2024	10	05/22/91	07/10/91	49			
24	4/5	Susquehanna	2	1038	Feb-85	Jan-2025	18	03/09/91	05/10/91	62			
25	6	Grand Gulf		1142	Jul-85	Jun-2025	18	10/01/90	11/14/90	44			
26	6	Limerick	1	1055	Feb-86	Jan-2026		09/07/91	09/28/91	21			
27	6	River Bend		936	Jun-86	May-2026	16	09/15/90	11/14/90	60			
28	6	Hope Creek		1031	Dec-86	Nov-2026	17	12/26/90	02/13/91	49			
29	6	Clinton	1	930	Apr-87	Mar-2027	13	10/14/90	02/05/91	114			
30	6	Perry	1	1205	Nov-87	Oct-2027	18	09/07/90	11/24/90	78			
31	4	Fermi	2	1075	Jan-88	Dec-2027	13	03/16/91	05/29/91	74			
32	6	Nine Mile Point	2	1072	Apr-88	Mar-2028	15	09/05/90	01/15/91	132			
33	6	Limerick	2	1055	Jan-90	Dec-2029	24	03/23/91	06/08/91	77			
Averages				868			17.21			64.37	54.9952	58.2952	65.1

Table F-4. PWR stations

PWR STATIONS												
# Type	Plant Name	Unit	MWE Net	Startup Date	Closure Date	Cycle Mos	Outage Start	Outage Finish	Days SD	CF 84-86	CF 87-89	Best Factor
B&W Plants												
1	Oconee	1	846	Jul-73	Jun-2013	13.9	07/25/91	09/18/91	55	77.2	77.8	77.8
2	Oconee	2	846	Sep-74	Aug-2014	14.2	12/12/91	02/05/92	55	77.8	76.2	77.8
3	Three Mile Island	1	808	Sep-74	Aug-2014	24.0	10/04/91	12/13/91	70	26.1	82.2	82.2
4	Arkansas Nuclear One	1	836	Dec-74	Nov-2014	14.0	10/01/90	12/07/90	67	59.8	54.1	59.8
5	Oconee	3	846	Dec-74	Nov-2014	14.6	02/13/91	03/23/91	38	69.8	74.5	74.5
6	Crystal River	3	821	Mar-77	Feb-2017		10/10/91	11/14/91	35	55.2	56.8	56.8
7	Davis-Besse		874	Jul-78	Jun-2018	15.9	09/01/91	11/09/91	69	26.2	56.9	56.9
Averages									55.5			69.4
CE Plants												
1	Palisades		768	Dec-71	Nov-2011	11.8	09/15/90	02/11/91	149	32.8	45.8	45.8
2	Maine Yankee		830	Dec-72	Nov-2012	18.0	10/05/91	11/30/91	56	77.1	73.7	77.1
3	Fort Calhoun		478	Sep-73	Aug-2013	18.0	09/28/91	01/08/92	102	71.6	71.5	71.6
4	Calvert Cliffs	1	825	May-75	Apr-2015		03/10/91	04/15/91	36	73.8	53.0	73.8
5	Millstone	2	863	Dec-75	Nov-2015	16.0	09/15/90	11/10/90	56	66.7	76.5	76.5
6	St. Lucie	1	839	Dec-76	Nov-2016	18.0	10/18/91	12/20/91	63	78.5	86.6	86.6
7	Calvert Cliffs	2	825	Apr-77	Mar-2017		10/04/91	11/17/91	44	80.8	58.0	80.8
8	Arkansas Nuclear One	2	858	Mar-80	Feb-2020	18.0	02/15/91	04/07/91	51	67.5	71.0	71.0
9	San Onofre	2	1070	Aug-83	Jul-2023	18.8	07/06/91	09/14/91	70	59.6	72.7	72.7
10	St. Lucie	2	839	Aug-83	Jul-2023	18.0	09/01/90	10/23/90	52	81.6	86.1	86.1
11	San Onofre	3	1080	Apr-84	Mar-2024							
12	Waterford	3	1075	Sep-85	Aug-2025	18.0	03/24/91	05/17/91	54			
13	Palo Verde	1	1221	Jan-86	Dec-2025		01/12/91	02/26/91	45			
14	Palo Verde	2	1221	Sep-86	Aug-2026	18.0	10/16/91	12/25/91	70			
15	Palo Verde	3	1221	Jan-88	Dec-2027	18.0	03/17/91	05/26/91	70			
Averages									65.5			74.2
									17.3			

Table F-4. (continued)

PWR STATIONS													
# Type	Plant Name	Unit	MWE Net	Startup Date	Closure Date	Cycle Mos	Outage Start	Outage Finish	Days SD	CF 84-86	CF 87-89	Best Factor	
West. Plants													
1	Yankee		167	Jul-61	Jun-2001	18.0	06/23/90	08/19/90	57	78.2	77.3	78.2	
2	2 Haddam Neck		565	Jan-68	Dec-2007	14.0	10/05/91	11/15/91	41	66.2	57.4	66.2	
3	2 San Onofre	1	436	Jan-68	Dec-2007	16.7	06/30/90	11/30/90	153	31.4	45.8	45.8	
4	2 R. E. Ginna		470	Jul-70	Jun-2010	10.0	03/15/91	04/19/91	35	84.0	84.2	84.2	
5	2 Point Beach	1	485	Dec-70	Nov-2010	10.9	04/05/91	05/17/91	42	78.2	84.2	84.2	
6	2 Robinson	2	665	Mar-71	Feb-2011	15.0	09/09/90	01/10/91	123	55.5	55.3	55.5	
7	2 Point Beach	2	485	Oct-72	Sep-2012	10.7	09/27/91	11/08/91	42	80.6	82.7	82.7	
8	3 Surry	1	781	Dec-72	Nov-2012	14.0	09/06/91	09/20/91	14	64.8	50.6	64.8	
9	2 Turkey Point	3	666	Dec-72	Nov-2012		12/01/90	09/21/91	294	69.7	43.3	69.7	
10	2 Surry	2	781	May-73	Apr-2013	18.5	04/05/91	06/18/91	74	66.5	44.7	66.5	
11	2 Turkey Point	4	666	Sep-73	Aug-2013	18.0	11/15/90	10/26/91	345	54.7	43.8	54.7	
12	2 Prairie Island	1	503	Dec-73	Nov-2013	17.0	06/05/91	07/02/91	27	83.6	84.7	84.7	
13	4 Zion	1	1040	Dec-73	Nov-2013	11.0	08/31/91	11/09/91	70	56.3	63.6	63.6	
14	2 Kewaunee		503	Jun-74	May-2014	10.2	03/08/91	05/05/91	58	80.7	82.9	82.9	
15	4 Indian Point	2	970	Aug-74	Jul-2014	22.0	01/12/91	05/27/91	135	58.2	68.3	68.3	
16	3 Zion	2	1040	Sep-74	Aug-2014					67.4	71.1	71.1	
17	2 Prairie Island	2	500	Dec-74	Nov-2014	17.0	09/12/90	10/10/90	28	81.6	87.6	87.6	
18	4 Donald C. Cook	1	1020	Aug-75	Jul-2015	15.5	10/20/90	01/13/91	85	60.2	66.2	66.2	
19	4 Trojan		1095	May-76	Apr-2016	9.8	03/27/91	06/05/91	70	63.0	54.6	63.0	
20	3 Indian Point	3	965	Aug-76	Jul-2016		05/11/91	05/25/91	14	64.2	65.1	65.1	
21	3 Beaver Valley	1	810	Oct-76	Sep-2016	16.1	04/12/91	06/21/91	70	70.2	65.5	70.2	
22	4 Salem	1	1106	Jun-77	May-2017	12.0	02/09/91	03/30/91	49	63.5	67.3	67.3	
23	3 Joseph M. Farley	1	824	Dec-77	Nov-2017	18.0	03/08/91	05/07/91	60	78.1	84.2	84.2	
24	3 North Anna	1	915	Jun-78	May-2018	18.0	01/04/91	03/19/91	74	66.6	61.9	66.6	
25	4 Donald C. Cook	2	1060	Jul-78	Jun-2018	15.5	06/30/90	09/13/90	75	53.2	48.4	53.2	
26	3 North Anna	2	915	Dec-80	Nov-2020	18.0	09/07/90	11/21/90	75	73.6	81.5	81.5	
27	3 Joseph M. Farley	2	828	Jul-81	Jun-2021	16.0	10/06/90	11/24/90	49	82.8	81.1	82.8	
28	4 Sequoyah	1	1148	Jul-81	Jun-2021	18.0	10/15/91	12/15/91	61	33.5	31.8	33.5	

Table F-4. (continued)

PWR STATIONS												
# Type	Plant Name	Unit	MWE Net	Startup Date	Closure Date	Cycle Mos	Outage Start	Outage Finish	Days SD	CF 84-86	CF 87-89	Best Factor
West. Plants												
29 4	Salem	2	1106	Oct-81	Sep-2021	16.0	11/30/91	02/07/92	69	46.1	68.1	68.1
30 4	McGuire	1	1129	Dec-81	Nov-2021	13.0	09/06/91	11/23/91	78	59.6	72.6	72.6
31 4	Sequoyah	2	1148	Jun-82	May-2022	18.0	09/10/90	11/21/90	72	39.5	32.9	39.5
32 3	Virgil C. Summer		885	Jan-84	Dec-2023	16.3	09/13/91	11/22/91	70	70.0	66.0	70.0
33 4	McGuire	2	1129	Mar-84	Feb-2024	13.0	09/01/90	11/19/90	79			
34 4	Callaway		1125	Apr-85	Mar-2025	18.0	09/21/90	11/16/90	56			
35 4	Diablo Canyon	1	1073	May-85	Apr-2025	18.0	02/10/91	04/15/91	64			
36 4	Catawba	1	1129	Jun-85	May-2025	12.3	03/12/91	06/12/91	92			
37 4	Byron	1	1105	Sep-85	Aug-2025	15.0	09/07/91	11/16/91	70			
38 4	Wolf Creek		1135	Sep-85	Aug-2025	15.0	09/12/91	11/21/91	70			
39 4	Diablo Canyon	2	1087	Mar-86	Feb-2026	18.0	09/15/91	11/17/91	63			
40 4	Millstone	3	1142	Apr-86	Mar-2026	19.0	02/02/91	03/30/91	56			
41 4	Catawba	2	1129	Aug-86	Jul-2026	13.6	10/24/91	12/28/91	65			
42 3	Shearon Harris		860	May-87	Apr-2027	18.0	03/01/91	05/10/91	70			
43 4	Alvin W. Vogtle	1	1079	Jun-87	May-2027	18.0	09/13/91	10/28/91	45			
44 4	Byron	2	1105	Aug-87	Jul-2027	15.0	09/01/90	11/10/90	70			
45 3	Beaver Valley	2	833	Nov-87	Oct-2027	18.0	09/01/90	11/09/90	69			
46 4	Braidwood	1	1120	Jul-88	Jun-2028	16.0	03/16/91	05/25/91	70			
47 4	South Texas Project	1	1250	Aug-88	Jul-2028	18.0	04/01/91					
48 4	Braidwood	2	1120	Oct-88	Sep-2028	15.5	09/14/91	11/23/91	70			
49 4	Alvin W. Vogtle	2	1100	May-89	Apr-2029	16.1	09/14/90	11/03/90	50			
50 4	South Texas Project	2	1250	Jun-89	May-2029	12.0	11/01/90					
51 4	Seabrook		1150	Jul-90	Jun-2030	12.0	07/27/91	10/02/91	67			
52 4	Comanche Peak	1	1150	Aug-90	Jul-2030		04/13/91	05/24/91	41			
53 4	Watts Bar	1	1177	Jan-93	Dec-2032							
54 4	Comanche Peak	2	1150	Jan-93	Dec-2032							
55 4	Watts Bar	2	1177	Jan-95	Dec-2034							
Averages						14.9			72.1			64.5

3.2 Study Parameters

For BWRs, the waste types that could be GTCC under the no concentration averaging case include:

- Activated metals
 - Control rod blades
 - Local power range monitors
 - IRM/SRM dry tubes
- Process wastes
 - Cartridge filters
 - Full RCS decontamination resins
- Decommissioning activated metals
 - Core shroud
 - Upper fuel guide.

For PWRs, these waste types included:

- Activated metals
 - Thimble plug assemblies
 - Incore detectors and instrument strings
 - Primary sources
- Process wastes
 - Cartridge filters
 - Full RCS decontamination resins
- Decommissioning activated metals
 - Core baffle (shroud)
 - Core barrel.

Two of these wastes types, activated metals and cartridge filters, are routinely generated. Decontamination resins are generated intermittently. Decommissioning metals are generated at plant closure. The integration of LWR operations parameters and these GTCC LLW generation mechanisms are discussed below.

3.2.1 Routinely Generated Potential GTCC LLW

Routine GTCC LLW type generation is largely dependent on individual plant operation. The information shown on Tables F-3 and F-4 was expanded to define the following parameters relevant to potential GTCC LLW generation.

- Fuel Cycles--To determine this parameter for BWRs, the latest information in Table F-3 for 33 of the 36 plants was compiled and reviewed. The three Browns Ferry plants were excluded since they have not operated for six years.

This review indicated that the average BWR fuel cycle was 17.4 months, with 64 days of downtime between cycles for refueling. A 20-month cycle, which includes three months for refueling, was assumed.

As shown in Table F-4, fuel cycle lengths varied among PWR reactor vendors. For PWRs, two different cycles were used: a 19-month cycle for the Babcock & Wilcox and Westinghouse plants and a 20-month cycle for the Combustion Engineering plants. Both of these cycles include three months for refueling.

- **Exposure Per Cycle**--For activated metals, the EFPD per cycle determines a components exposure and its radioactivity. The average of the best capacity factors reported was used in conjunction with a three-month downtime between fuel cycles to define an average activation factor. This factor was used with the average power level for all plants to define the average EFPD of activated metal exposure per fuel cycle.

For BWRs, the average cycle exposure is 390 EFPD, which corresponds to a 17-month operating cycle at 76% capacity. Two different cycle exposures were used for PWRs: 400 EFPD for Babcock & Wilcox and Westinghouse plants, which corresponds to a 16-month operating cycle at 83% capacity; and 430 EFPD for Combustion Engineering plants, which corresponds to a 17-month operating cycle at 83% capacity.

- **Cycles to Closure**--Most plants do not have GTCC LLW under current criteria. After January 1, 1993, compact criteria can lead to GTCC LLW. The volumes that could be generated between this date and closure will be operating cycle dependent. The number of plant-specific fuel cycles to closure was determined for each plant by dividing the time between January 1, 1993, and the closure date by the average fuel cycle lengths. 20 months was used for BWRs. Two different cycle lengths were used for PWRs: 19 months for Westinghouse and Babcock & Wilcox plants and 20 months for Combustion Engineering plants.

These parameters were used to estimate routinely generated GTCC volumes and activities as described in Sections 4 and 5.

3.2.2 Decommissioning Potential GTCC LLW

Decommissioning GTCC LLW consists exclusively of activated metals, and generation is dependent on the integrated flux to which vessel internals are exposed during the plant's operating lifetime. The variable parameters that define integrated exposure are the plant's power level and capacity factor. The fixed parameter used for this study is a 40-year operating lifetime.

For BWRs, the average for the best capacity factors, 65%, was used in conjunction with the 40-year lifetime to estimate the radionuclide activities at the time of BWR decommissioning. For PWRs, the average for the best capacity factors varied among vendors. As shown in Table F-4, capacity was about 70% for Babcock & Wilcox, 74% for Combustion Engineering, and 68% for Westinghouse. For this study, a weighted overall average factor of 70% was used in conjunction with the 40-year lifetime to estimate the radionuclide activities at the time of PWR decommissioning. These parameters were used to estimate decommissioning GTCC volumes and activities as described in Section 6.

4. ACTIVATED METALS GENERATION AND ACTIVITY

Activated metals consist of the nonfuel-bearing components that are activated through exposure to neutron flux within light water, research, and test reactors. Two sources were considered: those components that are periodically replaced during normal operations, and those structural components in the reactor vessel that are removed at decommissioning.

The generation of activated metals depends on the plant's type, size, fuel cycle length, and average exposure during each fuel cycle. After removal from the core, these components are usually allowed to accumulate in the spent fuel pool until sufficient quantities are available for shipment. While the quantities that could be GTCC are a fraction of the volume discharged, it was assumed that current practices would continue in the post-1993 time frame. Thus, activities of components were estimated after two cycles of decay in the spent fuel pool. This decay does not affect 10 CFR Part 61³ classification status. It merely allows for the decay of the short-lived activation product radionuclides Cr-51, Mn-54, Fe-55, and Co-60. Since Fe-55 and Co-60 are the dominant radionuclides in activated metals, their decay time significantly reduces total activity.

This section discusses routinely generated components. It presents the estimated GTCC LLW volumes and curie content for activated metal components from BWRs and PWRs based on the operating cycles discussed in Section 3. Section 6 discusses decommissioning components.

4.1 BWR Activated Metals

Current data indicate that the typical BWR operates on a 20-month cycle, with three months allowed for refueling. During the 17 months of power operation, the average plant operates at 76% of full power.

These average conditions were compared to the actual operating cycle histories of ten representative BWRs, which operated for between 4 and 12 fuel cycles. Average power operating cycle days ranged from 380 to 560 days. Approximately 480 days (16 months) was the average for all plants. Since this average included unplanned outages, the 17-month typical fuel cycle assumption is reasonable. The average integrated exposure during these cycles was 360 EFPD, which represents an activation factor of 75% (360 EFPD/480 days). This compares very favorably to the overall average of 76% for all BWRs determined in Section 3. For conservative purposes, it was assumed that components received 390 EFPD exposure per fuel cycle, which is based on a 17-month cycle at 76% of rated capacity. It was also assumed that these components would not be available for shipment until 40 months after discharge.

Component activities were determined using the methods employed by WMG to characterize actual BWR components for disposal. The materials compositions used are based on composition data from manufacturer heat data of material used to manufacture the components considered.

BWR activated metal components that can exceed Class C limits are discussed below.

4.1.1 Control Rod Blades

Each core contains from 137 to 185 control rod blades (CRBs), which are distributed radially throughout the core. The CRB volume was assumed to be 0.6 ft³.

Control rod operating patterns vary by station. Most plants operate in a conventional control rod pattern, where alternating banks of deep and shallow CRBs are slowly withdrawn from the core. Some plants also run in what is referred to as the controlled cell mode. In this mode, CRB movement is restricted to a fixed group of 25 in a 5 × 5 array (137 CRB cores) to 37 CRBs in a 5 × 5 array surrounded by a 3 × 3 array (185 CRB cores). CRBs in the controlled cell mode are fully inserted while the rest of the CRBs are completely withdrawn.

Routine CRB discharges are exposure dependent. For BWRs not running in the controlled cell mode, historical data indicate that CRBs are not discharged until after the sixth or seventh fuel cycle. For BWRs running in the controlled cell mode, half of the 25 to 37 CRBs that operated in this mode are discharged each fuel cycle.

CRB generation rates depend on the core size (137 vs 185 CRBs) and the mode of CRB operation. Consistent with the earlier study findings, 10% of normal CRBs and 25% of controlled cell CRBs were assumed to exceed Class C limits.

The 36 BWRs considered were divided into four categories, based on plant size and mode of CRB operation. Table F-5 summarizes the CRB generation rates for each category and presents the estimated GTCC CRB waste volumes per plant per fuel cycle.

For plants run in the conventional mode of operation, curie estimates were based on blades with six fuel cycles of operation, or 2340 EFPD, of exposure. These CRBs contained 2570 Ci and were 1.4 times greater than Class C limits.

For plants run in the controlled cell mode, estimates were based on two cycles, or 780 EFPD, of exposure. These CRBs contained 3420 Ci and were 1.3 times greater than Class C limits.

The above curie estimates exclude the four upper stellite bearings (rollers) on a CRB that become highly activated. Stellite bearing volumes range from 0.002 to 0.003 ft³ per CRB, dependent on the removal method. The activity of each set of four bearings ranges from 300 to 800 Ci. Alone, stellite bearings are 3 to 5 times greater than Class C limits. Most of the plants that have stellite bearings on their CRBs are gradually replacing them with nonstellite bearing CRBs. This replacement should be completed over the next five years, so stellites should not be a concern for the GTCC LLW project.

The above generation rates and curie contents assume that CRBs which are GTCC LLW will remain intact. This is not consistent with current practice. The CRB velocity limiter sections are almost always removed and, in many cases, the cruciform sections are sectioned prior to shipment. Where plants encounter GTCC CRBs, customary practices would be employed to reduce GTCC LLW volumes. This would, as a minimum, entail removal of the velocity limiter, reducing CRB metal volume from 0.6 to 0.53 ft³. If CRBs had to be processed for shipment, those axial sections that met Class C limits would also be segregated from the GTCC LLW portion of the CRB. For those CRBs that run in the normal mode of operation, the 0.53 ft³ of the cruciform could be reduced by half, to about 0.26 ft³. For those CRBs that run in the controlled cell mode, the cruciform volume could be reduced by 25%, to about 0.40 ft³.

Table F-5. Control rod blade generation

CONTROL ROD BLADE GENERATION				
Plant Size	Moderate	Moderate	Large	Large
Mode of Operation	Normal CRB	Controlled Cell CRB	Normal CRB	Controlled Cell CRB
1. Number of Plants	10	7	12	7
2. Number of CRB's	137	137	185	185
3. Number of CC CRB's	0	25	0	37
4. Normal CRB / Cycle	14	11	19	15
5. Percent Normal GTCC	10	10	10	10
6. CC CRB / Cycle	NA	13	NA	19
7. Percent CC GTCC	NA	25	NA	25
8. Total Cubic Feet GTCC/Cycle	0.6	2.4	1.2	3.6

4.1.2 Local Power Range Monitor Strings

Each core contains 31 local power range monitor (LPRM) strings. The metal volume of an intact LPRM string is 0.1 ft^3 and the weight is about 50 lb. When sectioned, the 13.5 ft hot end has a waste volume of 0.017 ft^3 and weighs 8.5 lb. The cold end has a waste volume of 0.083 ft^3 and weighs 41 lb.

LPRM strings are routinely discharged based on fission chamber depletion or when strings malfunction. Historic data indicate that 12 LPRM strings are discharged each fuel cycle. 75% of the LPRMs discharged were assumed to be GTCC LLW, for a rate of 0.9 ft^3 per plant per fuel cycle.

To estimate typical activities, it was assumed that the LPRMs received three cycles, or 1,170 EFPD, of exposure. These LPRMs contained 930 Ci and were 1.6 times greater than Class C limits.

Over 99% of the activity in the LPRM strings is contained in the hot end. Removal and shipment of those cold end sections that readily meet Class C limits would substantially reduce generation rates. For LPRM strings, the generation rate would be reduced from 0.9 ft^3 to about 0.15 ft^3 per plant per cycle. The hot end section would still contain about 930 Ci. Having essentially the same activity in the reduced volume would result in the hot end exceeding Class C limits by a factor of 9.4.

4.1.3 IRM/SRM Dry Tubes

Each core contains four source range monitor (SRM) dry tubes and eight intermediate range monitor (IRM) dry tubes. The metal volume of an intact dry tube is 0.08 ft^3 , and the weight is about 39 lb. When sectioned, the 13.5 ft hot end has a waste volume of 0.001 ft^3 and weighs 5 lb. The cold end has a waste volume of 0.007 ft^3 and weighs 34 lb.

Dry tubes are not routinely discharged. They are replaced only when damaged or embrittled. Historic data indicate that all 12 dry tubes are replaced at intervals of five to six fuel cycles. An average rate of 2 units or 0.16 ft^3 per plant per fuel cycle was assumed.

To estimate typical activities, it was assumed that the dry tubes received six cycles, or 2340 EFPD, of exposure. These dry tubes contained 1690 Ci and were 4.6 times greater than Class C limits.

Over 99% of the activity in the dry tubes is contained in the hot end. The removal and shipment of those cold end sections that readily meet Class C limits would substantially reduce generation rates. For dry tubes, the generation rate would be reduced from 0.16 ft^3 to about 0.02 ft^3 per plant per cycle. The hot end section would still contain about 1,630 Ci. Having essentially the same activity in the reduced volume would result in the hot end exceeding Class C limits by a factor of 37.

4.1.4 Radionuclide Content and Distribution

Table F-6 presents the estimated activities of the major radionuclides for these components at the time of shipment, and their status relative to Class C limits. The activities are based on 40 months of decay after discharge. For LPRMs and IRM/SRM dry tubes, the classification status is shown for the entire component as well as for the hot end by itself.

Table F-6. BWR hardware components

BWR HARDWARE COMPONENTS				
Activity Estimates				
1. Component Type	Control Rod Blade	Control Rod Blade (CC)	LPRM String	IRM/SRM Dry Tube
2. Volume (ft³)	0.6	0.6	0.1	0.08
3. Weight (lbs)	205	205	50	39
4. Number of Cycles	6	2	3	6
5. Exposure – EFPD (390 EFPD/Cycle)	2,340	780	1,170	2,340
6. Part 61 Status				
– Table 1 Fraction	1.36	1.28	1.57	4.61
– Table 2 Fraction	0.71	0.68	1.37	4.01
– Hot End Table 1	–	–	9.24	36.88
– Hot End Table 2	–	–	8.06	32.08
7. Activity (Curies)				
H-3	2.20E+01	2.30E+01	NA	NA
C-14	1.20E-01	1.10E-01	3.40E-02	8.10E-02
Mn-54	2.00E+01	4.50E+01	8.80E+01	1.20E+01
Fe-55	1.21E+03	1.78E+03	5.30E+02	9.05E+02
Co-60	1.23E+03	1.49E+03	3.60E+02	7.15E+02
Ni-59	5.30E-01	5.50E-01	1.80E-01	3.90E-01
Ni-63	8.50E+01	8.20E+01	2.70E+01	6.40E+01
Nb-94	3.75E-03	3.50E-03	6.24E-04	1.50E-03
TRU	NA	NA	3.00E-05	NA
Total Curies	2.57E+03	3.42E+03	9.30E+02	1.69E+03

4.2 PWR Activated Metals

As discussed in Section 3, current data indicate that typical PWRs operate on 19- to 20-month cycles, with three months allowed for refueling. The 20-month cycle is representative of the Combustion Engineering plants. The average plant operates at 83% of full power during the 16 to 17 months of operation.

These average conditions were compared to actual operating cycle histories for several PWRs from each manufacturer and were found to be reasonably consistent.

For PWRs, the activities of activated components vary with the manufacturer, and these are discussed separately below. It was assumed that Combustion Engineering plant components received 430 EFPD exposure per fuel cycle, and other PWRs received 400 EFPD exposure per fuel cycle. Like BWRs, activities of components were estimated after two cycles of decay in the spent fuel pool. It was assumed that Westinghouse and Babcock & Wilcox plants had 38 months decay time, and that Combustion Engineering plants had 40 months decay time.

Component activities were determined using the methods employed by WMG to characterize actual PWR components for disposal. The materials compositions used are based on actual data furnished by clients for disposal projects.

The basis for projecting volumes for potential GTCC LLW from routine PWR operations is presented below.

4.2.1 Thimble Plug Assemblies (Westinghouse)

For Westinghouse plants that have discharged thimble plug assemblies in the last five years, the average discharge rate is 16 units per plant per fuel cycle. Since only a small portion of these are GTCC, the generation rate is 1.6 units, or 0.04 ft³ per plant per fuel cycle.

To estimate activities, it was assumed that the units received five fuel cycles of exposure (2000 EFPD). These units contained 150 Ci and were 1.3 times greater than Class C limits.

4.2.2 Incore Detectors (Babcock & Wilcox)

Babcock & Wilcox plants use rhodium emitters with inconel lead wires and stainless steel sheaths. The entire detector has a metal volume of about 0.1 ft³. The 12 to 14 ft hot end section has a metal volume of about 0.013 ft³.

For those plants that have disposed of incore detectors, the average rate has been 16 detectors per plant per fuel cycle. Since 50% are GTCC, the generation rate is 8 detectors, or 0.8 ft³ per plant per fuel cycle.

To estimate activities, it was assumed that the detectors received three fuel cycles, or 1200 EFPD, of exposure. These detectors contained 360 Ci and were 2 times greater than Class C limits.

Over 99% of the activity is contained in the hot end section. The removal and shipment of those cold end sections that readily meet Class C limits would substantially reduce generation rates. For these incore detectors, the generation rate would be reduced from 0.8 ft³ to about 0.1 ft³ per

plant per cycle. The hot end section would still contain about 360 Ci. Having essentially the same activity in the reduced volume would result in the hot end exceeding Class C limits by a factor of 16.

4.2.3 Instrument Strings (Combustion Engineering)

Combustion Engineering plants use between 28 and 61 incore instrument assemblies. These instrument strings use rhodium emitters attached to inconel lead wires. The sheathing material is either inconel or stainless steel; the inconel-sheathed instrument strings are the worst case due to high nickel content. Both types have a total length of 35 to 116 feet and weigh between 7 and 34 lb. The typical volume is 0.07 ft³. The 12 to 14 ft hot end section has a metal volume of about 0.008 ft³.

For the past five years, these instrument strings have been changed out at an average rate of 24 units per plant per fuel cycle. Since 50% of these are GTCC, the generation rate is 12 instrument strings, or 0.84 ft³ per plant per fuel cycle.

To estimate activities, it was assumed that the instrument strings received three fuel cycles, or 1290 EFPD, of exposure. These instrument strings contained 385 Ci and were 4 times greater than Class C limits.

Over 99% of the activity is contained in the hot end. The removal and shipment of those cold end sections that readily meet Class C limits would substantially reduce generation rates. For these instrument strings, the generation rate would be reduced from 0.84 ft³ to about 0.1 ft³ per plant per cycle. The hot end section would still contain about 385 Ci. Having essentially the same activity in the reduced volume would result in the hot end exceeding Class C limits by a factor of 35.

4.2.4 Incore Detectors and Thimble Tubes (Westinghouse)

Each Westinghouse plant uses fixed and movable incore detectors, which contain fission chambers housed in a stainless steel sheath. These are inserted into about 50 thimble tubes located throughout the core. The thimble tube has a volume of 0.06 ft³, and the hot end section has a volume of about 0.007 ft³.

The thimble tubes that hold the detectors are changed out infrequently. For a typical Westinghouse plant, 14 thimble tubes may be replaced every seven cycles. Thus, the rate of thimble tube replacement is about 2 units per plant per fuel cycle. Since all of these are GTCC, the generation rate is 2 units, or 0.12 ft³ per plant per fuel cycle.

To estimate activities, it was assumed that the thimble tubes received seven fuel cycles, or 2800 EFPD, of exposure. These thimble tubes contained 1,060 Ci and were 4.3 times greater than Class C limits.

Most of the activity in the thimble tubes is contained in the hot end section. The removal and shipment of those cold end sections that readily meet Class C limits would substantially reduce generation rates. For thimble tubes, the generation rate would be reduced from 0.12 ft³ to about 0.015 ft³ per plant per cycle. The hot end section would still contain about 1,060 Ci. Having essentially the same activity in the reduced volume would result in the hot end exceeding Class C limits by a factor of 37.

4.2.5 Primary Sources

Primary sources greatly exceed Class C limits, and current practices are to accumulate them in the spent fuel pool. When DOE provides for disposal of GTCC LLW, the sources that have accumulated will be shipped with other GTCC LLW.

Babcock & Wilcox plants use two primary neutron sources that are stainless steel-clad americium beryllium. The waste volume of each unit is 0.02 ft³.

Combustion Engineering plants also use two primary neutron sources. The source assemblies are 99 to 117 in. long and weigh between 4.5 and 10.9 lb. The source assemblies are 316 stainless steel with inconel springs. The waste volume is 0.02 ft³ per unit. Each source assembly contains one antimony-beryllium and two plutonium-beryllium sources.

Westinghouse plants use two primary source rods in assemblies that also contain burnable poison rods. The primary source assemblies have a total weight of 19 to 52 lb. The primary source rods themselves only weigh 2.7 to 3.6 lb., and are 110 to 152 in. long. The transuranic source materials are either californium, or plutonium-beryllium. It was assumed that the source rods are separated from the assembly and that the two rods have a waste volume of 0.007 ft³.

Sources are typically expended over five fuel cycles. Since all units are GTCC, the generation rate is 0.4 units per plant per fuel cycle. GTCC volumes by plant type per fuel cycle are as follows:

<u>Reactor Vendor</u>	<u>GTCC Ft³ Per Cycle</u>
Babcock & Wilcox	0.0008
Combustion Engineering	0.008
Westinghouse	0.0028

Source strengths vary by vendor. Typical activities are 50 (Pu-Be) Ci for Westinghouse plants, 340 (Pu-Be) Ci for Combustion plants, and 25 (Am-Be) Ci for Babcock & Wilcox plants.

4.2.6 Radionuclide Content and Distribution

Table F-7 presents the estimated activities of the major radionuclides for these components at the time of shipment, and their status relative to Class C limits. The activities shown include two fuel cycles of decay after discharge. For incore detectors, instrument strings and thimble tubes, the classification status is shown for the entire component, as well as for the hot end by itself.

4.3 Activated Metals Summary

Table F-8 summarizes the generation rates for BWR and PWR activated metals components. Two sets of values are presented: a maximum set that addresses intact components, and a minimum set that assumes that plants will separate individual components to isolate those portions that are GTCC LLW from those that could be routinely disposed of with other metals.

This table also shows the projected cumulative volumes of GTCC LLW generated from January 1993 through plant closure. These cumulative values are the generation rates per cycle by reactor vendor times the number of fuel cycles remaining before closure for each plant considered. As shown, the impact of component separation practices can be significant. It reduces total activated metal volumes from about 2,060 ft³ or 58 m³ to about 740 ft³ or 21 m³.

Table F-7. PWR hardware components

PWR HARDWARE COMPONENTS							
Activity Estimates							
PWR Vendor	West.	West.	West.	CE	CE	B&W	B&W
1. Component Type	Thimble Plugs	Thimble Tubes	Primary Source Rods	Instrument Strings	Primary Sources	In-Core Detectors	Primary Sources
2. Volume (ft³)	0.024	0.06	0.007	0.07	0.02	0.1	0.002
3. Weight (lbs)	12	30	3.5	34	10	50	1
4. Number of Cycles	5	7	5	3	5	3	5
5. Exposure - EFPD	2,000	2,800	2,000	1,290	2,150	1,200	2,000
6. Part 61 Status							
- Table 1 Fraction	1.04	3.33	300,000	3.25	750,000	0.822	550,000
- Table 2 Fraction	1.28	4.29	NEG	4.05	NEG	2.059	NEG
7. Activity (Curies)							
C-14	7.80E-03	5.70E-02	NA	1.83E-02	NA	1.56E-02	NA
Mn-54	1.65E+00	8.40E+00	NA	2.08E-01	NA	1.89E-01	NA
Fe-55	8.85E+01	4.90E+02	NA	9.52E+00	NA	8.41E+00	NA
Co-60	5.45E+01	5.10E+02	NA	3.35E+02	NA	3.11E+02	NA
Ni-59	4.40E-02	4.20E-01	NA	3.26E-01	NA	3.33E-01	NA
Ni-63	5.30E+00	5.10E+01	NA	3.99E+01	NA	4.08E+01	NA
Nb-94	6.30E-05	5.80E-05	NA	1.34E-04	NA	1.24E-04	NA
TRU	NA	NA	50(Pu-Ba)	NA	340(Pu-Ba)	NA	25(Am-Ba)
Total Curies	1.50E+02	1.06E+03	5.00E+01	3.85E+02	3.40E+02	3.61E+02	2.50E+01

Table F-8. Summary of activated metals--GTCC LLW

SUMMARY OF ACTIVATED METALS GTCC WASTES								
	Number Of Plants	Unit Volume (ft ³)	Units Per Cycle	Percent GTCC	GTCC Units Cycle	GTCC (ft ³) Cycle	Cycles To Shutdown	Total GTCC (ft ³)
MAXIMUM ESTIMATES								
1. BWR Operations	36							
a. Control Rod Blades								
- Small Normal	10	0.6	14	10	1	0.6	127	76
- Small Controlled Cell	7	0.6	13/11	25/10	4	2.4	85	204
- Large Normal	12	0.6	19	10	2	1.2	220	264
	7	0.6	19/15	25/10	6	3.6	131	472
b. LPRM Strings	36	0.1	12	75	9	0.9	564	508
c. IRM/SRM Dry Tubes	36	0.08	2	100	2	0.16	564	90
							Subtotal	1,614
'XX/YY' - Controlled Cell/Normal								
2. West. PWR Operations	55							
a. Thimble Plug Assemblies		0.024	16	10	1.6	0.0384	978	38
b. Thimble Tubes		0.06	2	100	2	0.12	978	117
c. Primary Source Rods		0.007	0.4	100	0.4	0.0028	978	3
							Subtotal	158
3. CE PWR Operations	15							
a. Instrument Strings		0.07	24	50	12	0.84	249	209
b. Primary Sources		0.02	0.4	100	0.4	0.008	249	2
							Subtotal	211
4. B&W PWR Operations	7							
a. In-Core Detectors		0.1	16	50	8	0.8	100	80
b. Primary Sources		0.002	0.4	100	0.4	0.0008	100	<1
							Subtotal	80
							Maximum Totals	2,063
MINIMUM ESTIMATES								
1. BWR Operations	36							
a. Control Rod Blades								
- Small Normal	10	0.26	14	10	1	0.26	127	33
- Small Controlled Cell	7	0.37	13/11	25/10	4	1.48	85	126
- Large Normal	12	0.26	19	10	2	0.52	220	114
- Large Controlled Cell	7	0.37	19/15	25/10	6	2.22	131	290
b. LPRM Strings	36	0.017	12	75	9	0.153	564	86
c. IRM/SRM Dry Tubes	36	0.012	2	100	2	0.024	564	14
							Subtotal	663
2. Westinghouse PWR Operations	55							
a. Thimble Plug Assemblies		0.024	16	10	1.6	0.0384	978	38
b. Thimble Tubes		0.007	2	100	2	0.014	978	14
c. Primary Source Rods		0.007	0.4	100	0.4	0.0028	978	3
							Subtotal	55
3. CE PWR Operations	15							
a. Instrument Strings		0.008	24	50	12	0.096	138	13
b. Primary Sources		0.02	0.4	100	0.4	0.008	138	1
							Subtotal	14
4. B&W PWR Operations	7							
a. In-Core Detectors		0.013	16	50	8	0.104	100	10
b. Primary Sources		0.002	0.4	100	0.4	0.0008	100	<1
							Subtotal	10
							Minimum Totals	742

5. PROCESS WASTE GENERATION AND ACTIVITY

Process wastes consist of the wet wastes that are generated from the cleanup of liquids containing soluble and insoluble radioactive constituents. Potential GTCC LLW forms consist of cartridge filters and the cation bead resin produced during full reactor coolant system (RCS) decontamination. Filters are generated routinely, while decontamination resins are generated infrequently. These wastes are usually packaged and shipped shortly after their generation.

The activities of these waste types can best be estimated by reviewing empirical data for the waste types themselves. These empirical data consist of typical radionuclide content and distribution for the waste type and radiation levels. Decontamination of a complete RCS has not yet been performed, and actual data for these decontamination resins were not available. Data for smaller decontamination projects at both BWRs and PWRs were used in conjunction with several studies related to full RCS decontaminations.

5.1 Cartridge Filters

These process wastes are generated during cleanup of contaminated liquids. They represent relatively small unit waste volumes, which typically range from less than 0.1 ft³ to 1 ft³. In certain applications, when cartridge filters are changed out on pressure differential, filter radiation levels can range from 50 to 200 rem/hr. At these radiation levels, some filters are potential GTCC LLW. Changes in plant practices to remove filters from service at preset radiation levels would essentially eliminate all filters, except BWR control rod driveline (CRD) strainers, as GTCC LLW. Table F-9 illustrates the characteristics of filter cartridges that are potential GTCC LLW. These filters are discussed below.

5.1.1 BWR Operations

BWRs routinely generate control rod drive strainers and cartridge filters that are potential GTCC LLW.

- Control Rod Drive Strainers--Each control rod drive contains three metallic strainers. The two strainers at the spud end of the drive can exceed Class C limits. These strainers have volumes of 0.06 ft³ and 0.006 ft³ respectively. About 50% of the strainers generated from this source exceed Class C limits with no concentration averaging.

Control rod drive strainers are replaced when control rod drives are refurbished or replaced. About 15% of the control rod drives in a plant are refurbished during each refueling outage. Drives are replaced infrequently. For projection purposes, it was assumed all control rod drives would be replaced over 6 fuel cycles (ten years). This corresponds to a rate of about 1 ft³ per plant per cycle.

Strainer activities were estimated from typical direct sample data on strainer crud and strainer radiation levels from a representative plant. Typically, these filters can 150 rem/hr on contact, contain several curies, and are 1.4 times greater than Class C limits.

- Fuel Pool/Vacuum Filters--The typical filter is 6 in. in diameter and 30 to 36 in. long. This corresponds to a waste volume of about 0.22 ft³.

Table F-9. Cartridge filter summary

CARTRIDGE FILTER SUMMARY				
	Diameter (in.)	Length (in.)	Waste Volume (ft ³)	Weight (lbs)
BWR FILTERS				
a. Control Rod Drive (Inner)	2.1	3	0.006	0.5
b. Control Rod Drive (Outer)	3.75	9	0.06	2.5
c. Spent Fuel Pool / Vacuum	6	30	0.22	6
PWR FILTERS				
a. Reactor Coolant System	6.25	7.5	0.45	9
b. Seal Water Injection	2.75	20	0.1	2.5
c. Reactor Cavity Drain	2	30	0.1	4
d. Spent Fuel Pool	10	16	0.85	20
e. Crud Tank (B&W only)	10	17	0.95	40

These filters are usually generated during refueling outages and arise from vacuuming the spent fuel pool and the reactor cavity. During a typical outage, a dozen filters can be generated. About 25% of these, or three units (0.7 ft³) per plant per fuel cycle, can be GTCC without concentration averaging.

Data were reviewed for two different plants where several particularly hot filters were seen. The worst case sample data set in terms of 10 CFR Part 61 classification was used to estimate activities for this filter type. Typically, these filters register over 100 rem/hr on contact, contain about 4 Ci, and are about 1.5 times greater than Class C limits.

5.1.2 PWR Operations

PWRs generate a wide range of filter types, some of which can be GTCC. Generation rates are dependent on filter type and plant practices. The rates considered for this study are typical.

- **Reactor Coolant Filters**--These are the highest activity filters generated by a PWR. The typical filter is 6.25 in. in diameter and 17.5 in. long. This corresponds to a volume of about 0.45 ft³. These filters are usually generated at an average rate of about 8 units per fuel cycle. About 25% of these, or two units per fuel cycle, can be GTCC without concentration averaging. This corresponds to a generation rate of 0.9 ft³ per plant per fuel cycle.

Data from a single plant with filter radiation levels in the 200 rem/hr range were used to estimate activity. At these radiation levels, a typical filter contains about 4 Ci and is about 4.3 times greater than Class C limits.

- **Seal Water Injection Filters**--These are the second highest activity filters generated by a PWR. The typical filter is 20 in. in diameter and 2.75 in. long. This corresponds to a volume of about 0.1 ft³. These filters are usually generated at a rate of about 16 units per fuel cycle. About 25% of these, or two units per fuel cycle, can be GTCC without concentration averaging. This corresponds to a generation rate of 0.4 ft³ per plant per fuel cycle.

Data from a single plant with filter radiation levels in the 50 rem/hr range were used to estimate activity. At these radiation levels, a typical filter contains about 1 Ci and is about 2.9 times greater than Class C limits.

- **Cavity Drain Filters**--These are the usually relatively low activity filters, except in those infrequent cases when the cavity is decontaminated. The typical filter is 2 in. in diameter and 30 in. long. This corresponds to a volume of about 0.1 ft³. These filters are generated at a rate of about 2 units per fuel cycle. About 25% of these, or 0.5 units per fuel cycle, can be GTCC without concentration averaging. This corresponds to a generation rate of 0.05 ft³ per plant per fuel cycle.

Data from a single plant with filter radiation levels in the 50 rem/hr range were used to estimate activity. At these radiation levels, a typical filter contains about 1 Ci and is about 2.3 times greater than Class C limits.

- **Fuel Pool Filters**--The typical filter is 10 in. in diameter and 16 in. long. This corresponds to a volume of about 0.85 ft³. These filters are usually generated during

refueling outages, and arise from vacuuming the spent fuel pool and reactor cavity. During a typical outage, four such filters can be generated. About 25% of these, or 1 unit per fuel cycle, can be GTCC without concentration averaging. This corresponds to a generation rate of 0.85 ft³ per plant per fuel cycle.

Data from several plants were reviewed with filter radiation levels in the 100 rem/hr range. At these radiation levels, a typical filter contains about 3 Ci and is about 1.6 times greater than Class C limits.

- **Crud Tank Filter**--These are the highest activity filters in all PWRs and are generated only at Babcock & Wilcox plants. They collect the media backflushed from etched disk filters. The typical collection filter is 10 in. in diameter and 17 in. long. This corresponds to a volume of about 0.95 ft³. These filters are usually generated at a rate of about 1 unit every two fuel cycles. Since the filters always exceed Class C limits without concentration averaging, the generation rate is 0.48 ft³ per plant per fuel cycle.

Actual data for these filters were not readily available.

5.1.3 Radionuclide Content and Distribution

Cartridge filters represent very low volume sources of potential GTCC LLW and exceed Class C limits because the activity is distributed over a relatively small volume. Table F-10 presents a summary of potential GTCC cartridge filters. As shown, radiation levels are relatively high and, in all cases, Class C limits are exceeded due to transuranic concentrations.

5.2 Full RCS Decontamination Resins

Two decontamination processes have been employed to date at BWRs, which are referred to as the LOMI and CANDECON processes. Although the primary systems differ, the same two decontamination processes have also been employed at PWRs. A third process, referred to as CANDEREM, is also being evaluated for the full RCS system decontamination at PWRs. All these processes deposit the crud and scale removed from the reactor internal surfaces on deep bed resins.

Each of these decontamination processes involves several steps and uses different types of resin including cation, mixed, and anion beds. The chelation step generates the cation resins, which can contain up to 75% of the total activity removed during the decontamination. Thus, the cation resins have the highest curie loading and could exceed Class C limits.

Previous studies (References 1 and 4) indicate that some of the resins generated during full RCS decontamination could be GTCC LLW. One study⁴ that specifically addressed GTCC LLW concluded that these resins could be GTCC only when the RCS was decontaminated with fuel assemblies remaining within the core. This conclusion was based on a review of actual sample data for less than full RCS decontamination projects at both PWRs and BWRs. These projects involved relatively low curie loadings on resin beds, and there was some uncertainty as to whether the data were for the cation beds employed or for mixed beds. Since most of the activity is deposited on the cation resin beds, data for these beds are the best indicator of classification status. This study also considered full RCS decontaminations of BWRs, indicating that potential GTCC LLW could be generated with fuel assemblies remaining in the vessel. The estimated resin volumes, which are

Table F-10. Cartridge filters--activity estimates

CARTRIDGE FILTERS						
Activity Estimates						
Reactor Type	BWR	BWR	PWR	PWR	PWR	PWR
1. Filter Type	Control Rod Drive	Fuel Pool Vacuum	Reactor Coolant	Seal Water Injection	Cavity Drain	Fuel Pool
2. Volume(ft ³)	0.07	0.22	0.45	0.1	0.1	0.85
3. Weight(lbs)	3	6	9	2.5	4.0	20
4. Part 61 Status						
-Table 1 Fraction	1.35	1.52	4.27	2.90	2.25	1.63
-Table 2 Fraction	0.02	0.03	0.03	0.03	0.03	0.01
5. R/hr on Contact	150	100	200	50	50	100
6. Activity(Curies)						
H-3	1.34E-04	2.55E-03	1.44E-03	3.41E-04	8.55E-04	3.24E-03
C-14	3.43E-07	1.10E-03	1.06E-02	2.51E-03	5.13E-03	1.94E-02
Mn-54	3.10E-02	1.35E-01	7.56E-02	1.80E-02	8.55E-03	3.24E-02
Fe-55	3.03E+00	2.29E+00	1.51E+00	3.58E-01	2.37E-01	8.96E-01
Co-58	-	-	9.80E-01	2.17E-01	2.85E-01	1.08E+00
Co-60	5.08E-01	1.14E+00	9.12E-01	2.33E-01	2.74E-01	1.04E+00
Ni-63	2.82E-02	1.16E-01	2.33E-01	5.53E-02	5.79E-02	2.19E-01
Zn-65	-	2.85E-01	-	-	2.85E-03	1.08E-02
Sr-90	1.30E-05	1.22E-03	8.06E-04	1.91E-04	9.50E-05	3.60E-04
Cs-137	5.26E-04	1.80E-02	2.00E-03	4.75E-04	1.90E-04	7.20E-04
Pu-241	1.70E-03	2.05E-03	2.13E-02	5.06E-03	4.42E-03	1.67E-02
TRU	1.35E-04	3.48E-04	1.08E-03	5.61E-04	5.31E-04	9.06E-04
Total Curies	3.60E+00	3.99E+00	3.75E+00	8.90E-01	8.77E-01	3.32E+00

assumed to be cation resins, ranged from 207 ft³ for a large BWR LOMI to 275 ft³ for a large BWR CANDECON. Volumes for full RCS decontamination cation resins at PWRs were not estimated in the study.

Actual data available from two similar projects on cation resin beds, one BWR and one PWR, were reviewed for this project. These data confirmed that the scaling factors for the transuranics, which determine classification status, were within the bounds of the previous GTCC study (Reference 4). The resin curie loadings, which directly affect concentration and hence classification, were also relatively low for these data sets.

The other study,⁵ which specifically addressed full RCS decontamination projects with fuel assemblies out at PWRs, indicated relatively large curie loadings on the cation resin beds. These estimated curie loadings on the cation resin beds for three and four loop PWRs and the cation resin volumes are shown on Table F-11.

This study also estimated activity on the cation resins, but the transuranic scaling factors were at least one order of magnitude lower than those discussed in the decontamination resin GTCC study (Reference 4). When the cation bed resin loadings estimated for this study are adjusted to reflect the transuranic scaling factors found in actual data, the cation resins could be GTCC LLW with the fuel assemblies out.

The above studies bound the problem; however, full RCS decontamination projects have never been performed and will not be until the mid-1990s. For this study, it was assumed that the cation resins generated during full RCS decontamination with fuel assemblies out would not exceed Class C limits. On the other hand, decontamination operations performed with fuel assemblies in the vessel could lead to GTCC LLW at both BWRs and PWRs. While full RCS decontaminations are being considered for implementation in the mid-1990s, these operations will be performed with fuel out. Full RCS decontaminations with fuel assemblies in the vessel will be performed only after full RCS decontamination projects with fuel assemblies removed become standard practice in the industry.

To project full RCS decontamination resin volumes, it was assumed that 50% of the cation resins generated from a typical project would be GTCC LLW and that one such decontamination project would be performed at each plant. The estimated volume of GTCC LLW for a typical BWR was 100 ft³; volume for a PWR was 150 ft³. It was also assumed that plants with less than 20 years left in their 40-year life cycles would not perform full RCS decontaminations with fuel assemblies in. This eliminated 17 BWRs and 26 PWRs from consideration in the volume projections. Cumulative projected RCS decontamination resins volumes are 9,500 ft³, or about 270 m³.

Activities were also estimated. For PWRs, the estimates in the PWR full RCS decontamination study (Reference 5) were revised to increase transuranic scaling factors by a factor of 10. For BWRs, the plant-specific cation resin bed data evaluated for this project were used in conjunction with a higher curie loading on the cation resin bed. These estimates are shown in Table F-12.

This table also shows the transuranic scaling factors for the estimated radionuclide distributions. These scaling factors compare favorably to those reported from actual data on smaller decontamination projects in the GTCC decontamination study (Reference 4).

Table F-11. Full RCS decontamination resin volume estimates--PWRs

FULL RCS DECON RESIN VOLUME ESTIMATES – PWR'S					
	Total Curies	3 Loop Resin (ft^3)	4 Loop Resin (ft^3)	3 Loop (uCi/cc)	4 Loop (uCi/cc)
CAN-DEREM Process					
DEREM 0/1 Cation	5,032	300	300	592	592
DEREM 6/1 Anion	335	546	735	22	16
AP 0/1 Cation	470	990	990	17	17
AP 3/11 Cation	872	692	1,254	45	25
Total	6,709	2,528	3,279	676	650
LOMI Process					
LOMI 0/1 Cation	5,032	752	916	236	194
LOMI 1/0 Anion	335	616	832	19	14
AP 0/1 Cation	470	990	990	17	17
AP 3/11 Cation	872	692	1,254	45	25
Total	6,709	3,050	3,992	317	250

Table F-12. Fuel in/full RCS decontamination resin activities--PWRs

FUEL IN FULL RCS DECONTAMINATION RESIN ACTIVITIES – PWR'S			
Plant Type	PWR RCS Fuel-In	PWR RCS Fuel-In	BWR RCS Fuel-In
1. Decon Method	CAN DEREM	LOMI	LOMI
2. Cation Resin (ft ³)	150	150	100
3. Resin Weight (lbs)	7,500	7,500	5,000
4. Part 61 Status			
- Table 1 Fraction	4.68	1.56	5.84
- Table 2 Fraction	NEG	NEG	NEG
5. Activity (Curies)			
C-14	3.28E+00	1.09E+00	1.00E-02
Mn-54	5.91E+01	1.97E+01	4.27E+00
Fe-55	5.88E+02	1.95E+02	4.51E+01
Co-58	4.10E+02	1.37E+02	1.51E+00
Fe-59	2.63E+01	8.74E+00	1.20E-01
Co-60	1.17E+03	3.90E+02	2.10E+02
Ni-63	7.22E+01	2.40E+01	5.53E+00
Zn-65	1.64E+01	5.46E+00	2.23E+00
Sr-90	6.57E+00	2.18E+00	2.00E-02
Nb-95	4.93E+01	1.64E+01	2.40E-01
Cs-134	6.57E+00	2.18E+00	6.00E-02
Cs-137	9.85E+00	3.28E+00	7.00E-02
Ce-141	2.96E+01	9.83E+00	4.00E-02
Ce-144	6.57E+01	2.18E+01	6.00E-01
Pu-241	3.30E+00	1.10E+00	2.92E+01
Cm-242	1.60E+00	5.00E-01	2.00E-02
TRU	1.50E+00	5.00E-01	4.90E-01
Total Curies	2.52E+03	8.39E+02	3.00E+02
Activity uCi/cc	594	197	106
Transuranic Scaling Factors			
TRU/Ce-144	2.28E-02	2.29E-02	8.17E-01
Pu-241/Ce-144	5.02E-02	5.04E-02	4.87E+01
CM-242/Ce-144	2.44E-02	2.29E-02	3.33E-02
TRU/Co-60	1.28E-03	1.28E-03	2.33E-03
Pu-241/Co-60	2.81E-03	2.82E-03	1.39E-01
CM-242/Co-60	1.36E-03	1.28E-03	9.50E-05
Ni-63/Co-60	6.16E-02	6.16E-02	2.63E-02
C-14/Co-60	2.80E-03	2.80E-03	4.75E-05

5.3 Process Waste Summary

Table F-13 summarizes the generation rates for both BWR and PWR process waste. Two sets of values are presented: a maximum set that addresses all cartridge filters; and a minimum set that assumes that PWR plants will manage all filters to eliminate them as GTCC LLW, and that BWR plants will similarly manage fuel pool filters. RCS decontamination resin estimates are the same for both cases.

This table also shows the projected cumulative volumes of GTCC LLW generated from the plants considered from January 1993 through plant closure. These cumulative values are the generation rates per cycle by reactor vendor times the number of fuel cycles remaining before closure for each plant considered. The impact of filter cartridge practices can be significant. Improved filter practices can reduce total process waste volumes by about 25%. The maximum case volume of about 13,500 ft³, or 380 m³, is reduced to about 10,100 ft³, or 285 m³.

Table F-13. Summary of process GTCC LLW

SUMMARY OF PROCESS GTCC WASTES								
	Number Of Plants	Unit Volume (ft ³)	Units Per Cycle	Percent GTCC	GTCC Units Cycle	GTCC (ft ³) Cycle	Cycles To Shutdown	Total GTCC (ft ³)
MAXIMUM ESTIMATES								
1.BWR Cartridge Filters	36							
a. Control Rod Drives		0.07	30	50	15	1.05	564	592
b. Fuel Pool/Vacuums		0.22	12	25	3	0.68	564	372
							Subtotal	964
2.PWR Cartridge Filters	77							
a. Reactor Coolant Systems		0.45	8	25	2	0.9	1,327	1,194
b. Seal Water Injections		0.1	16	25	4	0.4	1,327	530
c. Cavity Drains		0.1	2	25	0.5	0.05	1,327	66
d. Spent Fuel Pools		0.85	4	25	1	0.85	1,327	1,128
							Subtotal	2,918
3.B&W Crud Tank Filters	7	0.95	0.5	100	0.5	0.475	100	48
4.BWR RCS Decon Resins	19	200	NA	50	100	NA	1	1,900
5.PWR RCS Decon Resins	51	300	NA	50	150	NA	1	7,650
							Maximum Totals	13,480
MINIMUM ESTIMATES								
1.BWR CRD Filters	36	0.07	30	50	15	1.05	564	592
2.BWR RCS Decon Resins	19	200	NA	50	100	NA	1	1,900
3.PWR RCS Decon Resins	51	300	NA	50	150	NA	1	7,650
							Minimum Totals	10,142

6. DECOMMISSIONING ACTIVATED METALS PROJECTIONS AND ACTIVITIES

The largest sources of potential GTCC LLW in terms of volume and activity are the activated metals within the reactor vessel. This section describes these components and estimates their volumes and activities at plant shutdown. It also describes the methods used to estimate component activities at shutdown.

6.1 Potential GTCC LLW Sources

These activated metal components consist of the structural components that support the fuel assemblies in the reactor vessel. Their physical characteristics vary with LWR plant type and the component application. 10 CFR Part 61 activation product radionuclide content is directly proportional to the integrated neutron flux received by the component during a plants operating lifetime. A previous WMG study (Reference 1) identified the components described below as potential GTCC LLW.

6.1.1 BWR Decommissioning Internals

The potential GTCC LLW in BWRs consists of the shroud and, in some cases, the upper fuel guide. These are described below.

- Core Shroud--The shroud is a large, cylindrical, type 304 stainless steel structure of about 90 to 180 ft³. It surrounds the core and provides a barrier to separate the upward flow through the core from the downward flow in the annulus. A flange at the top of the shroud mates with a flange on the steam separator assembly to form the core discharge plenum.
- Top Fuel Guide--The top fuel guide is a stainless steel structure that provides lateral support for the top section of the fuel assemblies. It is composed of about 10 to 12 ft³ of stainless steel beams that are joined at right angles, forming square openings for the fuel assemblies. The beams are fastened to a peripheral rim that rests on the core shroud.

As shown previously in Table F-3, 36 BWRs were considered. Nine of these plants were General Electric reactor models 2 or 3; 13 were reactor model 4; two were referred to as model 4/5; and 12 were reactor model 6, General Electric's largest plants. Generally, the size of the potential GTCC components increases with the reactor model number. This evaluation considered two of the larger reactor models, a BWR-4 and a BWR-6.

6.1.2 PWR Decommissioning Internals

The potential GTCC LLW in PWRs consist of the core baffle (shroud) and the lower core barrel. The thermal shield was also considered. These are described below.

- Core Baffle (Shroud)--The baffle is a type 304 stainless steel structure that surrounds the core. It provides for a limited water flow between the baffle and the core barrel for additional cooling of the barrel. The core shroud has a metal volume of about 45 to 90 ft³.

- **Lower Core Barrel**--The lower core barrel is a type 304 stainless steel cylinder that provides lateral support for the core and directs the flow of coolant water. The lower core barrel typically has a metal volume of about 75 to 125 ft³. For Combustion Engineering plants, the core barrel is relatively large with volumes in the 250 to 300 ft³ range. These units do not have thermal shields.
- **Thermal Shield**--The thermal shield is attached to the lower core barrel in the high flux region. It protects the reactor vessel by attenuating gamma radiation and fast neutrons that escape the core. The heat energy deposited in the thermal shield is removed by the reactor coolant. The thermal shield is type 304 stainless steel and has a metal volume of about 50 to 110 ft³.
- **Flow Skirt**--Combustion Engineering plants have an inconel flow skirt as part of the lower core barrel assembly. This component, which has a metal volume of about 10 ft³, will be GTCC LLW. This component was not identified in the previous WMG study (Reference 1).

As shown previously in Table F-4, 77 PWRs were considered. Seven of these plants were Babcock & Wilcox units, 15 were Combustion Engineering plants, and 55 were Westinghouse plants. This evaluation considered the larger plants from each reactor vendor.

6.2 Methodology Used to Determine Component Activities and Classification Status

The ANISN⁶ and ORIGEN2⁷ codes were employed to estimate end-of-life activities for potential GTCC components. These are the same codes used previously with typical PWR and BWR plant results listed in NRC reports^{8,9} on decommissioning. For these calculations, the reactor models differed slightly from those reported in the NRC documents. The ANISN code was used to develop radial neutron transport models. Unlike the NUREG approach, the ANISN results were then normalized to actual surveillance capsule flux data obtained from plant-specific surveillance reports. This normalization benchmarked the ANISN results to actual data and provided a more realistic determination of the flux levels and energy spectra at the radial locations of interest for the decommissioning components considered. The surveillance capsule normalized flux data were then used in conjunction with the operating histories discussed in Section 3 as input to ORIGEN2 to determine component activities. The ORIGEN2 cross section libraries used were more recent versions than those used in the NRC documents.

The stainless steel materials compositions used for the activation calculations were obtained directly from NUREG 3474 (Reference 2).

Details of how this methodology was employed, and one of the cases considered are presented below.

6.2.1 Reference Plants

Models could not be developed for all the LWRs considered. Table F-14 summarizes the characteristics of the six reactors chosen for detailed evaluation. For BWRs, the two plants chosen are the larger BWR-4 and BWR-6 models. For projection purposes, the BWR-4 plants will

Table F-14. Reference reactors summary

REFERENCE REACTORS SUMMARY						
Reference	A	B	C	D	E	F
Reactor Vendor	GE	GE	B&W	CE	West.	West.
Reactor Type	BWR-4	BWR-6	177a		Mid-Size	SNUPS
Number of Fuel Assemblies	560	748	177	217	173	193
Heat Output (MWt.)	2,440	3,580	2,770	3,390	3,020	3,410
Initial Enrichment	3.50 %	3.50 %	4.50 %	4.50 %	4.50 %	4.50 %
Potentially GTCC Components and Volumes (ft ³)	Core Shroud (102)	Core Shroud (176)	Core Baffle (52)	Core Baffle (88)	Core Baffle (45)	Core Baffle (71)
			Core Barrel (90)	Core Barrel (290)	Core Barrel (125)	Core Barrel (125)
			Thermal Shield (96)	Thermal Shield None	Thermal Shield (110)	Thermal Shield (45)
Total Volumes	(102)	(176)	(238)	(378)	(280)	(241)

conservatively estimate values for the smaller BWR-2 and BWR-3 models. Both the Babcock & Wilcox and Combustion Engineering plants selected are also among the largest from these vendors. Similarly, two of the larger reactor models from Westinghouse were selected. The mid-size plant evaluated will conservatively estimate values for the smaller Westinghouse plants.

6.2.2 One-Dimensional Transport Calculations

The ANISN computer code was used to perform the one-dimensional neutron transport calculations required by this evaluation. The ANISN cylindrical geometry source was used in the radial model calculations.

The CASK-81 cross section library was the source of neutron cross sections used. The data in the library were collapsed from a fine-group structure using a weighting function representative of a water-uranium mixture.

The GIP computer code¹⁰ was used to read the nuclide-organized CASK cross section library, and create a group-organized input library to be read by ANISN. GIP was utilized to homogenize the compositions of components or materials in given regions of the ANISN model and to create a working library of homogenized cross sections suitable for input into ANISN.

6.2.3 Point Neutron Activation Calculations

The ORIGEN2 computer code was used to calculate the activation and depletion of radionuclides of components exposed to neutrons. One-group neutron cross sections for 22 reactor models are available as input to the code, three of which were used specifically for this evaluation. The first of these one-group cross section sets is representative of a neutron spectrum collapsed across the reactor core of a PWR that has operated to a fuel burnup of 50,000 MW·d/MTU (PWR-UE, Reference 7). The second cross section data set is representative of a BWR which has a fuel burnup of 27,500 MW·d/MTU (BWR-USO, Reference 7). The third is a thermal cross section set, containing only thermal neutron cross sections.

For each location or component of interest and for each material of interest, the volume-averaged total neutron flux (neutrons/cm²·sec), as calculated by ANISN, was input into two ORIGEN2 calculations. One calculation used the reactor core cross section set as input (PWR-UE or BWR-USO); the other used the thermal cross section set. Thermal curie per gram values were adjusted for the local area temperatures to reflect the decrease of activation cross sections at elevated temperatures. Based on the ANISN spectral results of thermal-to-fast-neutron flux ratios, the two ORIGEN2 calculations were weighted to obtain location-specific neutron activation.

6.2.4 Operating Histories

For the two BWR cases, it was assumed that the plants operated for 40 years after commercial operation at a 65% capacity factor. For the four PWR cases, a 70% capacity factor over the same 40-year time period was used.

6.2.5 Radial ANISN Models

The modeling of the reference reactors for this activation analysis included geometry and materials considerations.

The PWR ANISN radial models consisted of the reactor core, the core baffle, the core barrel, the thermal shield, the reactor vessel cladding and wall, and the primary shield liner and primary shield wall in cylindrical geometry.

The BWR ANISN models consisted of the reactor core, the core shroud, the reactor vessel cladding and wall, and the sacrificial shield liner and wall in cylindrical geometry. Radial distances from the core centerline at the core midplane elevation for these components have been used. Small components that may be found as part of the reactor internals were neglected for this evaluation. The models included all air and water gaps.

The reactor core region of the reference reactor radial ANISN models were assumed to be comprised of three regions utilizing a fuel loading strategy whereby fuel with the most exposure resides in the core center, and fuel with the least exposure resides around the core periphery.

The core compositions were determined by running ORIGEN2 with plant-specific parameters for fresh, once exposed, and twice exposed assemblies over a fuel cycle. The resulting U-235, U-238, Pu-239, and Pu-240 concentrations for each core region were both volume averaged and cycle averaged, and they were used as ANISN input.

6.2.6 Radial Model ANISN Calculations

Two radial model ANISN calculations were performed for each reference reactor. The first of these models assumed an axially averaged, flat core power distribution, with the total neutron source equal to that calculated by ORIGEN2 for a three-region core. In the second ANISN radial model calculation, the neutron flux above 1 MeV was normalized to the results obtained from the analysis of the surveillance capsules.

6.3 Description of a Typical Reference Plant Case

The characteristics of the large Westinghouse unit (Case F) are summarized in Table F-15. The parameters used for the radial ANISN model are presented in Table F-16. The model neglected the reactor vessel thermal insulation, but did consider the primary shield wall. The biological shield was assumed to be six feet of standard concrete.

Usually, two ANISN cases are required per reactor model. The configuration of the thermal shield (four separate pads) in the large Westinghouse PWR required four ANISN cases. Two cases were required for the core baffle and barrel, and two other cases were required for the thermal shield pads. The first two cases assumed an axially averaged, flat radial core power distribution. The neutron source was set equal to that calculated by ORIGEN2 for a three-region core. In the second two cases, the ANISN-calculated neutron flux above 1 MeV was normalized to the results obtained by Westinghouse for the plant-specific surveillance capsule. The configuration of the core internals and the location of the surveillance capsule are illustrated in Figure F-1.

Based on the surveillance capsule analysis, the neutron flux above 1 MeV for the maximum axial and the maximum azimuthal position on the reactor vessel clad interface was calculated to be $3.14\text{E}+10$ n/cm²·sec. This value represents an average neutron flux above 1 MeV of $2.38\text{E}+10$ n/cm²·sec. The ANISN radial model was normalized to this value. A plot for the surveillance capsule normalized thermal flux as a function of radial location is shown on Figure F-2.

Table F-15. Large Westinghouse PWR--reactor summary

LARGE WESTINGHOUSE PWR Reference Case F Reactor Summary	
Rated Thermal Power (MWt.)	3,411
Number of Fuel Assemblies	193
Average Moderator Temperature	591.8
Average Boron Concentration in Moderator	800
Average Core Void Fraction	0
Reactor Assembly Pitch (in.)	8.466
Fuel Assembly Type	West 17x17 OFA
Number of Rod Locations	289
Number of Fueled Rods	264
Fuel Pellet O.D. (in.)	0.3088
Fuel Rod O.D. (in.)	0.36
Fuel Clad Thickness (in.)	0.0225
Fuel Clad Material	Zircaloy-4
Active Fuel Length (in.)	144
Fuel Pin Pitch (in.)	0.496
Metric Tonnes Uranium Metal (MT)	0.426
Initial Enrichment (w/o U-235) %	4.50
Core Volume Calculations	
Total Core Volume (cm.**3)	3.264E+07
Fuel Volume (cm.**3)	9.005E+06
Clad Volume	2.868E+06
Zircaloy-4 Volume (cm.**3)	4.870E+05
Inconel-718 Volume (cm.**3)	2.180E+04
Moderator Volume (cm.**3)	1.924E+07
Void Volume (cm.**3)	1.015E+06

Table F-16. Large Westinghouse PWR--core model parameters

LARGE WESTINGHOUSE PWR Reference Case F Core Model Parameters	
Core Fractions	
Fuel	0.2759
Moderator	0.5896
Zircaloy-4	0.1028
Inconel-718	0.0007
Void	0.0311
Total Core Fractions	1.0000
Moderator Density at Power	0.6884
Model Geometry (cm.)	
Core Equivalent Radius	168.54
Water Region	168.62
Core Shroud/Baffle Equivalent Outer Radius	170.84
Water Region	187.96
Core Barrel Outer Radius	193.68
Thermal Pad Thickness	6.98
Thermal Shield Outer Radius	200.66
Water Region	219.39
Vessel Cladding Outer Radius	219.71
Vessel Wall Outer Radius	41.62
Air Gap Outer Radius	307.34
Primary Shield Liner Outer Radius	307.98
Primary Shield Outer Radius	490.86

Figure F-1. Arrangement of surveillance capsules in the reactor vessel.

Westinghouse Large PWR Thermal Neutron Flux Surveillance Capsule Normalized

Figure F-2. PWR thermal neutron flux.

The thermal and total neutron fluxes were then determined using the normalized ANISN flux results for the core baffle, core barrel, and thermal shield. The total neutron flux was then input to four ORIGEN2 cases for each component, a total of twelve ORIGEN2 cases. For the simpler geometries of the other reference plants, only six ORIGEN2 cases were required. Normally, the fraction of the neutron flux that fell into the thermal energy range was calculated from the ANISN output data. The fraction of neutron flux above thermal energies was also calculated. The resultant spectral fractions were then compared to the ANISN spectral fractions in the core region. Finally, ORIGEN2 neutron activation calculations were performed to obtain radionuclide activities in curies per gram for the composite core spectrum. The resultant ORIGEN2 calculated activities in curies per m³, and the corresponding curie content for the three components considered, are presented in Table F-17. This table also shows the 10 CFR Part 61 classification status for the three components. As expected, the core baffle is well above Class C limits, and the thermal shield is below Class C limits. The core barrel is only slightly above Class C limits. This indicates that the core barrel may not be GTCC LLW in all cases. Since this component represents a significant GTCC LLW volume, investigations beyond those performed for this evaluation are warranted.

Since no axial ANISN model cases were run, the curie results are slightly conservative. They assume that the activation at the periphery of the core also applies to component masses above and below the core. Typically, 10–20% of the mass of the three components of interest lies above and below the core.

The results for this case were compared to those reported in the NUREG report (Reference 8) for a large PWR, the Trojan plant. These comparisons of concentrations for the radionuclides Fe-55, Co-60, Ni-63, and Nb-94 are shown in Table F-18. In some cases, the NRC report reported high or maximum radionuclide values. This evaluation determined average values for these radionuclides. Thus, the NUREG values shown in the Table F-18 have been reduced by a factor of 1.22 to account for peaking. As shown, the values obtained from this evaluation for the core baffle are comparable to the NUREG reported values. This is not the case for the core barrel, where results of this evaluation indicate activities well below the NUREG results.

6.4 Results of Reference Plant Cases

The results of the reference plant analysis for the two BWRs are summarized in Table F-19. As shown, the core shrouds for both the BWR-4 and BWR-6 are above Class C limits. This table also presents the ANISN-calculated and surveillance capsule normalized flux for the BWR-6 case. It is significant that surveillance capsule normalization for this case reduces the incident flux on the shroud by about 35% with corresponding reductions in activity. It is likely that a similar reduction in activity would result for the BWR-4 case if the surveillance capsule data were made available and analyzed.

The BWR cases evaluated were for the larger plants with relatively short operating histories. Some of the older, smaller BWRs have operating histories of up to 12 fuel cycles. Evaluations of these plants could reduce some of the uncertainties in the 10 CFR Part 61 classification status of BWR core shrouds, since the plants are farther along in their life cycles.

Table F-17. Large Westinghouse PWR--summary of results

LARGE WESTINGHOUSE PWR Reference Case F Summary of Results			
	Core Baffle	Core Barrel	Thermal Shield
1. Component Volume (ft ³)	71	125	45
2. Flux (n/cm ² -sec)	1.53E+14	7.90E+12	4.72E+12
3. Normalized Flux (n/cm ² -sec)	7.32E+13	3.98E+12	2.26E+12
4. ORIGEN2 Concentrations (Curies/M ³)			
C-14	1.75E+02	1.17E+01	3.50E+00
Mn-54	9.95E+04	5.00E+03	3.20E+03
Fe-55	9.61E+05	6.57E+04	1.90E+04
Co-60	5.67E+05	4.40E+04	1.71E+04
Ni-59	6.98E+02	6.74E+01	2.09E+01
Ni-63	1.12E+05	8.60E+03	2.52E+03
Nb-94	3.18E+00	1.99E-01	1.10E-01
Totals	1.74E+06	1.23E+05	4.18E+04
5. Part 61 Status			
- Table 1 Fraction	21.26	1.45	0.69
- Table 2 Fraction	16.01	1.23	0.36
6. Radionuclide Content (Curies)			
C-14	3.52E+02	4.14E+01	4.46E+00
Mn-54	2.00E+05	1.77E+04	4.08E+03
Fe-55	1.93E+06	2.33E+05	2.42E+04
Co-60	1.14E+06	1.56E+05	2.18E+04
Ni-59	1.40E+03	2.39E+02	2.66E+01
Ni-63	2.25E+05	3.04E+04	3.21E+03
Nb-94	6.39E+00	7.04E-01	1.40E-01
Totals	3.50E+06	4.37E+05	5.33E+04

Table F-18. Large Westinghouse PWR--radionuclide concentration comparison

LARGE WESTINGHOUSE PWR Reference Case F Radionuclide Concentration Comparison			
Core Baffle			
Radionuclides	Study Result (Curies/M³)	NUREG Result (Curies/M³)	Ratio NUREG/ Study (Factor)
Fe-55	9.61E+05	1.07E+06	1.11E+00
Co-60	5.67E+05	7.87E+05	1.39E+00
Ni-63	1.12E+05	9.84E+04	8.78E-01
Nb-94	3.18E+00	4.43E+00	1.39E+00
Core Barrel			
Radionuclides	Study Result (Curies/M³)	NUREG Result (Curies/M³)	Ratio NUREG/ Study (Factor)
Fe-55	6.57E+04	1.23E+05	1.87E+00
Co-60	4.40E+04	7.60E+04	1.73E+00
Ni-63	8.60E+03	1.23E+04	1.43E+00
Nb-94	1.99E-01	2.10E-01	1.06E+00

Table F-19. BWR decommissioning (activity estimates)

BWR DECOMMISSIONING Activity Estimates		
Reference Reactor	A	B
Vendor	GE BWR 4	GE BWR 6
1. Component Type	Shroud	Shroud
2. Volume (ft³)	102	176
3. Weight (lbs)	50,400	87,000
4. Flux (n/cm²-sec) (ANISN)	5.85E+12	5.57E+12
5. Normalized Flux (Surveillance Capsule)	Not Available	3.55E+12
6. Part 61 Status		
- Table 1 Fraction	2.76	1.58
- Table 2 Fraction	3.37	1.81
7. Activity - (Curies)		
C-14	9.23E+01	8.39E+01
Mn-54	1.28E+04	1.36E+04
Fe-55	5.45E+05	4.89E+05
Co-60	3.60E+05	3.57E+05
Ni-59	4.81E+02	4.69E+02
Ni-63	6.81E+04	6.32E+04
Nb-94	9.22E-01	9.33E-01
Total Curies	9.87E+05	9.23E+05

The results of the reference plant analysis for the four PWRs are summarized in Table F-20. Note that the Combustion Engineering plant does not have a thermal shield. It does, however, have an inconel flow skirt with a volume of about 10 ft³. This flow skirt will be well above Class C limits, and its volume is included in the core baffle volume of 88 ft³.

Surveillance capsule data were available for three of the four PWR cases analyzed. In two of the three cases, the normalized flux was consistently lower than the ANISN-calculated flux for each component resulting in reduced activities. For the moderate-size PWR (Case E), the normalized flux was higher than the ANISN-calculated flux. This apparent anomaly was discussed with the operator of this reference plant. During the early years of plant operation, measures to reduce flux at the vessel wall were not implemented, and this mode of operation is reflected in the surveillance capsule data. The current mode of operation at this reference plant will reduce fluxes at the core periphery. Thus, surveillance capsule normalization for PWRs also leads to more realistic estimates of flux for use in activation calculations.

As shown in Table F-20, PWR core baffles are well above Class C limits, and, as expected, the thermal shields are below Class C limits. There is still some uncertainty for some of the core barrels that represent a significant GTCC volume. The analyses indicate that these components in Westinghouse plants are just above Class C limits. Thus, additional investigation of core barrels appears warranted.

The PWR cases evaluated were for the larger plants with relatively short operating histories. Similar analyses of some of the smaller PWRs with longer operating histories would reduce some of the uncertainties in the 10 CFR Part 61 status of the PWR core barrels.

Table F-20. PWR decommissioning

PWR DECOMMISSIONING									
Activity Estimates									
Reference Reactor	E			E			F		
Vendor	West. 1			West. 1			West. 2		
1. Component Type	Core Baffle	Core Barrel	Thermal Shield	Core Baffle	Core Barrel	Thermal Shield	Core Baffle	Core Barrel	Thermal Shield
2. Volume(ft ³)	45	125	110				71	125	45
3. Weight(lbs)	22300	61700	54500				35200	46000	22200
4. Flux(n/cm ² -sec) (ANISN)	1.38E+14	1.94E+12	3.49E+11				1.53E+14	7.90E+12	4.72E+12
5. Normalized flux (Surveillance Capsule)	2.06E+14	2.91E+12	5.22E+11				7.32E+13	3.98E+13	2.26E+12
6. Part 61 Status									
-Table 1 Fraction	45.41	1.12	0.20				21.26	1.45	0.69
-Table 2 Fraction	37.36	0.94	0.16				16.00	1.23	0.36
7. Activity(Curies)									
C-14	5.95E+02	3.14E+01	4.62E+00				3.52E+02	4.14E+01	4.46E+00
Mn-54	3.09E+05	1.21E+04	1.94E+03				2.00E+05	1.77E+04	4.08E+03
Fe-55	3.09E+06	1.69E+05	2.49E+04				1.93E+06	2.33E+05	2.42E+04
Co-60	1.46E+06	1.18E+05	1.80E+04				1.14E+06	1.56E+05	2.18E+04
Ni-59	1.47E+03	1.84E+02	2.81E+01				1.40E+03	2.39E+02	2.66E+01
Ni-63	3.33E+05	2.33E+04	3.49E+03				2.25E+05	3.04E+04	3.21E+03
Nb-94	8.74E+00	5.42E-01	8.53E-02				6.39E+00	7.04E-01	1.40E-01
Total Curies	5.19E+06	3.23E+05	4.84E+04				3.50E+06	4.37E+05	5.33E+04

Table F-20. (continued)

PWR DECOMMISSIONING						
Activity Estimates						
Reference Reactor	C	C	C	D	D	D
Vendor	B&W	B&W	B&W	CE	CE	CE
1. Component Type	Core Baffle	Core Barrel	Thermal Shield	Core Baffle	Core Barrel	Core Barrel
2. Volume(ft ³)	52	90	96	78	290	290
3. Weight(lbs)	44400	25600	47600	43900	143200	143200
4. Flux(n/cm ² -sec) (ANISN)	1.26E+13	1.20E+14	4.45E+12	1.31E+14	1.23E+13	1.23E+13
5. Normalized flux (Surveillance Capsule)	7.34E+12	7.02E+13	2.60E+12	NA	NA	NA
6. Part 61 Status						
-Table 1 Fraction	21.50	2.90	0.97	33.01	4.59	4.59
-Table 2 Fraction	16.83	2.60	0.77	26.43	3.91	3.91
7. Activity(Curies)						
C-14	2.99E+02	6.41E+01	1.98E+01	5.50E+02	3.19E+02	3.19E+02
Mn-54	1.55E+05	2.77E+04	1.09E+04	4.09E+05	1.41E+05	1.41E+05
Fe-55	1.70E+06	3.75E+05	1.15E+05	3.84E+06	1.81E+06	1.81E+06
Co-60	8.56E+05	2.32E+05	7.94E+04	1.96E+06	1.13E+06	1.13E+06
Co-59	1.03E+03	3.44E+02	1.17E+02	2.16E+03	1.62E+03	1.62E+03
Ni-63	1.73E+05	4.65E+04	1.47E+04	4.09E+05	2.25E+05	2.25E+05
Nb-94	4.65E+00	1.01E+00	3.71E-01	1.12E+01	5.27E+00	5.27E+00
Total Curies	2.89E+06	6.82E+05	2.20E+05	6.63E+06	3.30E+06	3.30E+06

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Greater-Than-Class C Low-Level Radioactive Waste Characterization

Appendix G: Evaluation of Potential for Greater-Than-Class C Classification of Irradiated Hardware Generated by Utility-Operated Reactors

*Greater-Than-Class C Low-Level Waste
Management Program*

August 1991

**GREATER-THAN-CLASS C
LOW-LEVEL WASTE CHARACTERIZATION**

**Appendix G: Evaluation of Potential
for Greater-Than-Class C Classification
of Irradiated Hardware
Generated by Utility-Operated Reactors**

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ABSTRACT

This study compiles and evaluates data from many sources to expand a base of data from which to estimate the activity concentrations and volumes of greater-than-Class C low-level waste that the Department of energy will receive from the commercial power industry. Sources of these data include measurements of irradiated hardware made by or for the utilities that was classified for disposal in commercial burial sites, measurements of neutron flux in the appropriate regions of the reactor pressure vessel, analyses of elemental constituents of the particular structural material used for the components, and the activation analysis calculations done for hardware. Evaluations include results and assumptions in the activation analyses. Sections of this report and the appendices present interpretation of data and the classification definitions and requirements.

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APPENDIX G: EVALUATION OF POTENTIAL FOR GREATER-THAN-CLASS C CLASSIFICATION OF IRRADIATED HARDWARE GENERATED BY UTILITY-OPERATED REACTORS

1. SUMMARY AND CONCLUSIONS

The U.S. Department of Energy (DOE) has responsibility for receiving and disposing of all low-level radioactive waste classified as greater-than-Class C (GTCC). A large source of waste, potentially GTCC, is irradiated metal components from nuclear reactors, of which the commercial nuclear power industry is the largest source. Nonfuel-bearing hardware internal to the pressure vessel in nuclear reactors absorbs neutrons during reactor operation and becomes highly radioactive. The bulk of the activity is ^{60}Co , and other radionuclides more important to waste classification usually have concentrations several orders of magnitude below its level. If the neutron flux seen by the components is sufficiently high, these other radioactive constituents could reach concentrations high enough to classify the material as GTCC LLW, requiring its disposal by DOE. Irradiated hardware that is potential GTCC LLW falls into two categories: that discharged in normal refueling and maintenance operations, and that discharged in decommissioning the reactor at the end of its operating lifetime. The present study evaluates activity levels and volumes of nonfuel-bearing components from commercial power generation that could classify as GTCC material.

The nuclides ^{59}Ni , ^{63}Ni , ^{94}Nb , and, to a lesser extent, transuranics (TRU), generally control the waste classification of activated hardware. Nickel and TRU nuclides emit only weak x-ray, beta, or alpha activities that are nearly impossible to measure in the presence of the dominant cobalt radiation, and the niobium isotope, although emitting energetic gamma rays, is present at such low levels that it, too, is overwhelmed by the cobalt activity. Thus, practical waste classification techniques for irradiated hardware (as for most other utility-generated waste) relate all other activities to the observable ^{60}Co concentration through scaling factors. Since cobalt is present in most reactor structural materials as a relatively uncontrolled trace material, as is niobium in all but inconel, current classification schemes give rise large uncertainties in estimating concentrations for the desired activities. A lack of data from measurements of radioactivity in many irradiated components adds to uncertainty in the estimates, particularly for those items discharged at decommissioning; many of the estimates rely solely on activation calculations that use estimates of the elemental compositions of the material, the average neutron flux seen by that component, and, in some cases, the neutron activation cross section for the reaction. Uncertainties in these values exacerbate total uncertainty in estimated activities of the waste and, therefore, estimated volumes of GTCC material.

The present study has compiled and evaluated data from many additional sources to expand a base of data from which to estimate the activity concentrations and volumes of GTCC LLW that DOE will receive from the commercial power industry. Sources of these data include measurements of irradiated hardware made by or for the utilities that was classified for disposal in commercial burial sites, measurements of neutron flux in the appropriate regions of the reactor pressure vessel, analyses of elemental constituents of the particular structural material used for the components, and the activation analysis calculations done for hardware. Evaluations include results and assumptions in the activation analyses. Sections of this report and the appendices present interpretation of data and the

classification definitions and requirements. This section of the report summarizes and interprets the results.

1.1 Radionuclide Concentrations

Table G-1 summarizes the evaluation of radioactive concentrations of five radionuclides important for waste classification of reactor-activated hardware removed in normal plant operation: ^{59}Ni , ^{63}Ni , ^{94}Nb , ^{99}Tc , and TRU (transuranic activity, TRU, is a component of the surface contamination acquired from contact with the reactor coolant). Section 3.1 presents and discusses data from which the summary is made. The table lists the seven boiling water reactor (BWR) and one pressurized water reactor (PWR) components identified as having the greatest potential for classification as GTCC LLW: the control rod blade (CRB), three incore instruments—local power range monitors (LPRM), intermediate range monitors (IRM), and source range monitors (SRM) poison curtains, fuel support piece, fuel flow channel from boiling water reactors, and the PWR fuel assembly skeleton. Section 2.3 describes these components.

Table G-1 expresses concentrations as ratios to the Class C activity concentration upper limits for that nuclide and highlights values that exceed 1.0 (GTCC). Section 2.2 describes the waste classification rules and practices. Data in the table are for typical discharge frequencies. Values for the ranges result from statistical analyses of measurement results. The table demonstrates the dominance of ^{63}Ni , a 10 CFR 61 Table 2 nuclide, in classifying waste. The other four activities are 10 CFR 61¹ Table 1 nuclides and contribute primarily through the "sum of the fractions" rule. The data in the table indicate that only the BWR incore instrumentation and PWR inconel-718 portions of the fuel assembly skeleton may have concentrations that exceed the Class C limits. Only skeletons manufactured by Westinghouse have significant portions made of inconel. Table G-1 has two entries for the incore instruments LPRM, IRM and SRM. These components are about 43 ft long, only 12 ft of which are in the reactor core (hot end). The portion labeled "hot end" represents this 12-ft section, and the other entry averages the concentration over the entire 43-ft long piece. Although the incore instrument tubes are GTCC LLW, they are low volume items, and the utilities usually dispose of them as Class C waste under the "factor of ten rule" by mixing them with the larger volume CRBs and averaging over the volume of SS-304 in the casks.

Table G-2 lists reactor internal components with greatest potential for GTCC classification and summarizes the nuclide concentrations expected at the end of the reactor lifetime. Section 2.2 discusses these components. The table here also expresses concentrations as ratios to the Class C upper concentration limits for that nuclide and highlights values that exceed 1.0. All values assume a 40-year reactor lifetime and a 75% duty cycle of full power operation. The table summarizes data presented and discussed in Section 3.2. The spread between the high and low values is very large and represents the uncertainties from a lack of data. The high values are principally from computational estimates of neutron fluxes and cross sections, and the low estimates rely on measured scaling factors and a measurement set of neutron flux in a region outside the core. The base values are a log-mean average of the two and represent the best estimate that can be made at this time. Base values indicate that none of the hardware may reach GTCC classification status, even with plant life extensions of 20 years.

Table G-1. Summary of average radionuclide concentrations for irradiated hardware generated in normal operations

	COMPONENT		RATIO OF ACTIVITIES TO CLASS "C" LIMITS RADIONUCLIDE				
			Ni-59	Ni-63	Nb-94	Tc-99	TRU
BWR	CRB	Min:	0.02	0.10	0.03	0.0003	0.0032
		Max:	0.29	0.46	0.14	0.0200	0.0200
		Avg:	0.09	0.24	0.07	0.0100	0.0100
	LPRM Hot End	Min:	0.11	0.54	0.16	0.0018	0.0180
		Max:	6.16	9.86	2.80	0.0300	0.3200
		Avg:	1.59	4.29	1.20	0.0100	0.1400
	with Cold End	Min:	0.03	0.15	0.05	0.0005	0.0050
		Max:	1.72	2.75	0.78	0.0084	0.0023
		Avg:	0.44	1.20	0.33	0.0028	0.0008
	IRM Hot End	Min:	0.85	4.00	1.15	0.0100	0.1300
		Max:	6.43	10.29	2.95	0.0300	0.3400
		Avg:	2.22	6.00	1.70	0.0200	0.2000
	with Cold End	Min:	0.24	1.12	0.32	0.0028	0.0360
		Max:	1.79	2.87	0.82	0.0080	0.0950
		Avg:	0.62	1.67	0.47	0.0056	0.0560
	SRM Hot End	Min:	1.13	5.29	1.55	0.0200	0.1700
		Max:	3.57	5.71	1.65	0.0200	0.1900
		Avg:	2.06	5.57	1.60	0.0200	0.1800
	with Cold End	Min:	0.32	1.50	0.43	0.0050	0.0470
		Max:	1.00	1.60	0.60	0.0060	0.0530
		Avg:	0.57	1.55	0.50	0.0056	0.0500
	POISON CURTAINS	Min:	0.08	0.36	0.10	0.0012	0.0120
		Max:	0.42	0.67	0.19	0.0023	0.0220
		Avg:	0.19	0.51	0.15	0.0016	0.0170
	FUEL SUPPORT PIECE						
		Avg:	0.004	0.020	0.005	0.0001	0.0050
	FUEL FLOW CHANNEL	Min:	1.1E-04	1.7E-04	0.002		0.0870
		Max:	1.1E-03	2.9E-03	0.040		0.1400
		Avg:	3.6E-04	9.7E-04	0.010		0.0490
	PWR FUEL ASSEMBLY						
		SS-304	0.024	0.066	0.14		
		ZIRCALOY-4	0.013	0.035	0.34		
		INCONEL-718	0.93	2.50	12.30		

Table G-2. Summary of high and low radionuclide concentration estimates for reactor internal components

COMPONENT	RANGE	RADIONUCLIDE CONCENTRATION RELATIVE TO CLASS "C" LIMITS				
		Ni-59	Ni-63	Nb-94	Tc-99	
<u>PWR</u>	SHROUD	HIGH: (1)	8.5	13.3	10.3	0.22
		BASE: (2)	0.62	1.7	0.79	9.3E-03
		LOW: (3)	0.045	0.21	0.06	3.9E-04
	CORE BARREL	HIGH: (1)	1.1	1.8	0.77	0.01
		BASE: (2)	0.085	0.23	0.06	1.3E-03
		LOW: (3)	6.4E-03	0.03	4.6E-03	1.8E-04
<u>BWR</u>	CORE SHROUD	HIGH: (1)	2.7	4.3	1.7	2.7E-03
		BASE: (2)	0.13	0.55	0.13	3.6E-04
		LOW: (3)	5.9E-03	0.07	0.01	4.8E-05

1.2 Elemental Constituents in Reactor Hardware Alloys

Table G-3 summarizes the data and analyses presented in Appendix B for elemental constituents of alloys used for most reactor hardware. The relative abundances of these elements, many of which are only trace constituents, are very important for classifying the irradiated pieces. The table includes the nominal value (i.e., the manufacturer's specification, measured values previously reported for SS-304,² and the average values from two additional sources), material chemically analyzed for its components, and constituents deduced from measured activated radioactivity. The latter analyses include the activation cross sections and are a more pragmatic indicator of the expected radioactivities from activation. Where applicable, the values include root-mean-square (RMS) uncertainties resulting from the data analyses. Principal differences between the separate entries in Table G-3 are in the concentrations of niobium and cobalt. As the footnote in the table indicates, the low values of niobium deduced from the radioactivity data may result from inaccurate estimates of neutron capture cross section values. Cobalt is important not because it activates to a direct classification nuclide but because it gives rise to an activity that is measured directly and to which all other activities are indirectly related in practical classification techniques.³

1.3 Volume Estimates of Light Water Reactor GTCC LLW

Table G-4 summarizes the estimates of volumes of GTCC irradiated hardware wastes that result from normal operation and decommissioning of commercial nuclear power reactors. The estimates in the table are for typical 1000 MWe reactors operated for 40 years with a duty cycle of 75% full power operation. The values will not differ significantly if plant lifetime extensions add 10 or 20 years to the operation.

PWR

Data indicate that PWRs in normal operation generate only fuel assembly skeletons that have any real significance for irradiated hardware GTCC LLW. They also show that only the inconel-718 components of these skeletons will be GTCC, and only Westinghouse assemblies contain substantive quantities of inconel (in the grid support spacers). Table G-4 shows the volumes per Westinghouse assembly, since the present evaluation did not survey industry to ascertain the number of Westinghouse assemblies in use. The total volume of skeletons generated by industry is about 15 m³ per year, based on the data presented in Section 2.3.1.

An assumption that one-third are manufactured by Westinghouse and that the inconel portions are separated for disposal as GTCC yields an estimate of about 1 m³ per year of GTCC LLW. Such an analysis may be moot since the present status of thinking on fuel reconsolidation suggests that the assemblies will remain with the fuel and be a portion of the high-level waste. In such an event, the volume of GTCC LLW from normal operation of PWRs is estimated at zero. As the table indicates, for components from decommissioning, the high values of concentrations show a GTCC classification for the PWR shroud and core barrel and the BWR core shroud, whereas low values are well below this classification. Only the PWR shroud is in a GTCC category in the base case. Values in the table are the volumes per reactor. These estimates of decommissioning PWR GTCC LLW depend upon the actual neutron flux at the shroud and core barrels during reactor operation. The high values assume neutron fluxes at the levels predicted by the computations in Reference 1, whereas the low values use flux values scaled downward by the same ratio as found for BWRs (see the discussion in

Table G-3. Summary of data base for elemental constituents of reactor hardware alloys

MATERIAL	SOURCE	Ni	Co	Nb	Mo	Zr	Sn
<u>STAINLESS STEEL-304</u> <u>BWR uses:</u> CONTROL-ROD BLADES, IN-CORE INSTRUMENTS, FUEL SUPPORT PIECES, FEED-WATER SPARGERS, POISON CURTAINS, CONTROL-ROD GUIDES, CORE SHROUD	Nominal(2) NUREG/CR- 3474(1) Table B1 Table B2	8.92 10.0+,-0.7 8.9+,-0.5 8.9 (#1)	0.080 0.14+,-0.08 0.10+,-0.06 0.120	0.101 0.009+,-0.009 0.014+,-0.011 0.003 (#2)	0.00 0.26+,-0.15 0.39+,-0.22 0.15	0.00 - - -	0.00 - - -
<u>PWR uses:</u> FUEL ASSEMBLY PIECES, SHROUD, CORE BARREL							
<u>ZIRCALOY-4</u> <u>BWR uses:</u> FUEL FLOW CHANNELS <u>PWR uses:</u> FUEL ASSEMBLY PIECES	Nominal(2) Table B1 Table B2	0.002 <0.005 0.021 (#4)	0.001 <0.0027 0.0014	0.012 - 8.4E-6 (#2)	0.00 0.87 (#3) -	98.0 94+,-11 -	1.6 2.7+,-0.6 -
<u>INCONEL-718</u> <u>PWR uses:</u> FUEL ASSEMBLY PIECES	Nominal(2) Table B1 Table B2	52.0 49.7+,-2.5 49.7(#1)	0.47 0.087+,-0.036 0.16	5.55 4.4+,-1.1 1.1(#2)	3.0 2.79+,-0.11 -	0.0 - -	0.0 - -
#1 Value used to normalize scaling factor results.							
#2 Scaling factor results for Nb analyses show values about a factor of three lower than those from elemental analyses. The difference may arise from inaccurate neutron cross section values used in ORIGEN-2 computations (see discussion in text).							
#3 This is a "single-analysis" result.							
#4 Normalizing value results from analyses of concentration data in reference 3 and activation values from reference 4, discussed in the text.							

Table G-4. Estimates of volumes of LWR GTCC waste

TABLE 1.4 ESTIMATES OF VOLUMES OF LWR GTCC WASTE

COMPONENT REMOVED	YEARLY VOLUME OF GTCC WASTE (M-3)	
PWR		
<u>FUEL ASSEMBLY</u>		
SS-304, ZIRCALOY-4	High:	0.0
	Base:	0.0
	Low:	0.0
INCONEL-718	High:	6.4E-4 (#1)
	Base:	6.4E-4
	Low:	6.4E-4
<u>SHROUD</u>	High:	1.6 (#2)
	Base:	1.6
	Low:	0.0
<u>CORE BARREL</u>	High:	3.4 (#2)
	Base:	0.0
	Low:	0.0
BWR		
<u>CONTROL-ROD BLADE</u>	High:	0.0
	Base:	0.0
	Low:	0.0
<u>IN-CORE INSTRUMENTS</u>		
LPRM	High:	0.12 (#3)
	Base:	0.09
	Low:	0.05
IRM	High:	1.6E-3 (#2)
	Base:	1.6E-03
	Low:	1.6E-03
SRM	High:	8.0E-4 (#2)
	Base:	8.0E-04
	Low:	8.0E-04
<u>POISON CURTAINS, FUEL-SUPPORT PIECE, FUEL FLOW CHANNEL</u>	High:	0.0
	Base:	0.0
	Low:	0.0
<u>CORE SHROUD</u>	High:	3.3 (#2)
	Base:	0.0
	Low:	0.0
<p>#1. Inconel-718 is used for grid spacers only in Westinghouse assemblies. The value shown here is per Westinghouse assembly.</p> <p>#2. These components are normally removed only at the end of reactor life. The values given are per reactor.</p> <p>#3. These values assume that packagers mix cold ends with hot ends and average over the volume of items in each package.</p>		

Section 3.2). The base is a log-mean average of the two and presents the best estimate that can be made from the limited data base. The values have high uncertainties.

BWR

Analyses indicate that BWRs in normal operation produce GTCC material only in the form of incore instrumentation. LPRMs are low-volume SS-304 tubes that have generally been combined with other SS-304 components in burial casks for concentration averaging over the combined volume of material. IRM and SRM instrument tubes released at the end of plant life would be considerably more active and be GTCC material. The estimated volumes in the table are yearly total projections for all BWR plants except for the core shroud; the values are per plant. The high estimate assumes the high nuclide concentrations (using only the hot ends) in Table G-1 and that all of the instruments are shipped as GTCC LLW, whereas the low estimate assumes the low concentrations and that some averaging occurs. Base values are linear averages. Only the core shroud has potential for GTCC classification in wastes from BWR decommissioning. Estimates of GTCC volumes from this component have the high value from the calculated estimates in Reference 2 and the low value computed using the measured neutron fluxes at the shroud from Reference 21 (discussed in Section 3.2). Base values assume the base concentrations in Table G-2. The table indicates the BWR core shroud is probably not a candidate for GTCC LLW.

1.4 Recommendations for Efforts to Reduce Uncertainties in Estimated Volumes

The source of GTCC irradiated hardware waste with the largest potential volume is that arising from decommissioning of commercial nuclear power reactors. Estimates of projected volumes of these wastes have the greatest uncertainties, arising from a dearth of data on activities in these components that lie outside the reactor core. Principal causes of the large values are uncertainties in: (a) thermal and epithermal neutron flux values in these regions, (b) neutron capture cross sections--thermal and resonance--for important activations, especially for nickel and niobium, (c) the range of trace elemental constituents in the stainless steel-304 used in these components--cobalt and niobium--and (d) methods for practical waste classification of the large components resulting from decommissioning. Recommendations to decrease the spread in the estimated volumes include:

1. Perform underwater dose rate profiles of the core shroud in the three BWR plants retired from service--Humbolt Bay, Dresden Unit 1, and LaCross. These plants have the fuel removed from the core, and access to the shroud could be made easily by removing the pressure vessel top. The average dose rate of the shroud would be combined with dose to curie factors computed for this simple geometry and with decay- and irradiation-history-corrected scaling factors (from Appendix A) to estimate concentrations of the relevant nuclides. These values would be used to improve the neutron flux estimates for the larger reactors as well as for scaling the measured activities to the larger reactors and longer service life. Similar studies would occur at the retired Rancho Seco PWR on the shroud and core barrel.

2. Evaluate the available neutron cross section data--thermal and resonance--for nickel and niobium activation to ^{59}Ni , ^{63}Ni , and ^{94}Nb to obtain the best activation values. The evaluation would use reactor activation measurements using neutron energy-selective absorbers to separate thermal from epithermal activation and would utilize the DOE National Data Centers to access additional data and evaluations of known data on these cross sections.
3. Perform additional searches for industry data of measured or estimated neutron fluxes in the regions of core shrouds and barrels, particularly for PWRs. This would require additional contacts with the utility industry and with the three principal manufacturers of U.S. PWR plants.

1.5 Conclusions

1. PWRs generate little or no GTCC LLW in normal operational shutdowns; fuel assembly skeletons will probably remain with the spent fuel and go to the high-level waste repository.
2. BWRs, in normal refueling, generate only incore instruments that have GTCC LLW classification; they can be mixed with the higher volume CRBs, where concentration averaging will yield Class C waste.
3. It is uncertain whether PWR shrouds will be GTCC LLW at decommissioning because of uncertainties in neutron flux values at their location outside the core; core barrels will probably not classify as GTCC LLW.
4. BWR core shrouds will likely not be classified as GTCC material at decommissioning.
5. Data are needed from which to gain additional confirmation of activities in core shrouds and barrels.

2. BACKGROUND

2.1 Purpose and Scope of Evaluation

The U.S. Department of Energy is obligated to provide for disposal of high-level radioactive waste and low-level wastes that exceed the radioactivity limits for disposal in low-level surface repositories operated by the state compacts. Disposal of high-level waste, which consists primarily of spent nuclear fuel and concentrated fission product and transuranic activities that result from chemical processing of nuclear fuels, is separate from disposal of the low-level wastes and will probably be in deep repositories. The higher activity low-level wastes will require different facilities for their internment. These wastes come from many sources, and their activity levels and volumes are less known. Medical hospitals and research centers, industry (particularly the nondestructive testing segment), and electric utilities all generate significant volumes of GTCC LLW. To a large extent, levels of activity and volume for this material are unknown since disposal of the wastes has been minimal until the present. It is important to estimate these quantities in order to provide proper input for design of disposal facilities. EG&G Idaho, Inc., is developing a data base from which to project DOE needs for storage and disposal of this class of low-level wastes and desires assistance in providing additional data and assessments of current data in the base.

The purpose of the current study is to augment the EG&G data base by assessing current data and supplying new data on utility generated, low-level wastes that must be accepted by DOE. The emphasis is on nonfuel-bearing activated metal components used in operation of the utility reactors since studies have treated the other sources. Activated hardware results from normal refueling and maintenance operations and from decommissioning at the end of the plant life, currently presumed to occur after 40 years of operation. Current assumptions on the activity levels and volumes resulted from two studies by Battelle Pacific Northwest Laboratories (PNL).^{4,5} The present study will also evaluate assumptions in the current base against the new data and provide high (worst case) and low (best case) estimates of both the activity and volume levels for future GTCC LLW. Sources of new material are utility and industry data on classification and disposal of irradiated hardware as well as data on reactor operations. Only limited data are available from current or past decommissioning activities, and evaluations used data from other sources.

2.2 Definition and Classification of Greater-Than-Class C Wastes

In December 1982, the U.S. Nuclear Regulatory Commission (NRC) published an addition to the Code of Federal Regulations, 10 CFR Part 61, to take effect in December 1983 (Reference 1). The code defined three categories of low-level radioactive waste suitable for shallow-land disposal and some packaging regulations for the waste. Definition of the wastes was in terms of specific radionuclide concentrations, and the rules established Classes A, B, and C wastes for commercial disposal, Class C allowing the highest concentrations of these nuclides. Similarly, 10 CFR Part 60,⁶ defines high-level radioactive wastes. Wastes having activity concentrations higher than the limits in 10 CFR Part 61 for Class C material but falling outside the definition of high-level waste in 10 CFR Part 60 are designated as greater-than-Class C (GTCC) low-level wastes. DOE has been assigned the responsibility for disposal of this material.⁷ The NRC classification of wastes separates radionuclides into two categories, long-lived (Table 1) and short-lived (Table 2) nuclides. Nuclides

of relevance to irradiated hardware in Table 1 are ^{14}C , ^{59}Ni , ^{94}Nb , ^{99}Tc , and long-lived transuranics, where the Class C maximum permissible activity concentrations are, respectively, 80, 220, 3, and 0.2 Ci/m^3 and 100 nCi/g . Similarly, the relevant nuclide in Table 2 is ^{63}Ni , where the upper limit for Class C is 7000 Ci/m^3 . In addition to individual concentration limits, the rules define a "sum of the fractions" limit for the nuclides in each table where the sum of the fractional limits for each nuclide (i.e., the concentration divided by the concentration limit) may not exceed unity (sum of the fractions rule). Each package or component of low-level radioactive waste must be classified as either Class A, B, C, or GTCC material before shipping for disposal. An additional rule states that the shipping manifest must list concentrations of specific nuclides even though their activities are too low to influence the classification of the material. These nuclides include most of those listed above as well as ^{129}I , a fission product.

The radionuclides listed above as relevant in irradiated metal do not emit radiation that can be detected in the presence of other activities in the components, principally ^{60}Co , whose allowed concentration is unlimited in Class C waste because of its short (five-year) half-life. Their concentrations can only be measured directly through destructive radiochemical analyses that are impractical for each item. Unless the component is analyzed as a whole: (a) representative sampling is nearly impossible, (b) radiation levels are too high for analyses of more than very small samples, and (c) the cost is prohibitive. The NRC has provided guidance⁸ suggesting two alternative methods for classification of activated hardware: use of nuclide scaling factors determined through measurements and through activation analysis. A scaling factor is a ratio of activity concentrations of the desired nuclide (scaled nuclide) to a scaling nuclide (^{60}Co for hardware). Both methods rely on determining experimentally the average radiation dose rate of each package and applying a dose to curie factor to convert average dose rate to average concentration of the dose contributors, nearly always ^{60}Co .

The two methods differ in the determination of scaling factors to ^{60}Co . The measurement technique radiochemically analyzes small samples of the activated metal to measure the scaling factors for ^{14}C , ^{59}Ni , ^{63}Ni , ^{94}Nb , ^{99}Tc and TRU (TRU comes from surface contamination). Small samples are justified in this case since scaling factors require only activity ratios. Activation analysis uses assumed elemental constituent abundances of the component alloy, a data base of thermal and epithermal neutron cross sections, an estimated thermal and epithermal neutron flux, and the radiation history of the component to compute the scaling factors. (Activation analysis calculations also yield estimates of activity concentrations that are sometimes used where dose rate measurements are not available.) Activation analysis calculations usually use the two computer codes ORIGEN2⁹ and ANISN.¹⁰

The NRC also allows the packaging several components in the same disposal cask and averaging the nuclide concentrations over the volume of similar components in the container (Reference 8). Rules for the averaging require that the components be similar (i.e., same material and same geometry) and that the average concentration in each component not differ from that in the others by more than a factor of ten (factor of ten rule).

2.3 Sources of Possible GTCC Hardware

There are two sources of irradiated components that could have activity concentrations that exceed the Class C limits, those that are changed in normal reactor refueling or maintenance activities, and those that are large structures internal to the reactor pressure vessel and will come from major reconstruction or decommissioning. Some of the components associated with refueling, such as PWR fuel skeletons, may remain with the fuel and thus be shipped with the high-level waste.

2.3.1 Components from Routine Refueling and Maintenance

Table G-5 lists the principal hardware items of irradiated hardware generated in refueling and maintenance operations that could result in GTCC LLW. As can be seen from the listing, the main source is the BWR.

PWR Components. The large potential source from PWRs is the fuel assembly skeleton, shown pictorially in Figure G-1. There are five manufacturers of the skeletons used by the utilities Westinghouse, Exxon, General Electric, Combustion Engineering, and Babcock & Wilcox. Four of the manufacturers produce skeletons that are fundamentally the same from a waste classification standpoint, whereas the Westinghouse assembly differs by having a substantial amount of inconel-718 as grid support (or grid spacers); the others use zircaloy-4 for that component. Other materials in the assembly are SS-304 and zircaloy-4 as shown in Table G-5. The table also lists the approximate mass and volume for each of the major components in a single assembly. A typical large PWR has about 300 fuel assemblies in the reactor core during operation. Approximately a third of these are discharged in a refueling operation that occurs every 12–18 months. Consideration of fuel reconsolidation (removal of the fuel rods from spent assemblies) has received study recently¹¹⁻¹⁶ in efforts to gain more available space in the utility spent fuel pools. If reconsolidation is done, the skeletons are potential GTCC LLW; if not, the skeletons will probably remain with the spent fuel to go to the high-level waste repositories.

Other PWR incore components are thimble plug assemblies and burnable poison assemblies that have relatively low volumes. These components spend most of their incore life in regions of relatively low flux and do not have serious potential for being GTCC LLW.

BWR Components. Table G-5 also lists six components of irradiated hardware that are removed periodically from BWRs during outages. These pieces are considered possible GTCC LLW.

Control Rod Blades (CRBs). The CRB is the largest volume source of irradiated hardware discharged in refueling operations. Table G-5 lists the typical volume and mass of each blade, and Figure G-2 shows a drawing of a typical CRB. The shape is that of a cruciform, and the materials are SS-304, except for boron carbide pellets contained inside thin-wall tubes in the blades and bearing rollers of stellite components). Because the cobalt activates to such a high activity level, the rollers are usually separated from the spent blades while they are in the spent fuel pool. The boron carbide adds little activity to the component except ^3H , but even this is at sufficiently low concentration that it doesn't influence the classification. BWR cores have 150–200 CRBs in place at any given time, and approximately 15–30 blades are discharged during each refueling cycle that occurs every 12–18 months.

Table G-5. Summary of relevant characteristics of activated hardware generated in normal reactor operations

COMPONENT	MANUFACTURER	MATERIAL	VOLUME (M-3)	MASS (Kg)
PWR				
FUEL ASSEMBLIES TOP AND BOTTOM NOZZELS (TIE PLATES)	W,E,G,C,B	SS-304	1.3E-03	11.0
GUIDE TUBES	W,E,G,C,B	ZIRCALOY-4	1.2E-03	9.0
INSTRUMENT TUBES				
HOLD-DOWN SPRINGS	W,E,G,C,B	INCONEL-718	2.2E-04	1.8
SPACER GRIDS	W E,G,C,B	INCONEL-718 ZIRCALOY-4	6.4E-04	5.3
BWR				
CONTROL-ROD BLADES (CRUCIFIX)		SS-304 (B-4C)	2.7E-02	210.0
IN-CORE INSTRUMENTS		SS-304	2.0E-04	
POISON CURTAINS		SS-304	2.1E-02	160.0
SPARGER NOZZLES		SS-304	7.3E-02	
FUEL-SUPPORT PIECE		SS-304	3.5E-03	27.0
FUEL FLOW CHANNELS		ZIRCALOY-4	5.3E-03	42.0
W: Westinghouse, E: Exxon Nuclear, G: General Electric Co. C: Combustion Enginnering, B: Babcock and Wilcox

Westinghouse 17 x 17 PWR fuel assembly

Figure G-1. Sketch of typical PWR fuel assembly skeleton.

Figure G-2. Sketch of typical BWR control rod blade.

Incore Instruments. A BWR uses three forms of incore instruments that are potential GTCC LLW when discharged: local power range monitors (LPRM), intermediate range monitors (IRM), and source range monitors (SRM). The construction is a thin-wall tube of SS-304 about 13 m long containing typically four fission chambers with mg quantities of different isotopes of uranium. The fission chambers are easily removed from used instruments when necessary, although the chambers do not influence classification. A typical BWR core contains about 100 LPRMs, 8 IRMs, and 4 SRMs. The geometries, weights, and masses of the three, listed in Table G-5, are about the same, although the fission chambers are different. About 10–30 LPRMs are discharged each refueling cycle (12–18 months). IRM and SRM tubes are intended to remain the life of the core but are occasionally discharged for various reasons. Each instrument consists of a 3.7-m hot end that is in the core during service and a 9.4-m cold end that sees few neutrons.

Poison Curtains. Poison curtains are flat plates of borated SS-304 that are located near the core during the reactor's first cycle to adjust the flux distribution. They are usually removed after that cycle since partially burned fuel can fulfill their function. Table G-5 lists the volume and mass of typical poison curtains. Although the curtains see a relatively high flux, they remain in place for only one cycle.

Sparger Nozzles, Fuel Support Pieces, Tie Plates. The feedwater sparger nozzle is a large piece of SS-304 that is changed in maintenance when it shows evidence of stress corrosion cracking or is removed at decommissioning. It sits in a relatively low neutron flux while in core. The fuel support pieces and tie plates are relatively low-volume items of SS-304 that are near the top of the core during use.

Fuel Flow Channels. These components are hollow, square cross section pieces of zircaloy-4 alloy that surround each fuel element. They contain small flow channels for cooling the fuel. There are approximately 500–840 channels in a BWR core, one for each fuel assembly. They remain with the assembly and are discharged with the fuel but are separated after discharge. Thus, they remain in core for approximately three refueling cycles, or 36–54 months.

2.3.2 Components from Decommissioning

Particularly important candidates for GTCC LLW remain in the reactor pressure vessel during the life of the plant. Table G-6 lists these components and gives their characteristics. Shrouds and core barrels are rather massive systems and could contribute significantly to the volume of GTCC LLW if they are so classified. They surround the core inside the pressure vessel. In a PWR, the shroud lies just outside the region of fuel, and the core barrel lies about 19 cm outside the shroud. The neutron flux falls off relatively fast in these regions. The BWR core shroud lies about 20 cm beyond the fuel, and, here also, the neutron flux falls off rapidly in this region.

Table G-6. Characteristics of relevant activated hardware from decommissioning

COMPONENT	MATERIAL	VOLUME (M-3)	MASS (Tonne)
<u>PWR</u>			
SHROUD	SS-304	1.6	12.4
CORE BARREL	SS-304	3.4	26.5
<u>BWR</u>			
CORE SHROUD	SS-304	3.5	27.0

2.4 Status of Knowledge Base for Irradiated Hardware

The current DOE knowledge base for utility-generated waste with potential for classification as GTCC is relatively sparse. A survey taken of utilities turned up only limited information, primarily because the plants had not classified many of the wastes and did not know the activity concentrations. No measurement data are identified for activity levels or radiation dose rates of core shrouds or barrels. The data base relies heavily on the data and computational results reported in two NUREG documents issued in 1980 (References 4 and 5) on activities and volumes of irradiated hardware projected to be GTCC LLW. These estimates, particularly for decommissioning waste, use the results of neutron flux projections for areas outside the core for input into the activation calculations. They also use the activation cross sections in ORIGEN2 (Reference 9) and the nominal elemental abundances in the alloys. Activation cross sections for three of the important activities, ^{59}Ni , ^{63}Ni , and ^{94}Nb , are not well known¹⁷, either in the thermal or epithermal regions. No data are available on cross sections for activation of the individual isotopes of nickel. Only data on activation of a naturally occurring nickel element are in the literature, and, since the element contains five stable isotopes, estimates of individual cross sections have considerable uncertainty. For niobium, even though it has only a single stable isotope, the thermal and resonance energy activation cross sections are not reported, and the scatter in the data for total cross section gives considerable uncertainties in the accuracy of the estimated values.

Data from power reactors that have retired from service (Humbolt Bay, Dresden Unit 1, Rancho Seco, LaCross, and Shippingport) are not available for the core shrouds and barrels. No decommissioning demolition or activity surveys have been made at the first four plants,^a and the Shippingport reactor pressure vessel was buried in toto after being filled with concrete; no radiation surveys were made of the reactor internals.¹⁸ Other uncertainty concerns in projecting GTCC needs for disposal of irradiated hardware components include future rulings for averaging concentrations over packaged components. If the current NRC interpretation (discussed in Section 2.2) is allowed by the regional compact sites, the volume of GTCC LLW will be considerably lower than that from a less favorable set of rules. The compacts have not seriously addressed this issue yet, and only speculation can be used for estimates. This report assumes the current regulations will continue.

a. Personal communication with Paul Roth, Commonwealth Edison Co., December 1990, and personal communication with Seymour Rafferty, Dairyland Power Cooperative, December 1990.

3. COMPILATION OF DATA OF HARDWARE RADIOACTIVITY AND VOLUMES

3.1 Operations-Generated Components

Utilities have removed and shipped for disposal a substantial amount of irradiated hardware items from their spent fuel pools. These items have primarily been discharged from the reactor during refueling operations. Characterization and classification of the components was necessary before shipment, and the data are important to estimates of future GTCC materials generated in these activities. Many of the methods for classification used measurements of radionuclide scaling factors, stable element abundances in the component alloys, dose to curie conversion factors for ^{60}Co , or average dose rates for the components. These data are very important for adding to the knowledge data base and for comparisons to the computational estimates that form the basis for many of the current projections. Appendices A, B, and C include the important data from these measurements.

3.1.1 Components from Boiling Water Reactors

Table G-7 summarizes the data on operations-generated irradiated hardware from BWRs. The values in the table are summary results of the data in the three appendices. The first column lists the eight components considered and the approximate volume of each. The second column labels the maximum, minimum, and average dose rate and radionuclide concentration values for the data included in the summary. The third column gives the values for the average contact dose rate, each averaged from dose rate profiles taken along the component while it was in the fuel pool. The average is a linear average of the many measurements for that type of component. Minimum and maximum values are those occurring in the reported measurements. Data labeled dose to curie factor show the average measured dose to curie factor $[(\text{Ci}/\text{m}^3)/(\text{R}/\text{hr})]$ for ^{60}Co (converting a dose rate to a nuclide concentration).

Reference 3 discusses the method for measuring this quantity that can also be computed.¹⁹ Computed and measured factors are in close agreement. The uncertainties in the factors are RMS values of the dispersion in the experimental results. The fifth column gives the concentration of ^{60}Co from converting the dose rates to curie concentrations through the dose to curie factor. The last four columns give the concentrations of ^{63}Ni , ^{94}Nb , ^{99}Tc , and TRU, respectively. These values result from multiplying the ^{60}Co concentration in column 5 by the appropriate nuclide scaling factor, whose values derive from Appendix A. TRU, expressed in the required units of nCi/g rather than Ci/m^3 , is not from bulk activation of the metal but from surface deposition of material on the component during its tenure in the reactor pressure vessel. Unless the tightly bonded surface crud is removed, it becomes a portion of the component for classification and disposal. TRU is the more restrictive of the surface contaminants. Data in Table G-7 show that the control rod blades, incore instruments and poison curtains contain the higher concentrations of all activities.

Table G-7. Average radionuclide concentrations for activated reactor components changed out in normal operations

COMPONENT		AVERAGE CONTACT DOSE RATE (kR/Hr)	DOSE-TO- CURIE FACTOR (Co-60)	AVERAGE CONCENTRATION				of TRU (nCi/gm)
				of Co-60 (Ci/M-3)	of Ni-63 (Ci/M-3)	of Nb-94 (Ci/M-3)	of Tc-99 (Ci/M-3)	
CRB VOLUME: .0266 M-3	MIN:	3.3	3.3	1.1E+04	6.9E+02	5.6E-03	9.6E-04	0.32
	MAX:	15.6	+1.3	5.1E+04	3.2E+03	2.7E-02	4.5E-03	1.52
	AVG:	8.2	-0.9	2.7E+04	1.7E+03	1.4E-02	2.4E-03	0.80
LPRM VOLUME 2E-4 M-3	MIN:	1.1	55.1	6.0E+04	3.8E+03	3.1E-02	5.3E-03	1.78
	MAX:	19.8	+11.1	1.1E+06	6.9E+04	5.6E-01	9.6E-02	32.30
	AVG:	8.5	-9.2	4.7E+05	3.0E+04	2.4E-01	4.1E-02	13.92
IRM VOLUME 2E-4 M-3	MIN:	8.1	55.1	4.4E+05	2.8E+04	2.3E-01	3.9E-02	13.15
	MAX:	20.6	+11.1	1.1E+06	7.2E+04	5.9E-01	1.0E-01	33.61
	AVG:	12.1	-9.2	6.6E+05	4.2E+04	3.4E-01	5.9E-02	19.66
SRM VOLUME 2E-4 M-3	MIN:	10.7	55.1	5.9E+05	3.7E+04	3.1E-01	5.2E-02	17.49
	MAX:	11.6	+11.1	6.4E+05	4.0E+04	3.3E-01	5.7E-02	18.97
	AVG:	11.2	-9.2	6.2E+05	3.9E+04	3.2E-01	5.4E-02	18.24
POISON CURTAINS VOLUME: 2.1E-2 M-3	MIN:	0.99	40.0	4.0E+04	2.5E+03	2.0E-02	3.5E-03	1.17
	MAX:	1.86	+14.9	7.4E+04	4.7E+03	3.8E-02	6.6E-03	2.20
	AVG:	1.43	-14.9	5.7E+04	3.6E+03	3.0E-02	5.0E-03	1.69
SPARGER NOZZLE VOLUME: 7.3E-2 M-3	MIN:	0.01	6.7	6.7E+01	4.2E+00	3.5E-05	5.9E-06	0.002
	MAX:	0.06		4.0E+02	2.5E+01	2.1E-04	3.5E-05	0.012
	AVG:	0.04		2.7E+02	1.7E+01	1.4E-04	2.4E-05	0.008
FUEL-SUPPOR PIECE VOLUME: 3.5E-3 M-3	MIN:	N.A.	6.7	N.A.	N.A.	N.A.	N.A.	N.A.
	MAX:	N.A.		N.A.	N.A.	N.A.	N.A.	N.A.
	AVG:	0.26		1.7E+03	1.1E+02	9.0E-04	1.5E-04	0.052
FUEL FLOW CHANNELS VOLUME: 5.3E-3 M-3	MIN:	0.023	4.3	9.9E+01	1.2E+00	4.3E-04	N.A.	0.866
	MAX:	0.378	+6.5	1.6E+03	2.0E+01	7.0E-03	N.A.	14.239
	AVG:	0.131	-2.9	5.6E+02	6.8E+00	2.4E-03	N.A.	4.942

3.1.2 Pressurized Water Reactors

Components removed in refueling and maintenance operations in PWRs relevant to GTCC consideration are limited to fuel skeletons. Table G-8 summarizes the measurement data from PWR fuel assembly skeletons given in the appendices. Industry uses five types of skeletons made by Westinghouse, Exxon, General Electric, Combustion Engineering, and Babcock & Wilcox. The table lists characteristics of assemblies made by four of these manufacturers and gives the approximate volumes and masses of the main components in each. Assemblies made by Westinghouse differ substantively from the others in having grid spacers of inconel-718. The similar Babcock & Wilcox assemblies, not listed, also have zircaloy-4 grid spacers. Volumes and masses for the components in each of the five types of skeletons are similar. There is not so much data for PWR skeletons as for the BWR hardware, and Table G-8 lists the reference for each set of measurements. Dose to curie values are given only for measurements of Westinghouse skeletons reported in Reference 11. The other measurements took small samples of material from each portion and assumed that the concentrations were representative of the entire item, and classification measurements did not include determining the average dose rate of components. The last three columns in the table give a summary of the reported concentrations of ^{60}Co , ^{63}Ni , and ^{94}Nb . In all cases, scaling of ^{63}Ni and ^{94}Nb concentrations from ^{60}Co used measured scaling factors from radiochemical analyses.

Results from the measurements by the two organizations are significantly different where a comparison can be made for the Westinghouse skeletons. Factors of over 100 difference between some reported concentrations are hard to explain. The data from Table G-8, reference 20, are an average of 36 assemblies, all using the same set of scaling factors; the other data in the table are from smaller numbers of samplings. The skeletons analyzed for the data in the table experienced similar total neutron fluences ($\approx 35,000 \text{ MW} \cdot \text{d/MTU}$) and should show comparable concentrations for the same alloy. Thus, the bottom of the table shows log-mean averages and dispersions computed for each material (calculations of log-mean average and dispersion values are described in Appendix A). Values of dispersions (average factors of deviation from the average) are large, especially for nickel and niobium activities.

3.2 Decommissioning Generated Components

Almost no data are available on activity concentrations of reactor internals for either operating plants or plants that have been shut down at the ends of their lifetimes. Because of the very short operating times, data from decommissioning of the Elk River and Pathfinder reactors have limited applicability to large scale light water reactors with 40-year operating experience, and, as was previously discussed, no measurements data can be found for four other closed reactors. In estimating the activity concentrations and volumes of the reactor internal components, reliance must be on available neutron flux, activation cross section, alloy elemental abundance data, and on computations that use these estimates.

3.2.1 Boiling Water Reactors

PNL has computed and reported (References 2 and 4) expected radial neutron fluxes--thermal, epithermal, and fast--in regions beyond the core in large (1000 MWe) BWRs. The reports also give the axial flux along the core center. Using these values, the two references predict the activity concentrations expected after a 40-year life of the BWR as far beyond the core as the biological

Table G-8. Measured radionuclide concentrations in fuel assembly skeleton

COMPONENT	MASS Kg	VOLUME M-3	REFER- ENCE	DOSE- RATE KR/Hr	CONCENTRATION Ci/M-3		
					Co-60	Ni-63	Nb-94
WESTINGHOUSE PWR FUEL ASSEMBLY (Average Burnup 33,000 MWD/MTU)							
Upper & Lower Tie Plate — SS-304	8.83	1.1E-03	20	1.4	9.4E+03	5.9E+02	4.9E-03
Spacer Grids Inconel-718	5.2	6.4E-04	20	2.3	9.4E+04	2.0E+04	2.0E-02
			27		1.6E+04	4.1E+03	2.5E+00
			23		7.2E+05	5.4E+05	6.6E+02
Guide Tubes Zircaloy-4	7.8	1.2E-03	20	0.19	4.0E+04	3.1E+04	1.2E+02
					9.9E+03	1.2E+02	4.3E-02
EXXON PWR FUEL ASSEMBLY (Average Burnup 33,000 MWD/MTU)							
Upper & Lower Tie Plate — SS-304	9.67	1.2E-03	20		1.1E+03	3.3E+02	5.0E-02
Hold-down Springs Inconel-78	1.0	1.2E-04	20				
Spacer Grid, Guide & Instrument Tubes Zircaloy-4	12.7	1.9E-03	20		2.5E+02	3.7E+01	1.3E-02
GENERAL ELECTRIC PWR FUEL ASSEMBLY (Average Burnup 27,500 MWD/MTU)							
Upper & Lower Tie Plate — SS-304	9	1.2E-03	27		3.9E+04	5.1E+03	1.8E-01
Spacer Grid, Guide & Instrument Tubes Zircaloy-4	12	1.8E-03	27		1.6E+03	2.0E+02	3.3E-02
COMBUSTION ENGINEERING PWR FUEL ASSEMBLY (Average Burnup 41,800 MWD/MTU)							
Upper & Lower Tie Plate — SS-304	9	1.2E-03	27		6.3E+04	1.3E+04	3.4E-02
Spacer Grid, Guide & Instrument Tubes Zircaloy-4	12	1.8E-03	27		1.1E+03	5.5E+01	1.9E-01
Hold-down Springs Inconel-78	0.1	1.2E-05	27		3.6E+05	3.7E+05	2.1E+02
SUMMARY	SS-304			AVG:	1.9E+04	3.0E+03	3.1E-02
				DISP:	6.1	6.3	3.7
	ZIRCALOY-4			AVG:	2.8E+03	2.7E+02	2.1E-01
				DISP:	7.2	15.2	39.4
	INCONEL-718			AVG:	1.6E+05	9.3E+04	7.0E+01
				DISP:	7.7	15.2	19.2

shield. The computations show that only the BWR core shroud is important for consideration as GTCC LLW. In 1980 General Electric Co. at the Brown's Ferry Unit 3 plant instrumented the region between the core shroud and the pressure vessel with neutron flux monitors to determine thermal, epithermal, and fast neutron flux in this region.^{20,b} Oxides of ^{235}U , ^{238}U , ^{237}Np and ^{232}Th , salts of Fe, Ni, Cu, and Ti, and AlCo wires served as the neutron dosimeters. This combination (some capsules shielded with cadmium) gave redundant results for the entire neutron spectrum for verification purposes. Three radial locations were 4 and 30 cm from the shroud and 7 cm from the pressure vessel and at azimuthal angles of 3° and 45° around the core. The combination of locations showed radial axial and azimuthal flux variations and absolute values within this region of the vessel. Figure G-3 gives a cross-sectional view of the reactor pressure vessel and shows the locations of measurements relevant to flux values at the shroud and pressure vessel. Flux values at 45° should be slightly higher than those at 3° because the higher angle is at the corner of the reactor fuel array.

Measurements were taken at the core midplane and the core top. The values reported are substantially lower than those estimated in Reference 2. Table G-9 presents published values of neutron fluxes at the shrouds, calculated and measured, for both PWRs and BWRs. The table includes results for thermal, epithermal, and fast neutron fluxes from the BWR measurements taken at the location 4 cm from the shroud at both the midplane and core top. Computed estimates of flux in the table are from References 2 and 4, from ORIGEN2 and ANISN. The differences of nearly a factor of 100 between computed and measured values at the BWR core midplane and near 10 at the core top are very significant in estimating the activities and classification of the shrouds at decommissioning.

Other BWR reactor internal components close to the reactor core are the upper and lower fuel support pieces and feedwater sparger nozzles. Data presented in Section 3.1.1 included activities in feedwater spargers that had been removed after several years of use. The measured values indicated that the sparger activities, even at the end of the reactor life, would lie well below Class C limits. The limited data on the fuel support pieces and the neutron flux values estimated at their locations near the core indicate that these items also would not be candidates for GTCC LLW.

3.2.2 Pressurized Water Reactors

The two important components in PWRs that could qualify for GTCC classification are the shroud and the core barrel. Table G-9 includes the computed estimates of flux values at the location of these components in a large (1000 MWe) PWR. If the neutron fluxes are this high, the PWR shroud would constitute GTCC LLW. There are no identified measurement data similar to those made by GE for the BWR to compare to the calculations.

3.3.3 Summary

More measurement data are desirable from which to estimate the activities expected in the light water reactor (LWR) shrouds at the time of decommissioning, especially for PWRs where there are no available data. Radiation levels from these components in the shut down reactors at Humbolt Bay, Dresden, Rancho Seco and LaCross could be used to give valuable data.

b. Personal communication with Dr. Jerry Martin, General Electric Co., December 1990.

Reactor vessel cross section

Figure G-3. Cross section of BWR core showing positions of neutron flux measurements between the shroud and the pressure vessel.

Table G-9. Estimates of neutron flux at structural components

NEUTRON ENERGY RANGE		AVERAGE NEUTRON FLUX (n/cm-2/sec)	
<u>PWR</u>	<u>(#1)</u>	<u>SHROUD</u>	<u>CORE BARREL</u>
Thermal		1.3E+13	1.4E+12
Epithermal		3.5E+13	3.9E+12
Fast (>1 MeV)		7.0E+12	7.7E+11
Computed values from Reference 28; Computations used ORIGEN-2 and ANISN and are the flux at the core mid-plane			
<u>BWR CORE SHROUD</u>	<u>COMPUTED(#1)</u>	<u>MEASURED(#2)</u>	
		<u>@3 DEGREES</u>	<u>@45 DEGREES</u>
<u>CORE MID PLANE</u>			
Thermal	2.5E+13	3.0E+11	5.8E+11
Epithermal	3.0E+12	2.1E+11	3.8E+11
Fast (>1 MeV)	1.0E+12	7.6E+10	1.4E+11
<u>CORE TOP</u>			
Thermal	~6E+12	1.8E+11	
Epithermal	~5E+11	1.3E+11	
Fast (>1 MeV)	~1.5E+11	5.0E+10	
#1. Computed values from References 1 and 7 using ORIGEN2 and ANISN			
#2. Measured values taken from Reference 6.			

Table G-10 shows the results of estimates of activities in LWR internals at the time of decommissioning, 40-year lifetime with a full power factor of 75% (30 full power years). The table shows high, base, and low estimates of concentrations for the four radionuclides important for waste classification: ^{59}Ni , ^{63}Ni , ^{94}Nb , and ^{99}Tc . All values assume a 40-year lifetime for the reactor and a 75% duty cycle. The high values assume neutron fluxes at the level reported in Reference 2, and the low values assume values from Reference 21. The table applies the measured-to-computed flux ratios from the BWR data in Table G-9 to obtain the low values for PWRs. The ratio may not be valid unless the reasons for the difference between estimated and measured flux values outside the core in BWRs also apply to PWRs. Base values are a log-mean average of the high and low values.

The values listed for ^{59}Ni include an additional factor. ORIGEN2 computations and some measurements yielded a ratio for the two nickel activities, ^{63}Ni to ^{59}Ni , of about 150, whereas other measurements (References 3 and 14) reported the ratio as about 50. Two factors contribute to uncertainty in the actual ratio: (a) as previously discussed in Section 2.4, measured activation cross sections for ^{58}Ni and ^{62}Ni are not reported, and the computational codes assume a division of the total cross section for elemental nickel between its isotopes, and (b) no primary analytical standards are available for calibrating instrumentation in quantitative measurements of ^{59}Ni , and the laboratories rely on secondary standards. Recent measurements using a ^{59}Ni standard calibrated by the National Institute of Science and Technology (NIST)^c suggest that a ratio of about 60 may be appropriate. The high values for ^{59}Ni in Table G-10 use a ratio of 50 for computing the concentration, and the low values use a ratio of 150.

c. Private communication with F. Shima, National Institute of Science and Technology, August 1989.

Table G-10. Summary of high and low radionuclide estimates for potential GTCC-classified reactor internal components

COMPONENT	RANGE	RADIONUCLIDE CONCENTRATION (Ci/m-3)			
		Ni-59	Ni-63	Nb-94	Tc-99
<u>PWR</u>					
SHROUD	HIGH: (1)	1870	93000	2.1	0.66
	BASE: (2)	140	12000	0.16	2.8E-02
	LOW: (3)	10	1500	0.012	1.2E-03
CORE BARREL	HIGH: (1)	240	12600	0.15	0.03
	BASE: (2)	19	1600	0.012	3.9E-03
	LOW: (3)	1.4	200	9.2E-04	5.4E-04
<u>BWR</u>					
CORE SHROUD	HIGH: (1)	600	30000	0.34	8.1E-03
	BASE: (2)	29	3900	0.026	1.1E-03
	LOW: (3)	1.3	500	0.002	1.4E-04

1. Flux and elemental constituent estimates from Reference 1, Ni-59 to Ni-63 ratio from references 5,16.

2. Log-mean average of high and low values.

3. Flux estimate from Reference 6; Elemental constituent estimates from analysis of all available data (TABLE 1.2); Ni-59 to Ni-63 ratio from References 1 and 25.

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APPENDIX A

COMPILATION OF MEASURED RADIONUCLIDE SCALING FACTORS FOR IRRADIATED HARDWARE

APPENDIX A: COMPILATION OF MEASURED RADIONUCLIDE SCALING FACTORS FOR IRRADIATED HARDWARE

For the past several years, utilities have shipped for burial in commercial low-level waste sites irradiated reactor components generated in routine plant maintenance or refueling operations. The classification of many of these components used radionuclide scaling factors (activity ratios of specific non-gamma ray-emitting nuclides to ^{60}Co) resulting from radiochemical analysis of small samples of the component. The measured scaling factors are important data in assessing the activity levels in future irradiated components, both those acquired in normal plant operation and those resulting from future decommissioning. Three metals are commonly used for reactor internals: stainless steel-304 (SS-304), zircaloy-4 and inconel-318 (or 350). Even though ^{60}Co is the preeminent activity in each of these materials, dominating the dose rate and gamma ray spectrum, cobalt is, for the most part, an uncontrolled trace element in the metals. Hence, the ratio of the other and more important activities to the easily observed ^{60}Co can vary widely in different specimens of the same alloy.

Scaling to this activity, pragmatically used for classifying the components, can lead to results with large uncertainties. Furthermore, levels of other important constituents in these same alloys, niobium in SS-304 and zirconium-4, and nickel in zirconium-4, are also present in trace amounts, mostly uncontrolled, compounding the uncertainties. Compilation and analysis of the available scaling factor data, used in conjunction with other analyses, assist both in understanding the limitations in the use of these factors and in improving the accuracy of the analyses of irradiated hardware. Analyses of the data, summarized in the following sections, show that measured scaling factors have demonstrated reproducibility better than a factor of 5, well within the NRC limits of a factor of 10,¹ and can be used effectively with other data to classify radioactive components. Samples on which the analyses were made were grindings or cuttings from the component. An underwater sampling tool sampled both the bulk material and the surface contamination of many of the components. In other cases, samples were cuttings obtained with hot cell manipulators and cutting tools. The uncertainties reported for the analytical results were computed primarily from radioactive counting statistics and chemical yield determinations and do not reflect any potential biases from the sampling method. A typical magnitude for reported uncertainties was 20%, much smaller than the spread in the results from different samplings of the same alloy, dominantly caused by variations in the constituent concentrations. Radioanalytical procedures were methods adapted or developed by industry for analyzing utility radioactive wastes, maintained under inspected and audited Quality Control and Quality Assurance programs. Robertson et al.² reports some typical radioanalytic procedures used in the analyses.

Components of Stainless Steel-304

The bulk of the available data on measured scaling factors for SS-304 comes from analyses of BWR components, although there is a pair of data points from PWR fuel assembly end fittings. Nonetheless, the data should be equally applicable to PWR components since the effects of the different neutron spectrum in PWRs present only minor differences (well within the experimental tolerances of the measurements). Table G-11 gives a list of all the measured scaling factors that could be found in the literature survey. The table organizes them by reactor component. The second column in the table cites the reference for the data, listed in Section 4.0. The next five columns give the measured activities of the given five nuclides relative to ^{60}Co . The first four are activation products within the metal itself, whereas the TRU represents the transuranic material contained in

Table G-11. Measured activity concentration ratios in activated components of stainless steel (SS-304)

COMPONENT	REF.	MEASURED ACTIVITY RATIOS (SCALING FACTORS)									
		ACTIVITIES TO Co-60					ACTIVITIES TO Ni-63				
		C-14	Ni-63	Nb-94	Tc-99	TRU	C-14	Nb-94	Tc-99	TRU	
CRB	28	2.1E-07	1.0E-01	2.6E-06		2.0E-06	2.1E-06	2.6E-05		1.8E-05	
	29	1.2E-06	1.1E-01	5.8E-07		1.3E-06	1.1E-05	5.3E-06		2.0E-05	
	29	5.3E-07	6.6E-02	3.2E-07		5.6E-08	8.0E-06	4.8E-06		9.2E-07	
	30	3.8E-06	6.1E-02	<2E-7		1.2E-08	6.2E-05	1.7E-05		1.4E-07	
	31	2.2E-05	8.3E-02	1.4E-06		8.1E-08	2.7E-04	8.1E-06		8.1E-07	
	31	8.4E-04	1.0E-01	8.1E-07			8.4E-03				
	32	2.5E-05	1.9E-01	<2E-7			1.3E-04				
	32	6.1E-06	1.2E-01	4.8E-06		2.4E-08	5.1E-05	4.0E-05		2.0E-07	
	32	4.2E-06	1.0E-01	9.1E-07		1.9E-06	4.2E-05	9.0E-06		1.9E-05	
	34	3.4E-07	4.1E-03	<1E-9		1.6E-08	8.3E-05			3.9E-06	
	35	1.4E-06	6.0E-02	1.3E-05		8.3E-09	2.3E-05	2.2E-04		1.4E-07	
	36	4.8E-06	1.0E-01	<1E-6		1.6E-07	4.8E-05			1.6E-06	
	37	5.5E-05	4.3E-02	<5E-7		1.6E-06	1.3E-03			3.7E-05	
	37	7.7E-08	7.6E-02	<2E-7		2.0E-06	1.0E-06			2.6E-05	
	37	5.4E-06	1.8E-02	<2E-6		5.0E-07	3.0E-04			2.8E-05	
	38	1.1E-05	9.6E-02	<3E-7			1.1E-04				
	33	5.6E-06	6.0E-02	<2E-7		1.3E-05	9.3E-05			2.2E-04	
	33	1.0E-05	6.9E-02	<2E-7		2.4E-05	1.4E-04			3.5E-04	
	33	9.1E-06	5.4E-02	<1E-7		5.1E-06	1.7E-04			9.5E-05	
	33	8.6E-06	4.9E-02	<1E-7		1.7E-06	1.8E-04			3.5E-05	
	39	1.2E-07	8.2E-02	1.5E-06		4.7E-10	1.5E-06	1.8E-05		5.7E-09	
	40	8.8E-05	7.2E-02	2.0E-06	2.0E-07		1.2E-03	2.8E-05	2.8E-06		
	41	1.1E-05	9.6E-02			<3E-6	1.1E-04				
	41	9.8E-04	8.4E-02				1.2E-02				

Table G-11. (continued)

COMPONENT	REF.	MEASURED ACTIVITY RATIOS (SCALING FACTORS)					ACTIVITIES TO Ni-63			
		ACTIVITIES TO Co-60					C-14	Nb-94	Tc-99	TRU
LPRM	29	2.8E-05	3.8E-01				2.7E-06	7.4E-05		7.1E-06
	28	1.7E-06	6.6E-02	<1E-7				2.6E-05		
	29	8.9E-07	5.2E-02	<1E-7				1.7E-05		
	31	3.4E-07	9.9E-02	<2E-6			1.2E-07	3.4E-06		1.2E-06
	30	8.4E-06	3.3E-02	<9E-8			6.8E-08	2.5E-04		2.1E-06
	30	4.5E-06	5.0E-02	<2E-7			1.7E-07	9.0E-05		3.4E-06
	31	7.3E-05	4.8E-02	<2E-6			2.7E-07	1.5E-03		5.6E-06
	30	3.3E-06	3.7E-02	<2E-7			1.7E-07	8.9E-05		4.6E-06
	30	4.8E-06	4.9E-02	<4E-6			5.4E-08	9.8E-05		1.1E-06
	31	9.0E-05	9.2E-02	<3E-7			1.1E-07	9.8E-04		1.2E-06
	32	1.2E-05	5.7E-02	<4E-7			1.2E-07	2.1E-04		2.1E-06
	32	8.8E-06	3.7E-02				1.1E-08	2.4E-04		3.0E-07
	34	1.7E-06	7.0E-03	<6E-8			1.5E-07	2.4E-04		2.1E-05
	34	1.7E-06	6.7E-03					2.5E-04		
	35	1.7E-05	8.8E-02				3.3E-08	1.9E-04		3.8E-07
	36	2.3E-05	5.5E-02	<4E-7			6.8E-08	4.2E-04		1.2E-06
	37	4.6E-05	4.7E-02	<6E-6			1.6E-07	9.8E-04		3.4E-06
	37	4.7E-07	4.6E-03	<3E-7			1.9E-07	1.0E-04		4.1E-05
	38		4.7E-02	<6E-7						
	41	1.2E-05	6.8E-02	<2E-7			<2E-6	1.8E-04		1.1E-05
	33	6.6E-06	5.2E-02	<9E-8			5.7E-07	1.3E-04		1.1E-05
	33	7.9E-07	5.9E-02	<6E-8			6.5E-07	1.3E-05		1.3E-05
	33	6.0E-06	9.1E-02	3.3E-07			1.2E-06	6.6E-05	3.6E-06	
	42	3.4E-06	3.8E-02	<2E-7			<3E-9	8.9E-05		
	42	4.5E-06	3.0E-02	<2E-7			2.4E-08	1.5E-04		8.0E-07
	43	2.2E-08	5.4E-02	<2E-6			3.8E-08	4.1E-07		7.0E-07
	43	8.9E-08	4.2E-02	<9E-7		<5E-8	7.1E-08	2.1E-06		1.7E-06
	40	5.6E-04	4.2E-01	4.6E-06		1.2E-06		1.3E-03	1.1E-05	
	40	5.6E-04	4.2E-01	4.6E-06		1.2E-06		1.3E-03	1.1E-05	
	40	5.6E-04	4.2E-01	4.6E-06		1.2E-06		1.3E-03	1.1E-05	

Table G-11. (continued)

COMPONENT	REF.	MEASURED ACTIVITY RATIOS (SCALING FACTORS)						ACTIVITIES TO Ni-63			
		ACTIVITIES TO Co-60						C-14	Nb-94	Tc-99	TRU
IRM	36	2.2E-05	2.4E-01	9.3E-06	1.3E-07	1.4E-06	9.1E-05	3.9E-05	5.4E-07	5.8E-06	5.8E-06
	42	6.6E-07	4.7E-02	<6E-8		8.8E-08	1.4E-05			1.9E-06	1.9E-06
	39	2.5E-08	1.3E-01	<3E-6	<3E-8	4.9E-08	1.9E-07			3.8E-07	3.8E-07
	39	3.7E-08	1.1E-01	<5E-7	2.3E-08	6.0E-08	3.3E-07		2.1E-07	5.4E-07	5.4E-07
FUEL	37	2.4E-05	9.6E-02	<4E-6			2.5E-04				
SUPPORT PC.	33	2.3E-06	7.3E-02	<1E-6			3.2E-05				
FW SPARGER	33	<3E-7	2.0E-02	<3E-6							
	28	1.1E-05	1.1E-02	<6E-6			1.0E-03				
POISON CURTAIN	41	2.8E-06	1.8E-01	5.8E-06		<2E-6	1.5E-05	3.2E-05		9.6E-06	9.6E-06
	37	8.8E-06	5.1E-02	<8E-7		4.9E-07	1.7E-04			1.8E-05	1.8E-05
	37	4.6E-06	2.8E-02	<5E-7		5.0E-07	1.6E-04				
CRB GUIDE	28	1.5E-07	9.3E-02	<3E-6			1.6E-06				
UPPER & LOWER TIE PLATE, (PWR FUEL ASSEMBLY)	44	2.7E-06	1.4E-01	6.2E-06			2.0E-05	4.6E-05			
	25		4.4E-02	1.0E-07				2.3E-06			
	25		5.8E-02	9.7E-08				1.7E-06			
	25		1.3E-01	4.3E-07				3.4E-06			
	25		6.5E-02	1.9E-07				2.9E-06			
	25		6.3E-02	6.3E-08				1.0E-06			
	25		5.0E-02	3.7E-08				7.4E-07			
	16		2.1E-01	2.0E-07				9.5E-07			
	16		2.5E-01	1.1E-06				4.4E-06			
	16		1.9E-02	2.0E-07				1.1E-05			
	16		2.8E-02	1.3E-07				4.6E-06			
	17			3.5E-05							
	MINIMUM:		4.1E-03				1.9E-07	7.4E-07	2.1E-07	5.7E-09	5.7E-09
	MAXIMUM:		4.2E-01				1.2E-02	2.2E-04	2.9E-06	3.5E-04	3.5E-04
	LOG-MEAN AVERAGE:		6.3E-02				7.5E-05	8.3E-06	1.4E-06	3.5E-06	3.5E-06
	LOG-MEAN RMS DISPERSIO		2.5E+00				10.0	3.9	3.1	8.4	8.4
								In nCi/gm: 4.7E-04			

the crud layer on the component, deposited presumably (because of its similarity to crud deposits on other incore components) while the component was in the reactor pressure vessel. The TRU data represent only typical contamination levels and do not influence the component classification since the levels are usually well below any classification limits. Nonetheless, listing of this activity is required by 10 CFR 61.^{1,3} The last four columns relate four critical activities to ⁶³Ni. Scaling factors to the nickel isotope might be expected to show a smaller variation since nickel is a controlled major constituent of SS-304. Values at the bottom of the table give, respectively, the minimum and maximum values in each column and the log-mean average and RMS dispersion values. Averaging used a log-mean value because of the wide variance in the values and because measured scaling factors in utility-generated waste have demonstrated^{4,5} a log-normal distribution. The dispersion value is the RMS value of the ratios of the individual values to the log-mean average (i.e., a value of 2 means the individual data have a one-sigma spread of a factor of 2). The large values of dispersion for ¹⁴C and TRU are expected because of the origins of the activities and the lack of a physical reason for a generic correlation to cobalt.

Components of Zircaloy-4

Data for scaling factors for zircaloy-4 come from both PWR fuel skeleton assemblies and BWR fuel flow channels. In this material, both cobalt and nickel are trace elements, and the results were expected to show wide variances. Table G-12 presents the available measured scaling factor data for zircaloy-4. The table format is the same as that for presented for SS-304, Table G-11 above. Dispersion values are comparable to those in Table G-11, contrary to expectations.

Components of Inconel-718

Data for measured scaling factors in inconel-718 come entirely from measurements of spacer grids in Westinghouse fuel assembly skeletons. Table G-13 shows the data for this alloy in the same format as the previous tables. Inconel has high and presumably controlled nickel and niobium contents, and the analyses should show a stable value for the ⁹⁴Nb to ⁶³Ni scaling factor. However, the dispersion shown in the table for this factor is still large for unknown reasons. Uncertainties in the difficult radiochemical analysis for ⁹⁴Nb can account for only a small portion of the large spread.

Table G-12. Measured activity concentration ratios in activated components of zircaloy-4

COMPONENT	REF.	MEASURED ACTIVITY RATIOS (SCALING FACTORS)						ACTIVITIES TO Ni-63			
		ACTIVITIES TO Co-60			TRU			C-14	Nb-94	Tc-99	TRU
FUEL FLOW CHANNEL	28	1.0E-04	3.5E-02	<1E-5	<2E-4	2.9E-03		2.9E-03			1.6E-03
	29	1.9E-05	5.6E-02	<3E-6	8.7E-05	3.4E-04		3.4E-04			
	30	1.6E-05	1.1E-01	<3E-5		1.5E-04		1.5E-04			
	45	5.0E-07	7.0E-03	3.0E-06	3.1E-05	7.1E-05		7.1E-05	4.3E-04		4.4E-03
	45	1.8E-06	2.0E-02	3.5E-06	2.1E-04	9.0E-05		9.0E-05	1.7E-04		1.1E-02
	45	1.5E-06	2.1E-02	3.6E-06	3.3E-04	7.1E-05		7.1E-05	1.7E-04		1.6E-02
	34	1.2E-06	1.3E-02	<6E-8	2.8E-05	9.2E-05		9.2E-05			2.2E-03
	38	2.9E-04	6.0E-02	<2E-4		4.8E-03		4.8E-03			
GUIDE TUBES SPACER GRIDS (PWR FUEL ASSEMBLY)	44	1.5E-06	2.2E-01	<5E-6		6.8E-06		6.8E-06	2.8E-03		
	25		1.4E-03								
	25		1.7E-02	4.8E-05							
	25		1.1E-03								
	25		1.6E-02								
	25		8.5E-03								
	25		1.3E-02	4.6E-05					3.5E-03		
	25		6.5E-03								
	25		6.7E-03	6.3E-05					9.4E-03		
	25		2.0E-02	3.3E-06					1.6E-04		
	25		9.9E-03								
	25		1.4E-03								
	25		1.3E-02								
	25		4.0E-02	1.9E-06					4.8E-05		
	25		2.3E-02	2.0E-07							
	25		1.8E-02	1.8E-04					1.1E-05		
	17			1.0E-04							
	17										
MINIMUM: MAXIMUM: LOG-MEAN AVERAGE: LOG-MEAN RMS DISPERSIO			1.1E-03			6.8E-06		6.8E-06	1.1E-05	N.A	1.6E-03
			2.2E-01			4.8E-03		4.8E-03	9.4E-03	N.A	1.6E-02
			1.2E-02			1.8E-04		1.8E-04	3.6E-04	N.A	4.8E-03
			3.7E+00			7.4		7.4	8.8	In nCi/gm:	2.7 7.3E-01

Table G-13. Measured activity concentration ratios in activated components of Inconel-718

COMPONENT	REF.	MEASURED ACTIVITY RATIOS (SCALING FACTORS)						ACTIVITIES TO Ni-63			
		ACTIVITIES TO Co-60			TRU	Tc-99	Nb-94	C-14	Nb-94	Tc-99	TRU
		C-14	Ni-63	Nb-94							
SPACER GRID (PWR FUEL ASSEMBLY)	44	<3E-02	3.2E-01	8.1E-05					2.6E-04		
	25		7.2E-01	2.0E-04					2.8E-04		
	25		2.9E-01	1.0E-05					3.4E-05		
	25		3.1E-01								
	25		2.2E-01								
	25		2.0E-01								
	25		2.0E-01								
	25		2.4E-01								
	25		1.2E+00	8.4E-05					5.0E-04		
	25		2.6E-01	1.3E-04					6.3E-05		
	25		7.8E-01	4.9E-05					1.9E-03		
	17		1.6E-01	3.1E-04					4.0E-03		
	17		1.0E-01	4.0E-04					4.6E-03		
	17		1.2E-01	5.5E-04					3.4E-03		
	17		8.2E-02	2.8E-04							
	17			8.0E-04							
	17			7.1E-03							
	17			1.4E-03							
	17			1.3E-03							
MINIMUM:			8.2E-02					N.A.	3.4E-05	N.A.	N.A.
MAXIMUM:			1.2E+00					N.A.	8.8E+00	N.A.	N.A.
LOG-MEAN AVERAGE:			2.6E-01					N.A.	6.0E-04	N.A.	N.A.
LOG-MEAN RMS DISPERSIO			2.1E+00					N.A.	6.2	N.A.	N.A.
										In nCi/gm:	N.A.

APPENDIX B

**COMPILATION OF MEASURED
ELEMENTAL ANALYSIS DATA**

APPENDIX B: COMPILATION OF MEASURED ELEMENTAL ANALYSIS DATA

Computational analyses of radioactivity require the concentrations of all elemental constituents of the material being considered. An experimental study⁶ published in 1984 reported both major and minor constituents of common structural materials used in nuclear reactors. The results were extremely useful in reporting measured concentrations of the trace constituents of the materials, especially the cobalt and niobium important in radioactive waste management and classification. Activation analyses done since that time have widely used the results of that study. Although the study in Reference 6 analyzed several materials, the list included neither zirconium-4 nor inconel-718, used in some reactor internals. Some recent analyses added to the data base of analyses of SS-304 and include measurements of the other two structural alloys.

Table G-14 provides a summary of some later measurements of SS-304, zircaloy-4, and inconel-718. The table includes only the elements considered important for waste characterization and some major constituents of the alloys. Absent from the table is iron, whose activation does not influence GTCC LLW classification. Samples for the analyses were, for the most part, from unactivated reactor components obtained from utility plants. Analyses of principal and trace elemental constituents used combinations of precise instrumental analytical techniques of thermal neutron activation analysis, energy dispersive X-ray fluorescence, and inductively coupled argon plasma atomic emission spectrometry. These techniques are capable of measuring concentrations for the desired analyses to part-per-billion levels. Column 2 in the table lists the reference for the reported values. The elemental abundances are in terms of percent by weight. The table gives minimum, maximum, average and standard deviation (RMS) values for each element where there were sufficient data to warrant such evaluations. Averages are linear and not weighted. Table G-14 lists values for zirconium and tin only because they are the principal constituents of zircaloy and because tin activates to produce ¹²⁵Sb, a large contributor to the dose rate from activated zircaloy-4. For SS-304, the results of this compilation differ only slightly from those in Reference 6. The values for inconel-718 show a surprising and misunderstood variation in the reported niobium concentrations.

Analyses used measured relative radionuclide activities (scaling factors) to determine elemental constituent abundance in alloys. Computing elemental abundances from scaling factors also requires assumptions for the relative neutron cross sections of the activation. Thus, comparisons of abundances computed from scaling factors to those from directly measured stable abundances will show the uncertainties in these cross sections as well. Nonetheless, compilation of elemental abundances from scaling factor data provides additional data for the knowledge base. The results are directly applicable to hardware activation since they result from actual activation of the material in power reactor cores. Table G-15 gives a summary of elemental constituents of SS-304, zirconium-4 and inconel-718 from an analysis of scaling factor data presented in Appendix A. Scaling factor values yield only relative values, and normalization must be used to obtain absolute abundances. Analyses used an assumed abundance for nickel for normalization, and the second column in the table lists these values. For SS-304 and inconel, the normalization value is the average abundance of nickel from Table G-14. Since no positive analyses for nickel in zirconium-4 were found, the value in the table resulted from analysis of data reported by McCarten and Luksic. McCarten⁷ gives results from the analysis of 36 inconel components of Prairie Island fuel skeletons in units of Ci/gm. Luksic⁸ presents activation activities for fuel skeletons in units of Ci/gm of the element irradiated. These latter values were from computations using ORIGEN2 for fuel burnup comparable to that for the Prairie Island fuel ($\approx 32,000$ MW·d/MTU).

Table G-14. Measured elemental constituents of reactor components

MATERIAL	REF	ABUNDANCE (WT %)					
		Ni	Co	Nb	Mo	Zr	Sn
SS-304	4,25	8.3	0.149	0.0033	0.23		
	4,25	9.1	0.150	0.0024	0.25		
	4,25	9.4	0.051	0.0300	0.37		
	4,25	8.0	0.128	0.0156	0.87		
	4,25	9.7	0.115		0.41		
	4,25	8.5	0.147		0.23		
	4,25	9.6	0.092				
	4,25	9.8	0.148				
	4,25	8.4	0.026	0.0010			
	4,25	8.3	0.025	0.0007			
	4,25	8.7	0.210	0.0370			
	4,25	8.8	0.210	0.0180			
	16	8.9	0.025	0.0250			
	16	9.0	0.028	0.0085			
	16	9.1	0.030	0.0110			
	16	8.8	0.028	0.0240			
	16	8.5	0.071	0.0091			
	16	8.9		0.0100			
	MAX:	9.8	0.210	0.0370	0.87		
	MIN:	8.0	0.025	0.0007	0.23		
	AVG:	8.9	0.096	0.0140	0.39		
	STD:	0.5	0.064	0.0110	0.22		
ZIRCALOY-4	4,25	<0.005	<0.0027		0.87	91.8	2.2
	4,25					89.0	2.3
	4,25					91.7	2.4
	4,25					92.0	2.6
	4,25					92.5	3.0
	4,25					115.0	3.9
	4,25					106.0	3.4
	4,25					73.5	2.0
MAX:						115.0	3.9
MIN:						73.5	2.0
AVG:						93.9	2.7
STD:						11.4	0.6

Table G-14. (continued)

MATERIAL	REF	ABUNDANCE (WT %)				Zr	Sn
		Ni	Co	Nb	Mo		
INCONEL-718	4,25	50.8	0.071	4.5	2.86		
	4,25	53.0	0.148	4.4	2.79		
	4,25	46.5	0.089	4.6	2.66		
	4,25	48.0	0.115	2.3	2.83		
	4,25	50.8	0.131	2.3	2.98		
	4,25	47.0	0.119		2.65		
	4,25	49.2	0.130		2.87		
	4,25	49.3	0.111		2.67		
	16	51.9	0.051	4.6			
	16	52.0	0.055	4.9			
	16	44.0	0.043	4.8			
	16	48.9	0.074	4.8			
	16	52.5	0.035	5.1			
	16	49.9	0.052	4.9			
	16	52.0		5.6			
MAX:		53.0	0.148	5.6	2.98		
MIN:		44.0	0.035	2.3	2.65		
AVG:		49.7	0.087	4.4	2.79		
STD:		2.5	0.036	1.0	0.11		

Table G-15. Elemental composition analysis from measured nuclide scaling-factor data

MATERIAL	WEIGHT % NICKEL		SCALING FACTOR	a (Gm/Gm)	WEIGHT % COBALT		SCALING FACTOR	(Gm/Gm)	WEIGHT % NIOBIUM		SCALING FACTOR	(Gm/Gm)	WEIGHT % TC-MO	
STAINLESS STEEL-304	8.9	0.063	b	1.3x10 ⁻²	0.12		8.2x10 ⁻⁶	3.1x10 ⁻⁴	0.003		1.4x10 ⁻⁶	1.7x10 ⁻²	0.15	
ZIRCALOY -4	2.1x10 ⁻⁴	0.019	c	6.9x10 ⁻²	1.4x10 ⁻³		1.1x10 ⁻⁵	4.1x10 ⁻⁴	8.4x10 ⁻⁶					
INCONEL -718	49.7	0.26		3.2x10 ⁻³	0.16		6.0x10 ⁻⁴	2.2x10 ⁻²	1.11					

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a. Values of metal activation are from ORIGEN2 computations reported in reference 4 in units of Ci produced per gram of the element: ⁶³Ni: 7.42x10⁻², ⁶⁰Co: 89.6, ⁹⁴Nb: 1.99x10⁻³, ⁹⁹Tc: 5.99x10⁻⁶. These values will change with neutron fluent (i.e., reactor power level and position in vessel); however, since only ratios of the values are used, their accuracies should be adequate.

b. The equation for calculating the gram per gram ratios is:

$$\frac{\left(\frac{Ci}{Cc}\right)_A \times \left(\frac{Ci}{Gm}\right)_B}{\left(\frac{Ci}{Cc}\right)_B \times \left(\frac{Ci}{Gm}\right)_A} = \left[\frac{Gm_A}{Gm_B}\right]$$

where the first term has the scaling factors and the second term has the values from #1.

c. The normalizing value results from analyses of concentration data in Reference 7 and activation values from Reference 8 as discussed in the text.

The resulting value for nickel concentration is a factor of 10 below the detection limit reported for the single reported attempt to measure elemental nickel, shown in Table G-14. Columns 3-5 in the table give values for cobalt. Column 3 shows the log-mean average of the ^{60}Co scaling factor (relative to ^{63}Ni) from Appendix A. Column 5 gives the value for grams of cobalt per gram of nickel, computed using the equation in footnote b. The first ratio in the equation is the scaling factor, and the second ratio contains the activation factors listed in the footnote and extracted from Reference 8. Column 5 gives the abundance of cobalt obtained from multiplying the value in column 4 by the normalizing abundance of nickel in column 2. These analyses are repeated for niobium and molybdenum in the last six columns. The results are comparable to those in Table G-14 except for niobium, where the values are about a factor of four lower in Table G-15, not as surprising for SS-304, where the abundance is low, as for inconel, where analytical precision for both elemental and scaling factor analyses should be relatively high. No explanation is known except for possible errors in the assumed activation cross sections for niobium, both thermal and resonance, or systematic errors in measured niobium scaling factors.

APPENDIX C

MEASURED ^{60}Co ACTIVITIES IN COMPONENTS-- DOSE RATE DATA

APPENDIX C: MEASURED ^{60}Co ACTIVITIES IN COMPONENTS-- DOSE RATE DATA

For most bulk radioactive waste, and particularly for irradiated hardware, it is impractical from both radiation exposure and economic considerations to analyze completely an entire component for radioactive content. Furthermore, most wastes and irradiated hardware have radioactivity concentrations and sometimes materials that are inhomogeneous over the entire component. Thus, extrapolating and scaling the analyses of small samples to the entire volume is usually inaccurate and impractical. A more practical and most often used method of classifying bulk radioactive waste is a three-step process: (a) measure the average gamma ray dose rate from the component or package, (b) apply a predetermined dose to curie factor to convert the average dose rate to average concentration of ^{60}Co , and (c) apply a set of scaling factors to estimate the concentration of the other activities from the ^{60}Co activity. Scaling factors are discussed in Appendix A. Dose to curie conversion factors are both measured⁹ and computed,¹⁰ using known or estimated relative concentrations of gamma ray-emitting nuclides. Determination of average dose rate results from averaging values from a series of measurements at uniform intervals over the volume of the component.

Control Rod Blades

Table G-16 lists typical dose rate profile data along the lengths of BWR control rod blades. The table shows data from nine typical blades measured while they were in a utility fuel pool using an Eberline model R07 dose rate meter contained and calibrated in an immersable housing. The distance along the blade from the top end is as shown in the first column (the top is defined as the hot end containing the stellite rollers). In many of the rods, the dose rate at zero meters is highly influenced by the stellite (a cobalt alloy used by GE for bearings), and its effect must be removed before averaging. The reference for each set of data is shown at the top of the columns, and the maximum, minimum, and linear averages are shown at the bottoms of the columns. The more active rods were in the reactor core for a fuel burnup equivalency of about 35,000 MW·d/MTU, whereas the lower activity rods were removed prematurely for various reasons. This spread is typical for blades in utility fuel pools, and the low concentration rods are mixed with high activity rods in packages for burial to achieve lower averages over the contents.

Table G-17 lists the average dose rates for 45 scanned CRBs, together with the reference for the data and the utility rod identification where available. The bottom of the dose rate column gives the minimum, maximum, and linear average values for the compilation. The dose to curie factor for ^{60}Co (Reference 9), resulting from an average of measured values, is given in column 5. The last column lists the average concentration of ^{60}Co for each entry, resulting from multiplying the average dose rate by the dose to curie factor.

Table G-16. Typical dose profile data, control rod blades

DISTANCE (Meters)	DATA REFERENCE REPORT									
	29	29	32	32	32	31	31	37	37	37
			DOSE RATE (KR/Hr)							
0.0	14	15.8	20	22	9.8	22	21	13.8	16.5	
0.3	7.4	14.9	7.3	20	0.5	21	20	6.6	12.0	
0.6	7.4	12.4	5.3	18.8	0.35	19.7	18.1	5.7	12.1	
0.9	5.9	11.3	0.7	14.6	0.25	20	20	5.5	12.0	
1.2	5.4	10.9	0.7	12	0.13	20	19.8	5.6	11.4	
1.5	5.1	9.9	0.3	10.9	0.06	20	19.5	0.2	10.3	
1.8	0.9	9.8		9.3	0.01	20	18.7	3.3	9.5	
2.1	0.8	9.6	3.1	8.6	0.005	20	17.8	1.9	8.6	
2.4	3.9	10	2.3	7.5		20	16.2	0.7	7.5	
2.7	3.3	5.5	1.5	5.9		18.6	13	0.2	5.7	
3.0	1.1	3.1	0.9	0.1		15.7	8.1	0.06	2.9	
3.3	0.15	1.05	0.4	2.7		10.8	3.9	0.03	1.3	
3.6	0.01	0.06	0.15	1.5		4.9	1.6	0.01	0.27	
3.9		0.01	0.04	0.5		1.8	0.31			
4.2						0.13	0.04			
MAX:	14.0	15.8	20.0	22.0	9.8	22.0	21.0	13.8	16.5	
MIN:	0.01	0.01	0.04	0.1	0.005	0.13	0.04	0.01	0.27	
AVG:	4.3	8.2	3.3	9.6	1.4	15.6	13.2	3.4	8.5	

Table G-17. Average dose rate and Co-60 concentration data

COMP.	DATA REFERENCE	IDENT.	AVERAGE CONTACT DOSE RATE (kR/Hr)	DOSE-TO CURIE FACTOR (Co-60)	AVERAGE CONCENTRATION OF Co-60 (Ci/M-3)
VOLUME: .0266 M-3	46	6	7.43	3.3	24.5
	46	12	9.05	+1.3	29.9
	46	21	10.31	-0.9	34.0
	46	26	7.12		23.5
	46	32	7.43		24.5
	46	40	7.01		23.1
	46	44	6.50		21.5
	46	61	8.26		27.3
	46	71	8.66		28.6
	46	75	13.84		45.7
	46	77	7.45		24.6
	46	84	5.91		19.5
	46	87	5.63		18.6
	46	98	10.29		34.0
	46	109	9.92		32.7
	46	C3	8.25		27.2
	46	G6	11.91		39.3
	46	H9	6.64		21.9
	46	63	5.55		18.3
	46	91	4.75		15.7
	46	122	4.78		15.8
	46	124	3.69		12.2
	46	128	5.38		17.8
	46	52	6.39		21.1
	46	17	4.43		14.6
	46	106	8.60		28.4
	46	130	4.75		15.7
	46	504	12.63		41.7
	46	81	10.81		35.7
	46	316	12.80		42.2
	46	152	12.02		39.7
	46	491	12.48		41.2
	46	101	5.91		19.5
	46	107	5.77		19.0
	46	41	11.30		37.3
	46	126	8.68		28.6
	46	16	11.86		39.1
	31		15.60		51.5
	31		13.20		43.6
	37		3.40		11.2
	37		8.50		28.0
	29		4.30		14.2
	29		8.20		27.1
	32		3.30		10.9
	32		9.60		31.7
		MIN:	3.30		10.9
		MAX:	15.60		51.5
		AVG:	8.23		27.2

Incore Instrumentation

Table G-18 shows typical dose profile data for six local power range monitors (LPRM) and three intermediate range monitors (IRM). For these components, also, the higher activity LPRMs received a total neutron fluence equivalent to a fuel burnup of about 35,000 MW·d/MTU, whereas the others were in for shorter periods for various reasons. The range of activities is typical. The IRMs are designed to remain in the core for the reactor life and receive proportionally larger doses. The three listed in the table were removed prematurely for some reason but, nonetheless, show considerably more activity than the LPRMs. This, too, is typical. Both LPRMs and IRMs contain fission chambers having milligram quantities of fast and thermal fissioning isotopes of uranium; however, their fission product and fissile material contents, averaged over the volume of metal in the instrument, are well below Class C limits and do not influence the classification of the component.

Table G-19 summarizes average dose rate data from 17 LPRMs, 9 IRMs, and 2 SRMs, representing typical activities of these components currently in fuel pools. The references for the data and the utility identifications are also listed. Three of the LPRMs were identified by manufacturer (Reuter Stokes, General Electric and Westinghouse). No significance should be given to their different average dose rates other than the fact that they each received a different total neutron fluence. The table also includes the measured dose to curie (DTC) factor for ^{60}Co (Reference 9) for these three components (used for all three since the geometries are the same for each type of instrument). The last column gives the average concentration of ^{60}Co from application of the DTC factor to the average dose rate. The dose rate and ^{60}Co columns are averaged at the bottom of the columns.

Fuel Flow Channels

Table G-20 shows dose profile data from six typical BWR fuel flow channels. The channels typically are used with one fuel assembly during its life in the core ($\approx 35,000$ MW·d/MTU). Because of the low cobalt content in the zircaloy channel, the average dose rate is low. Further, the high tin content of zircaloy-4 causes over 50% of the dose rate to result from its activation to ^{125}Sb in fresh channels. This ratio that changes with time because of the different half lives of ^{60}Co and ^{125}Sb must be taken into account when using dose rate to classify components of zircaloy-4.

Table G-21 summarizes the available average dose rate data from fuel flow channels and from three other BWR components, poison curtains, feedwater sparger nozzles, and fuel support pieces. The table includes dose to curie factors (Reference 9) for ^{60}Co , average ^{60}Co concentrations for each component, and averages for each type of component.

Table G-18. Typical dose profile data, incore instruments

DISTANCE (Meters)	DATA REFERENCE REPORT									
	28	29	29	32	31	30	34 (IRM)	42 (IRM)	37 (IRM)	
	DOSE RATE (kR/Hr)									
0.0	0.7	0.4	0.0	2.5	1.7	2.2	21.0	2.6	3.0	
0.3	1.5	1.7	1.2	4.0	2.5	15.2	21.0	16.2	8.3	
0.6	1.7	3.1	3.2	4.9	3.0	17.0	21.0	23.5	12.3	
0.9	1.8	3.0	4.0	6.6	9.8	19.0	21.0	24.0	13.2	
1.2	1.8	3.1	4.5	6.7	11.8	19.5	21.0	25.0	12.9	
1.5	1.8	2.8	4.4	6.8	13.3	20.0	21.0	22.3	14.0	
1.8	1.4	2.8	4.0	6.8	13.4	19.0	21.0	23.6	13.3	
2.1	1.8	2.6	3.8	6.7	13.5	19.0	21.0	23.0	13.9	
2.4	1.4	2.5	3.4	7.9	14.0	18.3	21.0	22.5	13.0	
2.7	1.3	2.1	2.8	7.1	13.0	17.0	21.0	21.7	14.5	
3.0	1.2	2.0	2.0	6.7	12.0	16.6	16.5	18.1	14.8	
3.3	1.5	1.8	0.8	6.7	11.8	15.8	3.8	18.1	15.2	
3.6	0.4	0.9	0.3	6.8	11.5	11.6	1.0	5.0	15.2	
3.9				4.3	10.0	6.0		1.0	15.8	
4.2				3.5	7.0	1.0			13.3	
4.5				2.7	3.0				8.6	
4.8				1.1	1.0				1.6	
5.1				0.07					0.1	
MAX:	1.8	3.1	4.5	7.9	14.0	20.0	21.0	25.0	15.8	
MIN:	0.4	0.4	0.0	0.1	1.0	1.0	1.0	1.0	0.1	
AVG:	1.4	2.2	2.6	5.1	9.0	14.5	17.8	17.6	11.3	

Table G-19. Average dose rate and Co-60 concentration for incore instruments,
measured dose rates and scaling factors

COMP.	DATA REFERENCE	IDENT.	AVERAGE CONTACT DOSE RATE (kR/Hr)	DOSE-TO CURIE FACTOR (Co-60)	AVERAGE CONCENTRATION OF Co-60 (Ci/M-3)
LPRM VOLUME 2E-4 M-3	42	REUTER-ST GE WEST 40-25	14.00	55.1	771.4
	34		19.80	+11.1	1091.0
	29		2.20	-9.2	121.2
	29		3.00		165.3
	28		2.30		126.7
	28		1.09		60.1
	32		10.30		567.5
	32		6.00		330.6
	33		15.20		837.5
	30		7.40		407.7
	31		10.40		573.0
	31		6.80		374.7
	31		15.20		837.5
	31		4.50		248.0
	31		6.15		338.9
	31		11.80		650.2
	31		8.90		490.4
MIN:			1.09		60.1
MAX:			19.80		1091.0
AVG:			8.53		470.1
IRM	36	36-13	12.92		711.9
	36	20-29	11.26		620.4
	36	12-13	9.29		511.9
	36	20-21	11.79		649.6
	36	28-29	9.89		544.9
	36	28-21	12.01		661.8
	36	36-45	12.61		694.8
	36	12-45	8.06		444.1
	42		20.60		1135.1
MIN:			8.06		444.1
MAX:			20.6		1135.1
AVG:			12.05		663.8
SRM	45	12-29	11.63		640.8
	45	20-13	10.72		590.7
AVG:			11.18		615.7

Table G-20. Typical dose profile data, fuel flow channels

DISTANCE (Meters)	DATA REFERENCE REPORT					
	28	29	45	45	45	34
	DOSE RATE (kR/Hr)					
0.0	0.010	0.020	0.056	0.660	0.470	0.020
0.3	0.023	0.021	0.087	0.280	0.260	0.150
0.6	0.020	0.063	0.130	0.340	0.270	0.210
0.9	0.025	0.090	0.176	0.360	0.420	0.170
1.2	0.025	0.106	0.173	0.430	0.440	0.140
1.5	0.025	0.109	0.170	0.440	0.430	0.120
1.8	0.025	0.114	0.178	0.440	0.400	0.110
2.1	0.025	0.112	0.175	0.430	0.430	0.100
2.4	0.025	0.113	0.168	0.430	0.440	0.100
2.7	0.025	0.104	0.165	0.430	0.420	0.070
3.0	0.025	0.107	0.156	0.450	0.390	0.040
3.3	0.025	0.098	0.144	0.420	0.370	0.040
3.6	0.025	0.085	0.109	0.360	0.350	
3.9	0.019	0.053	0.035	0.300	0.300	
4.2		0.021	0.028	0.230	0.220	
MAX:	0.025	0.114	0.178	0.660	0.470	0.210
MIN:	0.010	0.020	0.028	0.230	0.220	0.020
AVG:	0.023	0.081	0.130	0.400	0.374	0.106

Table G-21. Average dose rate and Co-60 concentration for assorted components, measured dose rates and measured scaling factors

	DATA REFERENCE	IDENT.	AVERAGE CONTACT DOSE RATE (kR/Hr)	DOSE-TO CURIE FACTOR (Co-60)	AVERAGE CONCENTRATION OF Co-60 (Ci/M-3)
POISON CURTAINS VOL: 2.1E-2	41		1.44	40.0	57.6
	36		1.86	+14.9	74.4
	36		0.99	-14.9	39.6
SPARGER NOZZLE VOL: 7.25E-2	33		0.01	6.70	0.07
	28		0.06		0.40
FUEL-SPT PC VOLUME: 3.5E-3	33		0.26	6.70	1.76
FUEL FLOW CHANNELS INCONEL DENS: 6.56 VOLUME: 5.3E-3	28		0.023	4.3	0.10
	29		0.092	+6.5	0.39
	32	DN-814	0.136	-2.9	0.59
	32	DN-764	0.097		0.42
	32	DN-767	0.079		0.34
	34	DN-769	0.789		3.39
	1		0.113		0.49
MIN:			0.023		0.10
MAX:			0.789		3.39
AVG:			0.190		0.82

PWR Fuel Assemblies

Figure G-4 shows pictorially a dose rate profile of a typical Westinghouse fuel assembly skeleton. The Westinghouse skeletons have inconel grid spacers and zircaloy tubes, and the dose rate profile reflects these different materials, with the spacers averaging nearly 4000 R/hr and the guide tubes averaging less than 300 R/hr. The figure shows data from a typical full burnup Westinghouse fuel assembly skeleton. Skeletons from the other manufacturers do not show the high dose rates at the grid spacers.

SKELETON DOSE-RATE PROFILE

Assembly I.D.: C-73

Location: S. S.-11

Date: 3/16/88

Days Out of Core
As of 3/16/88: 3409

Recorder:

Burnup: 38,654 MWD/MTU

Figure G-4. Typical skeleton dose rate profile.

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Greater-Than-Class C Low-Level Radioactive Waste Characterization

Appendix H: Packaging Factors for Greater-Than-Class C Low-Level Radioactive Wastes

*Greater-Than-Class C Low-Level Waste
Management Program*

August 1991

**GREATER-THAN-CLASS C
LOW-LEVEL WASTE CHARACTERIZATION**

**Appendix H: Packaging Factors for
Greater-Than-Class C Low-Level Radioactive Wastes**

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August 1991

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ABSTRACT

This report develops and presents estimates for a set of three values that represent a reasonable range for the packaging factors for several waste streams that are potential greater-than-Class C low-level radioactive waste. The packaging factor is defined as the volume of a greater-than-Class C low-level waste disposal container divided by the original, as-generated or "unpacked," volume of the wastes loaded into the disposal container.

Packaging factors take into account any processes that reduce or increase an original unpacked volume of a greater-than-Class C low-level radioactive waste, the volume inside a waste container not occupied by the waste, and the volume of the waste container itself. The three values developed represent (a) the base case or most likely value for a packaging factor, (b) a high case packaging factor that corresponds to the largest anticipated volume of waste for disposal, and (c) a low case packaging factor for the smallest volume expected.

Three categories of greater-than-Class C low-level waste are evaluated in this report: activated metals, sealed sources, and all other wastes. Estimates of reasonable packaging factors for the low, base, and high cases for the specific waste streams in each category are shown in Table H-1.

Table H-1. GTCC LLW packaging factors to be used in volume projections

<u>Potential GTCC LLW Waste Stream</u>	<u>Base Case</u>	<u>High Case</u>	<u>Low Case</u>
Activated Metals			
BWR Operations			
Control Rod Blades	5	10	4
Incore Instruments	5	10	3
PWR Operations			
Thimble Plug Assemblies	20	30	10
Incore Instruments	5	10	3
Primary Sources	1	1	1
BWR Decommissioning			
Core Shroud	2	3	1.4
PWR Decommissioning			
Core Shroud	2	3	1.4
Core Barrel	2	3	1.4
Sealed Sources	380	1,000	2
All Other GTCC LLW			
Decontamination Resins	1	2	0.2
Cartridge Filters	2	4	0.1
Aqueous Liquids	0.2	3	0.1
Solidified Liquids	1	1.6	0.8

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APPENDIX H: PACKAGING FACTORS FOR GREATER-THAN-CLASS C LOW-LEVEL RADIOACTIVE WASTES

1. INTRODUCTION

This report develops and presents estimates for a set of three values that represent a reasonable range for the packaging factor (PF) for several waste streams that are potential greater-than-Class C low-level radioactive waste (GTCC LLW). The PF is defined as the volume of a GTCC LLW disposal container divided by the original, as-generated or "unpackaged," volume of the wastes loaded into the disposal container. The PF will be used to determine the packaged volume of GTCC LLW that the U.S. Department of Energy (DOE) will receive from commercial waste generators.

The method for determining a PF considers the unpackaged volume of a potential GTCC LLW prior to any treatment and packaging for disposal. A PF estimates what that unpackaged volume will become when the wastes are actually processed (if processed at all) and packaged into a waste disposal container for final disposal in the future.

A PF for a specific GTCC LLW waste stream is the number of times an unpackaged volume will be increased or decreased to take into account predisposal treatment (if any) and packaging for disposal. Packaging factors take into account any processes that reduce or increase an original unpackaged volume of a GTCC LLW, the volume inside a waste container not occupied by the GTCC LLW, and the volume of the waste container itself.

A set of three values is proposed as the range for a reasonable PF estimate for potential GTCC LLW streams. These values are determined by considering existing low-level radioactive waste processing, handling, packaging, and transportation technologies, and the uncertainties in forecasting which technologies will be used later when GTCC LLW is treated, packaged, and transported for disposal. The three values represent (a) the base case or most likely value for a PF, (b) a high case packaging factor that corresponds to the largest anticipated volume of waste for disposal, and (c) a low case packaging factor for the smallest volume expected. The high and low cases are not proposed as worst case values, but rather as reasonable values for the range of the base case or most likely value for the PF.

PF is the reciprocal of packaging efficiency (PE). Packaging efficiency is an unpackaged volume of GTCC LLW inside a waste disposal container, divided by the volume of the outside dimensions of the waste disposal container. The assessments of reasonable PFs are developed below by determining the PEs and then taking the corresponding reciprocals.

The three categories of GTCC LLW evaluated in this report are: (a) activated metals, (b) sealed sources, and (c) all other wastes. The specific types of materials evaluated in each category and the generators are listed in Table H-2.

Table H-2. Potential GTCC LLW and generators

<u>Potential GTCC LLW Waste Stream</u>	<u>Generator Nuclear Power Plants</u>
Activated Metals	
BWR Operations	
Control Rod Blades	
Incore Instruments	
PWR Operations	
Thimble Plugs	
Incore Instruments	
Primary Sources	
BWR Decommissioning	
Core Shroud	
PWR Decommissioning	
Core Shroud	
Core Barrel	
Sealed Sources	Manufacturers and Users
All Other GTCC LLW	
Decontamination Resins	Nuclear Power Plants
Cartridge Filters	Nuclear Power Plants
Aqueous Liquids	Industrial and Academic Users
Solidified Liquids	Fuel Testing and Bburnup Evaluation Facilities and Industrial Users

2. ACTIVATED METALS

PFs for activated metal GTCC LLW are developed in this section. The large components, core barrels and core shrouds, are evaluated in Section 2.1. The small components, BWR control rod blades, incore instruments, etc., are considered in Section 2.2. For each type of component, background information and assumptions on the scenarios for packaging of such materials are presented. The calculational models are then described and results provided. Estimates of the base, high, and low cases for reasonable PFs to be used in disposal volume projections are made after evaluating the results.

An important consideration in the analyses is the assumption that different waste streams of GTCC LLW are **not** mixed in the same waste disposal container. This assumption does not reflect the current practices for the packaging of LLW for burial at commercial LLW disposal sites. Within the guidelines acceptable at each disposal site, various small-size activated metal components are normally mixed together and disposed of as commercial LLW. While the acceptability of this practice for commercial and DOE disposal in the future will influence the types and quantities of potential GTCC LLW that ultimately are disposed of as GTCC LLW, this study purposely ignores the possibility that activated metal waste streams will be mixed in determining the reasonable PFs for each specific waste stream examined.

In using this approach, a higher PF will result for each individual waste stream than would be the case with mixing (i.e., more containers of GTCC LLW will be predicted than would actually be generated). This is because the mixing of different streams would enable small components to be placed into containers with pieces of large components, filling in some of the void spaces in those waste containers. Taking advantage of mixing to reduce costly disposal volume seems a reasonable practice that would be applied to GTCC LLW in the future.

However, by not considering mixing, the PFs for unmixed waste streams are expected to be only slightly higher than for the cases where mixing would be performed. As will be seen by the range of PF values in the results, such a small difference should not affect the usefulness of the values proposed for the PFs. Also, this approach is helpful since it allows computer modeling of an individual waste stream and prevents the problem of mathematically treating the many possibilities that arise if waste streams were evaluated as mixed.

2.1 Large Activated Metal Components

2.1.1 Background Information and Assumptions

The large activated metal components (LAMC) considered in this study are the boiling water reactor (BWR) and pressurized water reactor (PWR) core shrouds and the PWR core barrels. These LAMC are expected to be generated as potential GTCC LLW when nuclear power plants are decommissioned. Previous estimates of the PFs for LAMC are assumed to have been based on requirements (current at that time) for limiting the curies in waste disposal containers sent to commercial low-level waste burial sites. The relatively low amount of total curies allowed in a single disposal container compared to the high concentration of curies predicted to be in typical LAMC meant that large components would have needed to have been cut into many small pieces and distributed into many waste containers. Hence, low PEs and correspondingly high PFs were estimated. In contrast to the previous estimates, no constraint is placed on the total amount of curies in a disposal container of LAMC. Although the waste acceptance criteria for disposal have not been

established by DOE for GTCC LLW, a limit on curies in containers of LAMC is not a reasonable assumption for this study.

In evaluating reasonable PFs for LAMC, the unpackaged volume of the pieces that can be loaded into potential disposal containers is determined by the shapes and sizes of those pieces which can be cut to fit into the container being considered. Since the sizes of the LAMC at the reactor sites vary, and the size (or sizes) of the waste containers acceptable to DOE is not fixed, many combinations of component and container sizes are possible for future packaging configurations for LAMC. By looking at the range of the possible combinations, a reasonable estimate of the most likely or base case PF is determined. Similarly, the high and low cases are based on the spread of values for the PFs for the many combinations.

In using this approach to determine PFs for LAMC waste containers, an additional consideration enters into determining the acceptable sizes. The container's dimensions must be near the sizes considered for either truck or rail casks.

In the calculations, the size of the cylindrical waste container is fixed, and sections of the lower core barrel (LCB) are added to the inside as long as the next piece fits into the waste envelope or, for large waste containers, until an entire LCB is in a single waste container.

Model TITAN Truck Cask. The description of the calculational model for the model TITAN truck cask is the same as is used in these calculations except that the number of sections of LCB loaded into the waste container is not limited by the weight constraint of the TITAN cask. In these cases, the number of sections is increased until no additional section can fit into the waste container.

The results of the calculations are provided in Table H-14 in Appendix A for the TITAN truck cask design. The results show the effects on PEs as the inside diameter and thickness of the LCB are varied for cylindrical, volume-constrained waste containers. The PEs for an entire LCB vary from a low value of 0.567 to a high value of 0.711. Figure H-3 in Appendix A shows graphically the reason for the high PEs for the case with an LCB with an inner diameter of 148 in. and thickness of 2.5 in. For this case, the weight of the LAMC would be 15,407 pounds, and a total payload weight of 16,992 pounds must be accommodated by the cask. The total number of waste containers needed to package an entire LCB varies from three to five containers.

Model IF-300 Rail Cask. The description of the calculational model for the Model IF-300 rail cask is based on the fact that the number of sections of LCB loaded into the waste container is not limited by the weight constraint of the IF-300 cask. In these cases, the number of sections is increased until no additional sections can fit into the waste container or until an entire LCB is in a single waste container.

The results of the calculations are provided in Table H-15 in Appendix A for the IF-300 rail cask design with cylindrical waste containers. The results show the effect on PEs as the inside diameter and thickness of the LCB are varied for cylindrical, volume-constrained waste containers. The PEs for an entire LCB vary from a low value of 0.419 to a high value of 0.786. The total number of waste containers needed to package an entire LCB is one to two containers.

Model 125-B Rail Cask. The description of the calculational model for the Model 125-B rail cask is based on the fact that the number of sections of LCB is not limited by the weight constraint of the 125-B cask. In these cases, the number of sections is increased until no additional section can fit into the waste container or until an entire LCB is in the waste container.

The results of the calculations are provided in Table H-16 in Appendix A for the Model 125-B cask with cylindrical cross section waste containers. The results show the effect on PEs as the inside diameter and thickness of the LCB are varied for cylindrical cross section, volume-constrained waste containers. The PEs for an entire LCB vary from a low value of 0.421 to a high value of 0.726. The total number of waste containers needed to package an entire LCB is one container.

Volume-Constrained Square Cross Section Waste Containers. The calculational model for square cross section containers considers three variations of transport packages. In the calculations the size of the waste container is fixed, and sections of the LCB are added to the inside as long as the next piece fits into the waste envelope or, for large waste containers, until an entire LCB is in the waste container.

PWR Spent Fuel Casks. Any of the spent fuel casks existing or under development with baskets for PWR spent fuel assemblies have a square cross section cavity into which a waste container of sections of an LCB could be placed for transport by either truck or rail mode. The square cross section cavity is assumed in these calculations to be 8.75 in. on a side and deep enough to accept the waste container. A clearance of 0.25 in. is allowed between each inside surface of the square cavity and the outside surface of a waste container for loading it into and removing it from the cask.

The nominal outside dimensions of a square cross section waste container are 8.25 in. on a side and 164.5 in. in overall length. These outer envelope dimensions are used in calculating the volume of the waste container. The length is based on the length of a section of the LCB of 160.5 in., an axial clearance of 1 in., a bottom plate thickness of 1 in., and a top plate thickness of 2 in. The wall thickness of each side is assumed to be 0.25 in. These values may be reduced in thickness after analyzing for adequate structural strength during lifting. The axial space in the cask cavity not filled with the waste container is assumed to be filled by a lightweight spacer of the appropriate length.

A clearance of 0.25 in. is allowed between each inside surface of the square cross section cavity of the waste container and the outside surface of a waste envelope for loading sections of the LCB into the waste container. The waste envelope thus has a square cross section of 7.25 in. on a side and a length of 161.5 in. inside the waste container.

The results of the calculations are provided in Table H-17 in Appendix A for casks with PWR fuel baskets. The results show the effect on PEs as the inside diameter and thickness of the LCB are varied for square cross section waste containers. The PEs for an entire LCB vary from a low value of 0.498 to a high value of 0.614. Figure H-4 in Appendix A shows graphically the reason for the average PEs for the case with an LCB with an inner diameter of 148 in. and a thickness of 2.5 in. For this case, the weight of the LAMC would be 1,910 pounds, and a total payload weight of 2,103 pounds must be accommodated by the cask for each cavity in the spent fuel basket. The total number of waste containers needed to package an entire LCB varies from 21 to 36 containers.

Model GA-4 Truck Cask. A newly designed truck cask, the Model GA-4, has a square cross section cavity with 18 in. on a side and a length of 167.25 in. Using the same approach as outlined above, the dimensions for these cases are:

Cask cavity side length = 18 in.

Clearance between inside surfaces of the cask cavity and outside surfaces of the waste container = 0.25 in.

Outside dimensions of the waste container:

Side length = 17.5 in.

Overall height = 164.5 in.

Wall thickness of the waste container = 0.25 in.

Clearance between the inside surfaces of the waste container and the outside of the waste envelope = 0.25 in.

Outside dimension of the waste envelope:

Side length = 16.5 in.

Height = 161.5 in.

The results of the calculations are provided in Table H-18 in Appendix A for the GA-4 truck cask design without a fuel basket. The results show the effect on PEs as the inside diameter and thickness of the LCB are varied for square cross section, volume-constrained waste containers. The PEs for an entire LCB vary from a low value of 0.665 to a high value of 0.795. The total number of waste containers needed to package an entire LCB varies from four to seven containers.

Model 125-B Rail Cask. An existing rail cask, the Model 125-B, has a cylindrical internal cavity of 51.25 in. inside diameter and is 190 in. in depth. The inner containment vessel is assumed to be removed and replaced by a new design basket that has a square inscribed in the circular cavity. A square cross section waste container would be placed into the special basket for transport. A radial clearance of 0.625 in. would be allowed between the inside surface of the cask cavity and the outside surface of the special basket. The outer diameter of the special basket would then be 50 in. Assuming a 0.5 in.-thick wall for the special basket would result in an inside diameter of 49 in. The side of a square inscribed in the inside diameter would be 34.64 in. A square cross section cavity of 34.5 in. is assumed.

Using the same approach as outlined above, the dimensions for these cases are:

Cask basket cavity side length = 34.5 in.

Clearance between inside surfaces of the cask cavity and outside surfaces of the waste container = 0.25 in.

Outside dimensions of the waste container:

Side length = 34 in.

Overall height = 167.5 in.

Wall thickness of the waste container = 0.375 in.

Clearance between the inside surfaces of the waste container and the outside of the waste envelope = 0.25 in.

Outside dimension of the waste envelope:

Side length = 32.75 in.

Height = 161.5 in.

The results of the calculations are provided in Table H-19 in Appendix A for the Model 125-B

rail cask with an inner containment vessel replaced by a basket for square cross section waste containers. The results show the effect on PEs as the inside diameter and thickness of the LCB are varied for square cross section, volume-constrained waste containers. The PEs for an entire LCB vary from a low value of 0.417 to a high value of 0.781. The total number of waste containers needed to package an entire LCB varies from one to three containers.

2.1.2 Packaging Factor Estimates for LAMC

A total of 117 cases were evaluated, which represent nine different sizes of LCBs loaded into spent fuel casks in 13 weight or volume-constrained waste containers. The resulting PE for each case represents a possible loading of LAMC into a waste container. Based on the many PEs that were determined and are summarized in Table H-20 in Appendix A, the proposed base case for PFs for LAMC is a factor of 2 or a PE of 0.5. This value reflects the considerable number of uncertainties that will affect the actual PFs. These include the actual diameters and thicknesses of existing LCBs, the effect of flanges and protrusions that may be present on LCBs, the tolerances to which the sections of an LCB can be cut remotely underwater, the clearances that will be required between sections of an LCB for loading into a waste container, the types of casks to be used for transport, and the sizes of the waste containers that will be used. While some reactor sites may achieve a PF higher than 2, others will likely not achieve even this high a value. For the overall LAMC waste stream from all reactor sites, an average of all the possibilities will likely most fairly characterize the PF actually to be realized.

A PF of 2 (PE of 0.5) is slightly higher than the average value of all the results for the PE for an entire LCB. This value appears to be the most reasonable for a base case PF without additional information to reduce some of the uncertainties that are included in establishing the ranges of the values for the variables in the study. A slightly higher-than-average value appears warranted since the lowest PE cases correspond to cask cavity filling, weight-constrained waste containers that are less likely to be used than cask cavity inefficient, weight-constrained waste containers. If spent fuel casks are to be used in the future, the void need only be in the cask cavity and not in both the cask cavity and the waste container (which is in the cask cavity).

A value for the PF of 3 or a PE of 0.33 is proposed for the high case for the range of PFs. This value represents the largest volume of LAMC that would be expected for disposal. Neglecting some of the lowest PEs noted above for cask cavity inefficient, weight-constrained waste containers, several of the cases that had a PE this low or lower represent combinations of variables that individually are considered reasonable. Since these combinations of variables are possible, they may determine actual PFs. If these cases represent many of the LCBs in existence, a PF of 3 might be realized.

A value for the PF of 1.4 or a PE of 0.71 is proposed for the low case for the range of PFs. This value represents the smallest volume of LAMC that would be expected for disposal. Several of the cases had PEs this high or higher for the PE for an entire LCB, and many more were higher for the fully loaded waste containers. A value as high as the highest PE generated was not selected, since the theoretical aspects of this evaluation have not been tested against the real-world problems of cutting large components remotely to fit into tolerances as tight as those that were used in the calculations.

However, these values for the base, high, and low ends of the range for PFs should be reasonable estimates even if larger tolerances for remote cutting and handling are found to be necessary. The reduction of the length of each of the sections in a waste container by an inch would

not change the values for the proposed PFs since these are not at the extremes of the values resulting from the calculations.

Conversely, the results from the many combinations of variables that were evaluated show that there are certain combinations where the geometry of the waste containers and the geometry of the LCB sections (e.g., LCB inside diameter and thickness) combine to tightly pack the sections into the waste container. There may be combinations of sizes of waste containers and LCBs where reducing clearances slightly would improve PFs further for the volume-constrained cases. That is, an additional section of an LCB may be able to fit into a waste container if the clearance allowed for the waste envelope were used.

As noted previously, the assumption that mixing will not be permitted would tend to overestimate the packaged volume of LAMC if such mixing is allowed, since small parts of other components could be used to fill the void spaces in containers of large pieces. Perhaps more significant, pieces of other components could be used to complete the filling of the last waste container only partially filled with pieces of a large component. As the data shows, the PEs for a single fully loaded waste container filled with sections of the LCB are higher than the overall PE for an entire LCB when the last filled waste container is only partially filled. The low PE of the last, partially filled waste container brings down the average for the group of waste containers. The smaller the number of waste containers in the group, the larger this effect becomes. Just utilizing the space in the last, partially filled container would thus raise the overall PE for all activated metals from a site.

2.2 Small Activated Metal Components

2.2.1 Background Information and Assumptions

The small activated metal components (SAMC) considered in this study are BWR control rod blades and incore instruments as well as PWR thimble plugs, incore instruments, and primary sources. These SAMC are expected to be generated as potential GTCC LLW during nuclear power plant operations and at decommissioning. Previous estimates of the PFs for SAMC for current operations are not published and are based on the practice of mixing pieces of SAMC in waste disposal containers sent to commercial LLW disposal sites.

In contrast to the LAMC, which are characterized by a single large item with well-defined geometry and volume requiring disposal, the SAMC consist of many small items generated during on-going operations without a definite generation rate. The volume requiring disposal is therefore not as determinate as for the LAMC. Also, since each item of the SAMC is small compared to the internal volume of the waste containers previously considered for the LAMC, there is a possibility that an insufficient amount of a specific SAMC waste stream will be available to fill the volume available in a single waste container.

Calculational Model and Results. The PFs for each of the SAMC materials are based on calculational models or engineering judgements as described below. Each specific SAMC waste stream is discussed individually. The models are based on filling a single waste container with a specific waste stream. This assumes a sufficient volume of the waste stream will be available to fill the container.

BWR Control Rod Blades. A BWR control rod blade (CRB) consists of long thin metal plates in the shape of a cruciform. The plates sandwich a neutron poison material that controls the operation of the BWR plant. A CRB is almost 10 in. across, and the width of the metal plate sandwich is about 0.3 in. A CRB is approximately 175 in. long, and the end not inserted into the reactor core has a cylindrically shaped velocity limiter integral to the CRB assembly. A CRB is activated by neutrons when it is inserted into the core. Since not all of a CRB is inserted, there is a "hot end" and a "cold end."

Base Case--The evaluation of the base case PF assumes that the velocity limiter on the cold end is cut off and disposed of as commercial LLW. The hot end is flattened between rollers and then placed into a cylindrical cross section waste container. The volume of the hot end of the CRB is approximately 0.6 ft³. The envelope dimensions of the flattened CRB are assumed to be 2 in. by 10 in. by 160 in. Two sizes of cylindrical waste containers are evaluated for the base case. One is for transport in a Model TITAN truck cask, and the other is for the IF-300 rail cask.

A new-design truck spent fuel cask under development, the Model TITAN, will have interchangeable baskets for PWR or BWR spent fuel assemblies. With the basket removed, there is a cylindrical cavity into which a waste container with CRBs could be placed for transport by truck. The cylindrical cavity is 23.75 in. inside diameter and 180 in. deep. A radial clearance of 0.25 in. is allowed between the inside surface of the cask cavity and the outside surface of a waste container for loading it into and removing it from the cask.

The nominal outside dimensions of a cylindrical waste container are 23.25 in. on a side and 164 in. in overall length. These dimensions are used in calculating the outer envelope volume of the waste container. The length is based on a length of a hot end of a CRB of 160 in., an axial clearance of 1 in., a bottom plate thickness of 1 in., and a top plate thickness of 2 in. The wall thickness of each side is assumed to be 0.25 in. These values may be reduced in thickness after analyzing for adequate structural strength during lifting, storage, and disposal. The axial space in the cask cavity not filled with the waste container is assumed to be filled by a lightweight spacer of the appropriate length.

A clearance of 0.25 in. is allowed between the inside surface of the cylindrical cavity of the waste container and the outside surface of a waste envelope for loading CRBs into the waste container. The waste envelope has a cylindrical cross section of 22.25 in. in diameter and a length of 161 in. inside the waste container.

Figure H-5 in Appendix A shows an arrangement of boxes (2 in. by 10 in.) that represent the envelope dimensions of flattened CRBs in the waste container. The 13 CRBs in the container have a displaced volume (not envelope dimension volume) of 7.8 ft³. With a waste container outer volume of 40.29 ft³, the PE for this size container is 0.19.

An existing rail cask, the Model IF-300 with the basket removed, has a cylindrical cavity with a 37.5 in. inside diameter and a depth of 180.25 in.

Using the same approach as outlined above, the dimensions for these cases are:

Cask cavity inside diameter = 37.5 in.

Clearance between inside surface of the cask cavity and outside surface of the waste container = 0.375 in.

Outside dimensions of the waste container:

Outside diameter = 36.75 in.

Overall height = 164 in.

Wall thickness of the waste container = 0.375 in.

Clearance between the inside surface of the waste container and the outside of the waste envelope = 0.25 in.

Outside dimensions of the waste envelope:

Diameter = 35.5 in.

Height = 161 in.

Figure H-6 in Appendix A shows an arrangement of boxes that represent the flattened CRBs in the waste container. The 37 CRBs in the container have a displaced volume (not envelope dimension volume) of 22.2 ft³. With a waste container outer volume of 100.67 ft³, the PE for this size container is 0.22.

From these two results, the base case for the PE for BWR control rod blades is 0.2, or a PF of 5.

Low Case--The evaluation of the low case PF assumes the velocity limiter on the cold end is cut off and disposed of as commercial LLW. The hot end is flattened between rollers and then placed into a square cross section waste container. The volume of the hot end of the CRB is approximately 0.6 ft³. The envelope dimensions of the flattened CRB are assumed to be 2 in. by 10 in. by 160 in. Two sizes of square cross section waste containers are evaluated for the low case. One is for transport in the Model GA-4 truck cask, and the other is for the Model BR-100 rail cask.

The Model GA-4 cask under development has a square cross section cavity into which a waste container with CRBs could be placed for transport by truck. The square cross section cavity is 18 in. on a side and has a depth of 167.5 in. A clearance of 0.25 in. is allowed between each inside surface of the square cavity and the outside surface of a waste container for loading it into and removing it from the cask.

The nominal outside dimensions of a square cross section waste container are 17.5 in. on a side and 164 in. in overall length. These outer envelope dimensions are used to calculate the volume of the waste container. The length is based on the length of a hot end of a CRB of 160 in., an axial clearance of 1 in., a bottom plate thickness of 1 in., and a top plate thickness of 2 in. The wall thickness of each side is assumed to be 0.25 in. These values may be reduced in thickness after analyzing for adequate structural strength during lifting, storage, and disposal.

A clearance of 0.25 in. is allowed between each inside surface of the square cross section cavity of the waste container and the outside surface of a waste envelope for loading the flattened CRBs into the waste container. The waste envelope has a square cross section of 16.5 in. on a side and a length of 161 in. inside the waste container.

Figure H-7 in Appendix A shows an arrangement of boxes that represent the flattened CRBs in the waste container. The 11 CRBs in the container have a displaced volume (not envelope dimension volume) of 6.6 ft³. With a waste container outer volume of 29.06 ft³, the PE for this size container is 0.23.

A new rail cask currently under development, the Model BR-100, has a cylindrical internal cavity of 58.5 in. inside diameter and 180 in. in depth. For these cases, the basket for spent fuel assemblies is assumed to be removed and replaced by a new design basket that has a square inscribed in the circular cavity. A square cross section waste container would be placed into the special basket for transport. A radial clearance of 0.75 in. would be allowed between the inside surface of the cask cavity and the outside surface of the special basket. The outer diameter of the special basket would then be 57 in. Assuming a 0.5 in. radial dimension for the structure of the special basket would result in a usable inside diameter of 56 in. for the waste container. The side of a square inscribed in the inside diameter would be 39.6 in. A square cross section cavity of 39.5 in. is assumed.

Using the same approach as outlined above, the dimensions for these cases are:

Cask basket cavity side length = 39.5 in.

Clearance between inside surfaces of the cask cavity and outside surfaces of the waste container = 0.25 in.

Outside dimensions of the waste container:

Side length = 39 in.

Overall height = 164 in.

Wall thickness of the waste container = 0.375 in.

Clearance between the inside surfaces of the waste container and the outside of the waste envelope = 0.25 in.

Outside dimension of the waste envelope:

Side length = 37.75 in.

Height = 161 in.

Figure H-8 in Appendix A shows an arrangement of boxes that represent the flattened CRBs in the waste container. The 63 CRBs in the container have a displaced volume (not envelope dimension volume) of 37.8 ft³. With a waste container outer volume of 144.35 ft³, the PE for this size container is 0.26.

From these two results, the low case for the PE for BWR control rod blades is 0.25 or a PF of 4.

High Case--The evaluation of the high case PF for CRBs assumes the velocity limiter on the cold end is not cut off, and the entire CRB unit is disposed of as GTCC LLW. Also, the hot end is not flattened between rollers. The CRB unit, as removed from the reactor, is placed into a square cross section waste container. The volume of the CRB unit is approximately 0.64 ft³. One size of square cross section waste container is evaluated for the high case PF. The square cross section waste container for transport in the Model GA-9 cask is similar to the previously described GA-4 cask. The GA-9 cask also has a square cavity 18 in. on a side but is 178 in. in length, which accommodates the full length of the CRB unit.

Figure H-9 in Appendix A shows an arrangement of CRB units in a waste container. The four CRBs in the container have a displaced volume (not envelope dimension volume) of 2.56 ft³. With a waste container outer volume of 31.89 ft³, the PE for this size container is 0.08. Based on this result and the likelihood that most CRBs would be flattened, the high case PE for CRBs is 0.1 or a PF of 10.

BWR and PWR Incore Instruments. Incore instruments for both BWR and PWR reactors consist of long, thin-walled metal tubes. The tubes contain neutron detectors that monitor operations of BWR and PWR reactors. An incore instrument tube is approximately 0.4 in. to 0.7 in. in diameter and can be over 100 ft long. An incore instrument tube is activated by neutrons when part of it is inserted into the core. Since not all of an incore instrument tube is inserted, there is a hot end and a cold end.

Base, High, and Low Cases--The evaluation of the PFs assumes the cold end is cut off and disposed of as commercial LLW. The hot end is placed into a square cross section waste container. The displaced volume of the metal tube is approximately 0.005 to 0.2 ft³, depending upon wall thickness of the metal tube. The small displaced volume and envelope dimensions of the incore instrument tube relative to the volume of the waste container require rather gross estimates for the base, high, and low case PFs.

The base case PF is estimated to be 5 or a PE of 0.2. This value is, however, no more than about the center of the range for the high and low cases. The high case PF is estimated to be 10 or a PE of 0.10. The low case PF is estimated to be 3 or a PE of 0.33. These proposed values consider the following calculations:

- The number of incore instrument tubes in the smallest size waste container was determined for several sets of variables. The smallest size container is one transported in a spent fuel cask for PWR assemblies as described above. The outside length of the square side of the container is 8.25 in., and the length of the side of the waste envelope is 7.25 in.
- The sets of variables considered the outer diameter (OD) of the incore instrument tubes, their wall thickness (THK), and the gap between the tubes. The gap represents the volume in a container that can not be efficiently used due to the difficulty in loading the tubes in a triangular pitch array in a waste container remotely underwater. The larger the gap between tubes, the fewer number of tubes that the container will hold.
- A PE of 0.34 results from tubes 0.7 in. in OD, 0.12 in. THK, with a small gap between tubes. A total of 108 tubes would be in the container. If the gap increases to 0.2 in. between tubes, the number of tubes in the container would drop to 64, and the PE would drop to 0.20.
- A PE of 0.28 results from tubes 0.4 in. in OD, 0.06 in. THK, with a small gap between tubes. A total of 358 tubes would be in the container. If the gap increases to 0.2 in. between tubes, the number of tubes in the container would drop to 144, and the PE would drop to 0.11.

These results show the sensitivity of the PE to the amount of volume not used for incore instrument tubes due to the difficulty in loading them remotely. Small changes in assumed wall

thickness likewise result in substantial changes in PEs. The three proposed values for the PFs should, however, account for the possible combinations and be reasonable estimates of the PFs for incore instrument tubes.

PWR Thimble Plugs. PWR thimble plugs or orifice rod assemblies resemble PWR control rod assemblies except that the rods attached to the flat plate or spider that rests on top of a PWR fuel assembly are much shorter than control rods. The rods are only a few inches long. A thimble plug has an envelope dimension of approximately 6 in. square cross section by 11 in. overall height. The thimble plugs are activated by neutrons since they are inserted into the core at the top of a PWR assembly. The entire assembly is activated and considered "hot," although the inconel spring is the activated part that causes a thimble plug to become GTCC LLW.

Base, High, and Low Cases--The evaluation of the PFs assumes that the thimble plug (or orifice rod assembly) is removed from the PWR fuel assembly and disposed of as a unit. The unit is placed into a square cross section waste container. The displaced volume of the unit is approximately 0.02 to 0.03 ft³. The small displaced volume and relatively large envelope dimensions of a thimble plug result in low estimates for the base, high, and low case PFs.

The base case PF is estimated to be 20, or a PE of 0.05. This value is, however, no more than about the center of the range for the high and low cases. The high case PF is estimated to be 30, or a PE of 0.03. The low case PF is estimated to be 10, or a PE of 0.1. These proposed values consider the following calculations:

- The number of thimble plugs in a square cross section waste container for the GA-4 truck cask was determined for several packaging configurations for the thimble plugs. The outside length of the square side of the container is 17.5 in., and the length of the side of the waste envelope is 16.5 in.
- The packaging configurations considered the units both as removed from the fuel assembly and with the rods flattened to be parallel to the top plate rather than perpendicular to it. The envelope dimensions in the flattened condition are 6 in. by 7 in. by 12 in. The unflattened units can either be unnested or nested.
- The unnested configuration in a container is due to the difficulty in loading the units remotely underwater as boxes based on their envelope dimensions in a waste container. The unnested configuration approximates a random dropping of units into a waste container.
- A PE of 0.09 results when thimble plugs are assumed stacked into a container standing on end as if in a fuel assembly. The rods in one plug should nest into the coupling of the plug below it and rest on the plate of the lower plug. In this type of nested stacking, there would be 23 layers of four plugs each, or a total of 92 plugs per container. If the displaced volume is 0.03 ft³, the PE would be 0.09. In comparison, if the displaced volume is 0.02 ft³, the PE would drop to 0.06.
- A PE of 0.06 results when thimble plugs are assumed to be stacked into a container lying horizontally. The rods of one plug would not be nested into another plug. In this type of nested stacking, there would be 27 layers of two plugs each, or a total of 54 plugs per container. If the displaced volume is 0.03 ft³, the PE would be 0.06. In comparison, if the displaced volume is 0.02 ft³, the PE would drop to 0.04.

- A PE of 0.09 results when thimble plugs are assumed to be stacked into a container lying horizontally and are nested with the rods from one plug nested into the coupling or rods of another plug. In this type of nested stacking, there would be 23 layers of four plugs each, or a total of 92 plugs per container. If the displaced volume is 0.03 ft^3 , the PE would be 0.09. If the displaced volume is 0.02 ft^3 , the PE would drop to 0.06.
- A PE of 0.11 results when thimble plugs are assumed flattened and then stacked into a container. The flattened rods are assumed to point up out of the container and are alternately all against the wall of the container in one layer and in the center of the container in the next layer. In this type of nested stacking, there would be 27 layers of four plugs each, or a total of 108 plugs per container. If the displaced volume is 0.03 ft^3 , the PE would be 0.11. In comparison, if the displaced volume is 0.02 ft^3 , the PE would drop to 0.07.

These results show the lack of sensitivity of the PF to the arrangement of the thimble plugs. The difficulty in loading them remotely will likely cause the PEs to be lower rather than higher. The three proposed values should, however, account for the possible random or stacking configurations and be reasonable estimates of the PFs for thimble plugs.

PWR Primary Sources. A PWR primary source is a neutron source material located in two rods in a burnable poison rod assembly. A burnable poison rod assembly without a primary source is not GTCC LLW. The rods containing a primary source can be cut off the assembly and then constitute a very small volume of as-generated GTCC LLW. With a displaced volume from 0.002 to 0.016 ft^3 per rod and only a few rods per reactor, the packaged volume would be very large if the rods are placed into any of the previously described waste containers by themselves. Primary rods are the one waste stream that does not warrant a waste stream-specific PF. These items should be mixed into waste containers with other types of GTCC LLW. For purposes of the model to predict packaged volumes, a PF of 1 should be used for the base, high, and low cases. This value will ensure that primary sources are not ignored as a specific waste stream. Also, the small value of the unpackaged volumes times a PF of 1 will result in a small contribution for this waste stream in comparison to the other activated metal components.

3. SEALED SOURCE

This section on sealed source GTCC LLW describes the types, quantities, and packaging factors for the major contributors of sealed source material. The principal references of data for this section are provided by the NRC survey^a on surplus sources and direct input from manufacturers and users of sealed sources within the industry.^b The section provides the current best estimate of GTCC LLW sources under Specific Licenses (>27,000) and General Licenses (>65,000). Also included are the major types of sources and the sizes specific to the sources; the devices; and the packages provided for handling, transport, and storage. To simplify the process in determining packaging scenarios and packaging factors based on the significantly large number of sealed source configurations (>2000), the sources and devices were characterized under three categories. The three major categories are moisture/density gauges, well logger units, and fixed/test gauges. These three types of sources are analyzed and modeled to determine the scenarios for the best estimate base case packaging factor as well as the high and low cases.

3.1 Background

Previous estimates of packaging factors for surplus sealed sources ranged from encapsulating nonreusable sources (currently $\approx 3,700$) with cement in 55-gal containers (packing factor of 6,600) to extracting and consolidating unshielded sources in separate HLW canisters (packaging factor of 14.5). The high case encapsulation scenario was based on stabilization requirements for commercial disposal of Class C LLW waste and therefore was not considered appropriately applicable to the GTCC LLW projections. Additionally, these previous cases did not specifically consider the practical condition of some surplus sources contained in devices, gauges, and deployment units and the potential return of such devices to DOE in the original package and transport containers.

The NRC survey concluded that over 92,500 GTCC sources currently exist. Approximately 6% of these GTCC sources and devices are currently awaiting disposal or transfer to a storage location. The predominant source materials contain americium-241, cesium-137, curium-244, and plutonium-238 and -239. Of the inactive sources, some have been returned to the manufacturer for storage, but the majority are being held by the users because of cost constraints and/or manufacturers' refusal to accept returns. The users predominantly store these sources and devices in their original receiving configuration. Of those returned to the manufacturers, the majority of devices and source gauges have been disassembled to the extent practical to remove the electronics and ancillary equipment for optimum storage and consolidation.

The majority of GTCC sources and devices are transported from the manufacturer in 7A (type A) containers. However, some sources have been transported in type B containers due the higher

a. U.S. Nuclear Regulatory Commission, NRC Sealed Source Survey, December 1990.

b. Information derived from Troxler Electronic Labs, Inc., Cornwallis Road/Alexander Drive, Research Triangle Park, NC 27709; Nuclear Environmental Engineering, Inc., Registry of Sealed and Device Sources, Amersham/Searle, personal communication with Brian Baker; Parkwell Laboratories, Inc., Registry of Sealed Sources and Devices; New England Nuclear, Registry of Sealed Sources and Devices; CPN Company, 2830 Howe Road, Martinez, CA 94533; Monsanto Research Corporation, Dayton Laboratory, Dayton, Ohio.

specific activity and associated dose rate/transport index (TI) and special shielding requirements. The majority of the type A containers are retained by the users, while the type B containers are usually returned to the manufacturer. Generally, the users and manufacturers retain the shipping containers for storage and transport of the devices/sources.

3.2 Types, Quantities, and Sizes of Seal Sources

The NRC survey represented a sampling ($\approx 25\%$) of their licensed users and manufacturers of sealed sources. These sample data were then extrapolated for both specific and general licensed users, including those under Agreement State licensing, to project the current estimate of GTCC source quantities. The NRC survey was focused primarily on the users who maintain a specific license for use of sealed source material. General license projections were based on NRC knowledge and data base records. The best estimate of quantities of GTCC sealed sources and devices that are currently in use or inactive is in excess of 92,500. The uncertainty of these results is projected to be within 10%. Table H-3 provides a summary of the major quantities of GTCC sealed sources and devices.

Table H-3. Major GTCC LLW sealed sources and devices

TYPES	SPECIFIC LICENSE (> 27,500)	GENERAL LICENSE (> 65,500)
TRU-bearing Sources		
Well Logger Units	$\approx 2,000$	-
Density/Moisture Gauges	$\approx 14,200$	< 1 %
Gamma Gauges	≈ 200	< 1 %
Pacemakers	≈ 130	-
X-Ray Fluorescent	< 1 %	-
Smoke Detectors	-	< 1 %
Non-TRU-bearing Sources		
Test Gauges (Industrial/Analytical)	$\approx 3,300$	> 45 %
Fixed Gauges (Gamma/Beta, Thickness)	≈ 800	> 45 %
Calibration Sources	≈ 480	< 1 %
Category I Sources	≈ 450	-
Irradiator Sources	≈ 440	-
Cat. II, III, IV		

The major quantity of sealed sources and devices currently in use under specific licenses is identified with well logger units, density/moisture gauges, fixed/test gauges. The major sealed sources under the general (broad) license are of the fixed/test gauge type.

Sealed sources and devices present a problem, because most sources are small but require much larger containers for shielding, handling, deployment, storage, and eventual disposal. The actual

A typical moisture/density gauge source contains Cs-137 and americium-241. The maximum activities typically are 10 mCi of Cs-137 and 60 mCi of Am-241. The maximum threshold activity used by the NRC to determine these types of sources as GTCC LLW is 27 mCi for Am-241 and other transuranics and 910 mCi of Cs-137. A typical moisture/density source is 1.5 cm in length and 1.0 cm in outside diameter ($\approx 1.2 \text{ cm}^3$).

A typical oil well logger source contains Am-241, Pu-238, and Cs-137. The maximum activities range from 10–60 Ci of transuranic material and 20 mCi of Cs-137. The sizes of oil well logging sources are larger than the other types of sources and typically are 2.54 cm in diameter and 7.5 cm in length (38.6 cm^3).

A typical GTCC test/fixed gauge source contains Cs-137, Sr-90, Am-241, and CM-244. The sources, for the most part, are smaller and contain significantly smaller quantities of source material than the oil well logger sources. The test/fixed gauges typically are 0.8 cm in length and 4 cm in outside diameter, with a total volume of 10 cm^3 . Appendix B provides a detailed description of the major sources currently used for analytical and industrial applications. The range in sources sizes is identified below.

Packaged sealed sources are normally contact handled, but the sealed source capsule or jacket is not normally contact handled. The source is usually located within a container or device that is shielded to provide direct contact handling. The size of these devices is usually a function of the type of source, the shielding and containment requirements, and the manner in which the unit is to be deployed. These devices are usually transported in special cases or packages provided by the manufacturers to protect the devices. Table H-4 is a summary of the range of sizes for the individual source, the shielded container, the device, and the transport package for each of the major sealed sources identified within the NRC survey.

Table H-4. Source and package size ranges

MOISTURE/DENSITY GAUGES ($\approx 14,000$)			
Source size	0.75 cm ³	-	4.6 cm ³
Shield size	3,670 cm ³	-	4,140 cm ³
Gauge size	14,680 cm ³	-	20,700 cm ³
Package size	96,500 cm ³	-	104,950 cm ³
FIXED/TEST GAUGES ($\approx 4,100 + 66,000 +$)			
Source size	4.24 cm ³	-	13.5 cm ³
Shield size	198 cm ³	-	655 cm ³
Gauge/Container size	5,678 cm ³	-	9440 cm ³
Package size	20,600 cm ³	-	32,180 cm ³
OIL WELL LOGGING UNITS ($\approx 2,000$)			
Source size	14.3 cm ³	-	38.6 cm ³
Shield size	1073 cm ³	-	1442 cm ³
Container size	3,785 cm ³	-	5,680 cm ³
Package size			
Type A	20,600 cm ³	-	32,180 cm ³
Type B	71,800 cm ³	-	71,800 cm ³

3.3 Sealed Source Transport to DOE

The identified sealed sources and devices (i.e., moisture/density gauges, fixed/test gauges and well logging units) represent over 90% of the potential surplus sources to be shipped to DOE. These sources could be transported to DOE in three potential configurations. The choice of configuration would be influenced by DOE requirements for acceptance, U.S. Department of Transportation/NRC transport regulations, and the cost of source disassembly.

The smallest practical size of a source/device unit that requires transport to the DOE could be represented by a stripped down, shielded/contact-handled configuration. This configuration would require the source owner to use a vendor/manufacturer's service in completely dismantling the source to the extent that only the shielded container would remain. This would eliminate all electronics and support components of the sealed source. This disassembly could reduce the total volume of a device/source unit by as much as a factor of 25. While this may be the smallest practical size possible for handling purposes, some difficulty may exist in qualifying or demonstrating this component suitable as a type A package for transport. The requirement for some additional overpack for meeting the Transport Index and related requirements is probable. However, this configuration would minimize additional handling and disassembly at the DOE site prior to storage.

with all external deployment/ancillary equipment removed. This would require some vendor/manufacture service effort but could also significantly reduce the size of the source equipment transported to DOE. This configuration should be readily acceptable as a type A package with minimal requirements for overpacks, etc. Based on size comparison, it is estimated that the volume of sealed source equipment shipped in the minimum device/source configuration would be reduced by a factor of 5 when compared to the original fully packaged equipment.

The third configuration is a that of a fully packaged source in the original transport package. This configuration, while acceptable for receipt and transport requirements, would possibly represent the most inefficient volume of material for storage and handling purposes and require substantial disassembly for optimum storage and eventual disposal.

3.4 Sealed Source Packaging Factors and Scenarios

The development of a simplified packaging factor for all sealed sources and devices is difficult at best because of the range of sizes and configurations represented by diverse types of surplus sources that exist. By narrowing down the predominant types of sources in the previous section, it is possible to develop a weighted packaging factor for modeling purposes.

3.4.1 Base Case Scenario

In the base case scenario, all sealed sources and devices will be loaded into high-level waste (HLW) or equivalent canisters in the smallest shielded/contacted-handled configuration practical. This base case is highly probable in that no remote handling should be required to process the sources, and substantial volume reduction is accomplished. As sources are accumulated, each canister will be loaded completely to minimize any void volume.

The packaging factor for this scenario is developed based on the ratio of shielded source volume to the unshielded source volume and any additional voids created in the package. Since three sources are considered as the major contributors, a weighted value was used to determine the average packaging factor, shown in Table H-5.

Table H-5. Base case scenario packaging factor

$$PF = \{VOLUME_{Shielded} / VOLUME_{Source}\} \{1 / VOID_{Fraction}\}$$

Weighted Values:

Moisture/Density 4140 / 4.6 = 900	× # of Units 14,000	=	Volume Units 12.6 E06
Fixed/Test Gauges 655/13.5 = 48.5	70,000		3.39 E06
Well logger 1442/38.6 = 37.4	<u>2,000</u>		<u>.07 E06</u>
TOTALS	86,000		16.06 E06

$$VOL_{Shielded} / VOL_{Source} = 16.06 \text{ E06} / .086 \text{ E06} = 190$$

Void Fraction is estimated to be $\approx 45\%$ to 50% based on small cylindrical and spherical-shaped sources with shields having no significant wall effects due to the relatively large high-level waste canisters (0.057 m by 0.45 m).

$$PACKAGING \text{ FACTOR}_{BASE \text{ CASE}} = \{190 \times 1 / 0.5\} = 380$$

3.4.2 Low Case Scenario for Sealed Sources

In the low case scenario, all sealed sources and devices will be loaded into HLW canisters or their equivalent in their smallest unshielded conditions. This assumes that all contacted-handled units established in the base case scenario are further volume reduced by extracting the capsuled source from the shielded device. This scenario is also considered probable because some vendors-/manufacturers (moisture/density and well logger devices) have disassembled units down to the source/capsule size for both recycling and storage while awaiting disposal options.

The packaging factor for this low case scenario is based on the loading of small (0.75 cm^3 to 38.6 cm^3) cylindrical sources into the relatively larger cylindrical HLW canister. It is assumed that the loading is done with no special orientation of the source. The PF will therefore be primarily influenced by the void fraction created by the random loading of the canister. As with the base case scenario, no special factors for wall effect voids are considered likely. The void fractions are expected to be

$\approx 45\%$ to 50% .

$$\text{Packaging Factor} = \{1 / VOID_{FRACTION}\} = 1/0.5 = 2$$

3.4.3 High Case Scenario for Sealed Sources

The high case scenario considers options ranging from direct loading of the devices and holders in storage and disposal containers (very inefficient) to selective loading of contact-handled sources and devices in special containers for transport. One vendor/manufacturer^a is currently developing a method/process of loading multiple sources into a 7A drum (55 gal) that has special pipe sleeves and shielding for collecting surplus sources. The container design varies as a function of the source (i.e., neutron, gamma, etc.) and the associated shielding/Transport Index considerations. Based on nominal sealed source considerations, it is expected that from 10 to 20 sources may be loaded in this configuration. The immediate value to DOE is that these transport containers could be used as temporary storage containers and reduce the initial handling and storage impact at the receiving facility.

The high case scenario is defined as the loading of sources into a modified transport and storage container that can be transported and used by DOE for storage and eventual downloading to an HLW canister. This scenario is also highly probable and applicable to the three major source configurations identified in Table H-6.

The packaging factor is based on loading up to 20 sources in a 55-gal drum. The sources can be loaded in an unshielded condition with added shielding to the container or a shielded condition requiring minimal shielding by the drum.

Table H-6. High case scenario for sealed sources packaging factor

$PF_{\text{High Case}} = \{ \text{Drum}_{\text{Vol}} / \# \text{ of Sources} \times \text{Vol}_{\text{Source}} \} \{ \text{Weight Factor} \}$		
Moisture/Density	× # of Units	= Volume Units
208,206/ 20 × 4.6 = 2,263	14,000	3.17 E0
Fixed/Test Gauges		
208,206/ 20 × 13.5 = 771	70,000	5.4 E07
Well logger		
208,206/ 20 × 38.6 = 270	<u>2,000</u>	<u>0.05 E07</u>
TOTALS	86,000	8.62 E07
$PF_{\text{High Case}} = 8.62 \text{ E07} / .086 \text{ E06} = 1000$		

The NRC staff recommended that an additional high case scenario be considered based on the return of the sealed sources in their original package configuration. The scenario appears appropriate when considering the majority of the fixed/test gauges are encased in steel cases with the sources and primary shields retained within. The shielding for this configuration includes both the case body and the shielded source within the case. An assumption was made based on measured

a. Personal communications with Bill Walker, Troxler Electronics, Inc., January 1991.

package scenario was developed for loading these source device cases in either a 55-gal drum or a standard box design. The drum configuration identified a maximum of 12 cases per drum. The comparable volume box configuration would hold 22 cases per box. The packaging factor of 12 cases of sources per drum is 1285 ($208,206 / 12 \times 13.5$). The packaging factor of 22 cases per box is 700 ($208,206 / 22 \times 13.5$). The average PF of these two configurations is 990. These packaging factors bound the range for the high case scenario and demonstrate that the weighted average approach should be acceptable for modeling purposes in determining sealed source package volumes for DOE projections.

4. ALL OTHER GTCC LLW

The volume of GTCC LLW generated by commercial facilities will not be the volume disposed of by the DOE. The volume of the waste containers into which the GTCC LLW will be placed will constitute the disposal volume. Empty volume in a waste container will be minimized by the generators if minimization is made cost-effective compared to costs for storage and disposal of the empty space. Although costs for DOE acceptance of GTCC LLW are not firm, a high cost per unit volume of waste container will tend to move generators toward minimizing void volume in a waste container for most GTCC LLW forms.

Since many GTCC LLW forms are similar to Class A, B, and C wastes that can be disposed of in licensed disposal facilities, and since disposal costs at such facilities are high enough to require waste generators to pursue volume reduction operations, the current industry practices for volume reduction of Class A, B, and C wastes will likely be considered for treatment of GTCC LLW. If there is a relatively higher cost for disposal of GTCC LLW by the DOE than for commercial disposal of low-level wastes, additional technologies for volume reduction of GTCC LLW that are cost-effective would likely be introduced in the future. Likewise, some currently available but uneconomical technologies may become cost-effective in the future.

A generic base case for a packaging factor for all other GTCC LLW is a factor of 1.0. This value reflects the uncertainty in the volume reduction technologies that can be used for GTCC LLW and in the amounts of the different types of materials that will be generated. From current projections of all other unpackaged GTCC LLW, the mix of such wastes will be primarily compactible and noncompactible materials. A reasonable assumption appears to be that the volume decrease from compaction, incineration, and other volume reduction technologies will offset the volume increase from packaging noncompactible materials into waste containers. With this assumption, the gains from volume reduction technologies would offset the historical increase in the overall volume of wastes just due to the packaging of the wastes.

The generic low value for the packaging factor for all other GTCC LLW is 0.5, or a prediction that a net reduction will occur from the amount generated to the amount disposed. This value reflects the significant advances in supercompaction and radioactive waste incineration that have recently become available for low-level waste treatment in the United States. With volume reduction factors of 300 to 1 for incineration and 10 to 1 for supercompaction, treatment of significant amounts of the GTCC LLW by these methods would bring the packaging factor down considerably. The proposed value reflects that these new technologies will be available and used to some extent by the GTCC LLW generators.

The generic high value for the packaging factor for all other GTCC LLW is expected to be 1.5, or a packaging efficiency of 0.67. This value represents an inability to use volume reduction technologies to a significant extent for GTCC LLW due to the radionuclide concentrations that make the waste GTCC LLW in the first place. Commercial treatment centers may be unwilling to allow treatment because of the potential for significant contamination of expensive equipment. Also, GTCC LLW generators may not have the volume of GTCC LLW to justify installation of volume reduction equipment. The proposed value reflects the unique nature of this waste and the potential difficulty in processing it economically.

4.1 Decontamination Resins

The two principal sources of decontamination resins include those generated in purification demineralizers for PWR and BWR reactor coolant systems and processing resins for major system and component decontamination solutions. The reactor coolant system purification resins are not normally GTCC LLW. However, if significant levels of failed fuel exist and resins systems are operated until resin exchange capacity is exhausted, then GTCC LLW resins are possible. Normally, these resin systems are replaced because of high differential pressure or high radiation fields, which normally preclude this as GTCC LLW.

The other principal source of GTCC LLW ion-exchange resins is from processing highly concentrated chemical decontamination solutions. These solutions contain oxidizers and chelating agents that readily remove the activated corrosion products (i.e., Ni-58 and Nb-94) from the surface oxides of the metals. The resins act as a concentration mechanism to highly concentrate these relatively long-lived corrosion products.

In determining the packaging scenarios and related packaging factors, the following assumptions are considered valid. First, the generators will use those economic incentives to maximize volume reduction technologies once it is concluded that the ion-exchange resins are determined to be GTCC LLW. Secondly, operational constraints on maximum concentration limits, capacities, and transport packages (Type B) will not restrict, influence, or constrain loading decisions. Thirdly, the packaging scenario will be directly influenced by the treatment method and any associated volume reduction, which will provide greater or lesser package loading efficiency. The following represents the best reasonable estimate of packaging scenarios/factors for spent resins.

4.1.1 Packaging Factors for Spent Resins (Base, High, and Low Case)

Unless driven by high cost, waste generators would normally be expected to minimize processing and handling of GTCC LLW spent resins. The primary requirement for transporting and receiving wet solids, which is consistent with current requirements, would be to dewater the resins and load them into a high-integrity container (HIC). This effort would only require a sluicing system and dewatering system with minimal requirements at the generator site. Typically, this is done with most resin operations identified in the industry. The loading into a high-integrity container satisfies most isolation and stabilization requirements and should be suitable for DOE acceptance. The loading efficiencies of this type of operation are only influenced by the resins' packaging efficiency and are not affected by the size of the container. The void fraction for small spherical bead resin is approximately 40-50%. Since the volume of resins identified in the source projection studies included the void volumes, a base case projection for spent resins was assumed to be dewatered resins with a volume packaging equivalent to unity.

$$\text{PACKAGING FACTOR}_{\text{Base Case}} = 1 \quad (\text{Dewatered Resins})$$

The high case scenario assumed that requirements for stabilizing the resins other than in a high-integrity container would be imposed. Options for stabilizing the material with no volume reduction could include cementation or solidification by encapsulation with polymer materials. The technology for organic encapsulation typically shows no volume change relative to the dewatered resin volume. For cementation, the typical volume ratio of resins to cement ranges from 1:1 to 1:0.6. A reasonable high case scenario for expected packaging factors would be a solidified resin matrix with volume increase of 2.

PACKAGING FACTOR_{High Case} = 2 (Dewatered and Solidified Resins)

The low case scenario assumes that substantial economic incentives exist to maximize volume reduction (VR). This scenario concluded that the latest technology would be used to sinter the resins with compaction to a VR of 10:1. The resins would then be stabilized through encapsulation or cementation, with a volume increase to 2:1. The resultant packaging factor for the low volume case would be a net volume change of 0.2.

PACKAGING FACTOR_{Low Case} = 0.2 (Sintered and Cemented Resins)

4.2 Cartridge Filters

The principal sources of filter cartridges are BWR and PWR control rod drive filters, seal water injection and return filters, fuel pool, reactor coolant makeup, and purification system filters. Typically, filters range in size from 0.1 ft³ (4 in. OD by 12 in. length) to 1.0 ft³ (8 in. OD by 3 ft length). The predominant radionuclide contaminants are activated metals (C¹⁴, Nb⁹⁴, Tc⁹⁹) and TRU sources (Cm²⁴², Pu²⁴¹, other traces transuranics). Cartridge filters generally contain metal endcaps for stability which are retained with the discarded filter.

4.2.1 Packaging Factors for Cartridge Filters (Base, High, and Low Case)

Generator decisions on packaging and transport to DOE will be influenced by least cost/impact processes for volume reduction, handling, and packaging. Cartridge filters do not require any special processing for collection and handling other than dose and contamination control. Consistent with current practices, it is reasonable to conclude that spent filters would be dewatered and packaged in high integrity containers with no consideration of volume reduction. Currently, highly contaminated filters (e.g. Class B or C) are randomly placed (dumped) in open HICs with flange closure lids. The loading efficiency with these various-sized filters in a relatively larger HIC container would be expected to range from .25 to 0.5. This scenario is considered the most likely (base case) because of minimal cost and accepted practice for storage and disposal.

Packaging Factor_{Base Case} = 2 (Dewatered Filters/No VR)

In the high volume case, where encapsulation (e.g., cementation or equivalent) might be a requirement, it was assumed that the encapsulation material would be used to fill the voids created by the filter cartridges. The packaging efficiency for this scenario is considered to be the worst case, (0.25) with a PF of 4.

Packaging Factor_{High Case} = 4 (Encapsulated Filters/No VR)

In the low volume case, it is assumed that significant economic incentive exists for the generator to look at available technologies and service organizations to volume reduce and package the waste filters. Current technology services available include shredding and encapsulation. A volume reduction of 10:1 or greater is expected.

Packaging Factor_{Low Case} = 0.1 (Shredding and Encapsulation)

4.3 Aqueous Liquids

The primary sources of aqueous liquids include americium salts and UO_2 dissolved in nitric acid and hydrofluoric acid. These liquids generally contain both transuranic and activated/fission product components. The treatment and processing considerations for transport, storage, and disposal usually require solution neutralization and solidification. The transuranic activities in these liquids influence their GTCC LLW classifications. Since GTCC LLW liquids cannot be easily transported, the generators will usually use best available technologies for volume reduction and stabilization. These technologies and treatment services include ion-exchange processing (similar to treatment of decontamination solutions), evaporation, and direct solidification. These aqueous liquids, after pH adjustment (neutralization), are readily amenable to all demonstrated treatment technologies.

4.3.1 Packaging Factors for Aqueous Liquids (Base, High, and Low Case)

One current method for concentrating and stabilizing the aqueous liquids is the use of ion-exchange resins. These resins could be processed as discussed in Section 4.1. Based on the expected high dissolved salt content and the associated resin capacity, the volume reduction in this process would not be expected to be much greater than 10:1. Because of the relatively high concentration loading of TRU and salts, it may be expected that these resins would be placed into an HIC or solidified by cementation. The volume increase with cementation is ≈ 1.6 , and the overall processing/packaging volume reduction would be expected to be ≈ 0.2 . The container used to solidify the material would be the same as the disposal container, and therefore the packaging factor would be 0.2.

$$\text{PACKAGING FACTOR}_{\text{Base Case}} = 0.2 \quad (\text{Ion-Exchange/Solidified Liquids})$$

The high volume case for processing liquids assumes that a base solution neutralization step will be required, which could double the volume of the waste liquid prior to further processing. This step would then be followed by a cementation process, which could increase the volume by 1.6. The net increase in volume for the high volume scenario is 3:1.

$$\text{PACKAGING FACTOR}_{\text{High Case}} = 3.0 \quad (\text{pH Adjusted/ Solidified})$$

The low volume case scenario assumes that the liquids will be evaporated to near-dryness and then solidified in cement. Because the acid and dissolved salt content is relatively high (1–2 wt%), the concentration factor for evaporation would be less than 100. The cementation step would increase the volume, with a net volume reduction of approximately 10.

$$\text{PACKAGING FACTOR}_{\text{Low Case}} = 0.1 \quad (\text{Evaporation/Solidified Liquids})$$

4.4 Solidified Liquids

The sources of GTCC LLW solidified liquids include TRU bearing solids from fuel fabrication facilities and highly concentrated process liquids from BWR and PWR facilities. It is assumed that no additional requirement exists for processing this material. Additionally, no special restrictions exist for packaging, transporting, storage, and disposal of this material. The physical form of this material is considered as a solidified monolith or as solid granular material.

4.4.1 Packaging Factors for Solidified Liquids (Base, High, and Low Case)

Since this material exists in a solid form, there may be no economic incentive to further process and volume reduce this waste form. The most probable packaging and handling scenario would be to leave this material in the existing transport package or transfer this material into transport containers and load to the maximum extent practical. The material would be expected to fill each container with minimal voids and loading inefficiencies. The packaging volume would be similar to the source projection volume.

$$\text{PACKAGING FACTOR}_{\text{Base Case}} = 1 \quad (\text{Dry with No Further Processing})$$

The high volume case assumes that the solid form of material will be encapsulated because of stabilization requirements. Consistent with previously used methods, this could include cementation, with a volume increase factor of 1.6. The waste form would be formed in the storage, transport, and disposal container with minimal voids.

$$\text{PACKAGING FACTOR}_{\text{High Case}} = 1.6 \quad (\text{No Volume Reduction/ Encapsulation})$$

The low volume case assumes the use of the best available and demonstrated technology. It is also assumed that the solid material is not amenable to incineration or similar processes. The low case scenario considers the vitrification/melting of solidified liquids, with a potential volume reduction of 20%. The waste form would be formed in the storage and disposal container, with optimum packaging efficiency.

$$\text{PACKAGING FACTOR}_{\text{Low Case}} = 0.8 \quad (\text{Vitrification})$$

APPENDIX A

Figures and Tables for Activated Metals

Figure H-1. Arrangement of LCB sections in a weight constrained, cylindrical cross section waste container.

Example: Model TITAN truck cask without basket
 CB dimensions: ID = 148 in. Thk = 2.5 in.

Figure H-2. Arrangement of LCB sections in a weight-constrained, square cross section waste container.

Example: PWR spent fuel cask with basket in place
CB dimensions: ID = 148 in. Thk = 2.5 in.

Figure H-3. Arrangement of LCB sections in a volume constrained, cylindrical cross section waste container.

Example: Model TITAN truck cask without basket
 CB dimensions: ID = 148 in. Thk = 2.5 in.

Figure H-4. Arrangement of LCB in a volume constrained, square cross section waste container.

Example: PWR spent fuel cask with basket in place
 CB dimensions: ID = 148 in. Thk = 2.5 in.

Figure H-5. Arrangement of BWR control rod blades in a cylindrical cross section waste container.

Example: Model TITAN truck cask without basket
 Flattened CRB envelope dimensions: 2 in. by 10 in.

Figure H-6. Arrangement of BWR control rod blades in a cylindrical cross section waste container.

Example: Model IF-300 rail cask without basket
 Flattened CRB envelope dimensions: 2 in. by 10 in.

Figure H-7. Arrangement of BWR control rod blades in a square cross section waste container.

Example: Model GA-4 truck cask without basket
 Flattened CRB envelope dimensions: 2 in. by 10 in.

Figure H-8. Arrangement of BWR control rod blades in a square cross section waste container.

Example: Model BR-100 rail cask without basket
 Flattened CRB envelope dimensions: 2 in. by 10 in.

← WASTE CONTAINER INSIDE LENGTH = 17" →

Figure H-9. Arrangement of BWR control rod blades in a square cross section waste container.

Example: Model GA-9 truck cask without basket
CRB envelope dimensions: 10 in. diameter

Table H-7. Packaging efficiencies for the Model TITAN Truck Cask without a spent fuel basket

Packaging Efficiencies for the Model TITAN Truck Cask without a Spent Fuel Basket						
Weight Constrained						
Cylindrical Waste Containers						
Payload weight is maximized up to 6,235 pounds limit for the cask						
Case Number	Lower Core Barrel Dimensions inches		Packaging Efficiency for entire LCB	Number of waste containers needed	Packaging Efficiency for a full waste container	Weight of a full waste container pounds
	Outer Diameter	Inner Diameter	Thickness			
1	153	148	2.5	12	0.222	6,235
2	152	148	2	10	0.213	6,235
3	154	148	3	15	0.225	6,235
4	163	158	2.5	13	0.214	6,235
5	162	158	2	11	0.225	6,235
6	164	158	3	16	0.206	6,235
7	143	138	2.5	12	0.214	6,235
8	142	138	2	9	0.208	6,235
9	144	138	3	14	0.221	6,235
					0.214	6,235

Table H-8. Packaging efficiencies for the Model IF-300 Rail Cask without a spent fuel basket

Packaging Efficiencies for the Model IF-300 Rail Cask without a Spent Fuel Basket						
Weight Constrained						
Cylindrical Waste Containers						
Payload weight is maximized up to 21,000 pounds limit for the cask						
Case Number	Lower Core Barrel Dimensions inches		Packaging Efficiency for entire LCB	Number of waste containers needed	Packaging Efficiency for a full waste container	Weight of a full waste container pounds
	Outer Diameter	Inner Diameter	Thickness			
1	153	148	2.5	4	0.267	21,000
2	152	148	2	3	0.284	21,000
3	154	148	3	4	0.321	21,000
4	163	158	2.5	4	0.285	21,000
5	162	158	2	3	0.303	21,000
6	164	158	3	5	0.274	21,000
7	143	138	2.5	3	0.332	21,000
8	142	138	2	3	0.265	21,000
9	144	138	3	4	0.300	21,000
					0.333	21,000

Table H-9. Packaging efficiencies for the Model 125-B Rail Cask without an inner containment vessel

Packaging Efficiencies for the Model 125-B Rail Cask without an Inner Containment Vessel									
Weight Constrained									
Cylindrical Waste Containers									
Payload weight is maximized up to 57,565 pounds limit for the cask									
Case Number	Lower Core Barrel Dimensions inches	Inner Diameter	Thickness	Packaging Efficiency for entire LCB	Number of waste containers needed	Packaging Efficiency for a full waste container	Weight of a full waste container pounds		
1	Outer Diameter 153	148	2.5	0.283	2	0.527	57,565		
2	152	148	2	0.451	1	0.451	50,119		
3	154	148	3	0.340	2	0.527	57,565		
4	163	158	2.5	0.302	2	0.527	57,565		
5	162	158	2	0.481	1	0.481	53,043		
6	164	158	3	0.363	2	0.527	57,565		
7	143	138	2.5	0.264	2	0.527	57,565		
8	142	138	2	0.421	1	0.421	47,194		
9	144	138	3	0.318	2	0.527	57,565		

Table H-10. Packaging efficiencies for casks with PWR spent fuel baskets

Packaging Efficiencies for Casks with PWR Spent Fuel Baskets									
Weight Constrained									
Square Cross-section Waste Containers									
Payload weight is maximized up to 1,515 pounds limit for each cask cavity position									
Case Number	Lower Core Barrel Dimensions inches	Thick	Packaging Efficiency for entire LCB	Number of waste containers needed	Packaging Efficiency for a full waste container	Length of the side of a waste container inches	Weight of a full waste container pounds	Number of sections of LCB per full waste container	
1	Outer Dia. 153	2.5	0.597	43	0.609	6.7	1,498	2	
2	152	2	0.469	40	0.472	7	1,306	2	
3	154	3	0.350	81	0.352	7	1,025	1	
4	163	2.5	0.609	45	0.610	6.7	1,499	2	
5	162	2	0.466	43	0.472	7	1,307	2	
6	164	3	0.351	86	0.352	7	1,026	1	
7	143	2.5	0.600	40	0.609	6.7	1,497	2	
8	142	2	0.461	38	0.471	7	1,305	2	
9	144	3	0.348	76	0.351	7	1,024	1	

Table H-11. Packaging efficiencies for the GA-4 Truck Cask without a spent fuel casket

Packaging Efficiencies for the GA-4 Truck Cask without a spent fuel basket									
Weight Constrained Square Cross-section Waste Containers									
Payload weight is maximized up to 8,760 pounds limit for the cask									
Case Number	Lower Core Barrel (LCB) Dimensions-----		Packaging Efficiency for entire LCB	Number of waste containers needed	Packaging Efficiency for a full waste container	Length of the side of a waste container inches	Weight of a full waste container pounds	Number of sections of LCB per full waste container	
	Outer Dia.	Inner Dia.	Thck						
1	153	148	2.5	7	0.704	0.734	8,755	5	
2	152	148	2	6	0.663	0.711	8,392	6	
3	154	148	3	9	0.610	0.678	8,762	4	
4	163	158	2.5	8	0.657	0.735	8,762	5	
5	162	158	2	6	0.708	0.712	8,397	6	
6	164	158	3	9	0.659	0.682	8,710	4	
7	143	138	2.5	7	0.657	0.733	8,747	5	
8	142	138	2	6	0.619	0.711	8,387	6	
9	144	138	3	8	0.641	0.677	8,753	4	

Table H-12. Packaging efficiencies for the 125-B Rail Cask with an inner containment vessel replaced by a special square cavity basket

Packaging Efficiencies for the 125-B Rail Cask with an Inner Containment Vessel replaced by a Special Square Cavity Basket									
Weight Constrained Square Cross-section Waste Containers									
Payload weight is maximized up to 51,808 pounds limit for the cask or cavity size (34")									
Case Number	Lower Core Barrel (LCB) Dimensions inches		Thickness	Packaging Efficiency for entire LCB	Number of waste containers needed	Packaging Efficiency for a full waste container	Length of the side of a waste container inches	Weight of a full waste container pounds	Number of sections of LCB per full waste container
	Outer Diameter	Inner Diameter							
1	153	148	2.5	0.490	2	0.807	34	48,533	12
2	152	148	2	0.781	1	0.781	34	47,065	15
3	154	148	3	0.590	2	0.805	34	48,384	10
4	163	158	2.5	0.522	2	0.807	34	48,537	12
5	162	158	2	0.417	2	0.810	34	48,678	15
6	164	158	3	0.629	2	0.805	34	48,397	10
7	143	138	2.5	0.457	2	0.807	34	48,532	12
8	142	138	2	0.729	1	0.729	34	44,142	14
9	144	138	3	0.551	2	0.804	34	48,372	10

Table H-13. Packaging efficiencies for the BR-100 Rail Cask with a spent fuel basket replaced by a special square cavity basket

Packaging Efficiencies for the BR-100 Rail Cask with a Spent Fuel Basket replaced by a Special Square Cavity Basket									
Weight Constrained Square Cross-section Waste Containers									
Payload weight is maximized up to 40,000 pounds limit for the cask									
Case Number	Lower Core Barrel Dimensions--inches--	LCB Efficiency for entire	Number of waste containers needed	Packaging Efficiency for a full waste container	Length of the side of a waste container inches	Weight of a full waste container pounds	Number of sections of LCB per full waste container		
	Outer Dia.	Inner Dia.	Thick						
1	153	148	2.5	0.605	0.818	39,882	11		
2	152	148	2	0.498	0.788	37,291	13		
3	154	148	3	0.700	0.785	39,925	9		
4	163	158	2.5	0.645	0.818	39,892	11		
5	162	158	2	0.532	0.848	39,961	14		
6	164	158	3	0.747	0.786	39,942	9		
7	143	138	2.5	0.565	0.817	39,873	11		
8	142	138	2	0.465	0.788	37,290	13		
9	144	138	3	0.654	0.785	39,909	9		

Table H-14. Packaging efficiencies for the Model TITAN Truck Cask without a spent fuel basket

Packaging Efficiencies for the Model TITAN Truck Cask without a Spent Fuel Basket									
Volume Constrained Cylindrical Waste Containers									
Case Number	Lower Core Barrel Dimensions--inches--	LCB Efficiency for entire	Number of waste containers needed	Packaging Efficiency for a full waste container					
	Outer Diameter	Inner Diameter	Thickness						
1	153	148	2.5	0.667	4	0.747			
2	152	148	2	0.709	3	0.713			
3	154	148	3	0.642	5	0.715			
4	163	158	2.5	0.711	4	0.750			
5	162	158	2.2	0.567	4	0.600			
6	164	158	3	0.685	5	0.716			
7	143	138	2.5	0.662	3	0.744			
8	142	138	2	0.623	4	0.719			
9	144	138	3	0.600	5	0.713			

Table H-15. Packaging efficiencies for the Model IF-300 Rail Cask without a spent fuel basket

Packaging Efficiencies for the Model IF-300 Rail Cask without a Spent Fuel Basket						
Volume Constrained			Cylindrical Waste Containers			
Case Number	Lower Core Barrel (LCB) Dimensions inches		Packaging Efficiency for entire LCB	Number of waste containers needed	Packaging Efficiency for a full waste container	
	Outer Diameter	Inner Diameter	Thickness			
1	153	148	2.5	2	0.493	0.812
2	152	148	2.2	1	0.786	0.786
3	154	148	3	2	0.593	0.793
4	163	158	2.5	2	0.526	0.812
5	162	158	2.2	2	0.419	0.821
6	164	158	3	2	0.633	0.795
7	143	138	2.5	2	0.460	0.810
8	142	138	2.2	1	0.734	0.734
9	144	138	3	2	0.554	0.790

Table H-16. Packaging efficiencies for the Model 125-B Rail Cask withoug an inner containment vessel

Packaging Efficiencies for the Model 125-B Rail Cask without an Inner Containment Vessel						
Volume Constrained			Cylindrical Waste Containers			
Case Number	Lower Core Barrel (LCB) Dimensions inches		Packaging Efficiency for entire LCB	Number of waste containers needed	Packaging Efficiency for a full waste container	
	Outer Diameter	Inner Diameter	Thickness			
1	153	148	2.5	1	0.565	0.565
2	152	148	2	1	0.451	0.451
3	154	148	3	1	0.681	0.681
4	163	158	2.5	1	0.603	0.603
5	162	158	2.2	1	0.481	0.481
6	164	158	3	1	0.726	0.726
7	143	138	2.5	1	0.526	0.526
8	142	138	2.2	1	0.421	0.421
9	144	138	3	1	0.636	0.636

Table H-17. Packaging efficiencies for casks with PWR spent fuel baskets

Packaging Efficiencies for Casks with PWR Spent Fuel Baskets									
Volume Constrained				Square Cross-section Waste Containers					
Case Number	Lower Core Barrel Dimensions inches		LCB (LCB)	Packaging Efficiency for entire LCB	Number of waste containers needed	Packaging Efficiency for a full waste container	Length of the side of a waste container inches	Weight of a full waste container pounds	Number of sections of LCB per full waste container
	Outer Diameter	Inner Diameter	Thickness						
1	153	148	2.5	0.498	34	0.511	8.25	1,910	2
2	152	148	2	0.614	22	0.616	8.25	2,248	3
3	154	148	3	0.600	34	0.612	8.25	2,235	2
4	163	158	2.5	0.502	36	0.512	8.25	1,911	2
5	162	158	3	0.600	24	0.616	8.25	2,250	3
6	164	158	3	0.604	36	0.612	8.25	2,238	2
7	143	138	2.5	0.510	31	0.511	8.25	1,908	3
8	142	138	3	0.600	21	0.615	8.25	2,247	2
9	144	138	3	0.595	32	0.611	8.25	2,233	2

Table H-18. Packaging efficiencies for the GA-4 Truck Cask without a spent fuel basket

Packaging Efficiencies for the GA-4 Truck Cask without a Spent Fuel Basket									
Volume Constrained				Square Cross-section Waste Containers					
Case Number	Lower Core Barrel Dimensions inches		LCB (LCB)	Packaging Efficiency for entire LCB	Number of waste containers needed	Packaging Efficiency for a full waste container	Length of the side of a waste container inches	Weight of a full waste container pounds	Number of sections of LCB per full waste container
	Outer Diameter	Inner Diameter	Thickness						
1	153	148	2.5	0.748	5	0.775	17.56	12,072	6
2	152	148	2	0.746	4	0.829	17.56	12,871	8
3	154	148	3	0.751	6	0.772	17.56	12,035	5
4	163	158	2.5	0.665	6	0.775	17.56	12,081	6
5	162	158	3	0.795	4	0.829	17.56	12,878	8
6	164	158	3	0.686	7	0.773	17.56	12,046	5
7	143	138	2.5	0.698	5	0.774	17.56	12,062	6
8	142	138	3	0.696	4	0.828	17.56	12,863	8
9	144	138	3	0.701	6	0.771	17.56	12,023	5

Table H-19. Packaging efficiencies for the 125-B Rail Cask with an inner containment vessel replaced by a special square cavity basket

Packaging Efficiencies for the 125-B Rail Cask with an Inner Containment Vessel replaced by a Special Square Cavity Basket														
Volume Constrained				Square Cross-section Waste Containers										
Case Number	Lower Core Barrel Dimensions inches		Thickness	Packaging Efficiency for entire LCB	Number of waste containers needed	Packaging Efficiency for a full waste container	Length of the side of a waste container inches	Weight of a full waste container pounds	Number of sections of LCB per full waste container					
	Outer Diameter	Inner Diameter												
1	153	148	2.5	0.490	2	0.807	34	48,533	12					
2	152	148	2	0.781	1	0.781	34	47,065	15					
3	154	148	3	0.590	2	0.805	34	48,384	10					
4	163	158	2.5	0.522	2	0.807	34	48,537	12					
5	162	158	2	0.417	2	0.810	34	48,678	15					
6	164	158	3	0.629	2	0.805	34	48,397	10					
7	143	138	2.5	0.457	2	0.807	34	48,532	12					
8	142	138	2	0.729	1	0.729	34	44,142	14					
9	144	138	3	0.551	2	0.804	34	48,372	10					

Packaging Efficiencies for all cases

Sorted in descending order of packaging efficiency for the entire LCB

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Table H-20. continued

Packaging Efficiencies for all cases

Sorted in descending order of packaging efficiency for the entire LCB

Table Number	Case Number	Core Dimensions inches		Thickness	Packing Efficiency for entire LCB	Number of waste containers needed	Packing Efficiency for a full waste container
		Lower Diameter	Outer Diameter				
4	7	143	138	2.5	0.600	40	0.609
8	9	144	138	2.5	0.600	5	0.713
11	1	153	148	2.5	0.597	43	0.609
13	9	144	138	3	0.595	32	0.611
13	3	154	148	3	0.593	2	0.793
13	3	154	148	3	0.590	2	0.805
13	3	154	148	3	0.590	2	0.805
8	5	162	158	2	0.567	4	0.600
10	7	153	148	2.5	0.565	1	0.565
17	7	143	138	2.5	0.554	2	0.817
9	9	144	138	3	0.551	2	0.790
6	9	144	138	3	0.551	2	0.804
13	9	144	138	3	0.551	2	0.804
17	5	162	158	2	0.532	2	0.848
9	4	163	158	2	0.526	2	0.812
10	7	143	138	2.5	0.526	1	0.526
13	4	163	158	2.5	0.522	1	0.807
6	4	163	158	2.5	0.522	2	0.807
11	7	143	138	2.5	0.510	31	0.511
11	4	163	158	2.5	0.502	36	0.512
11	1	153	148	2.5	0.498	34	0.511
17	2	152	148	2	0.498	2	0.788
9	1	153	148	2.5	0.493	2	0.812
6	1	153	148	2.5	0.490	2	0.807
13	1	153	148	2.5	0.490	2	0.807
10	1	162	158	2	0.481	1	0.481
3	5	152	148	2	0.481	1	0.481
4	2	162	158	2	0.469	40	0.472
4	5	162	158	2	0.466	43	0.472
7	8	142	138	2	0.465	2	0.788
4	8	142	138	2	0.465	38	0.788
4	7	142	138	2	0.461	2	0.471
9	7	143	138	2.5	0.460	2	0.810
6	7	143	138	2.5	0.457	2	0.807
13	7	143	138	2.5	0.457	2	0.807
10	7	152	148	2	0.451	1	0.807
3	2	152	148	2	0.451	1	0.451
10	3	142	138	2	0.451	1	0.451
3	8	142	138	2	0.421	1	0.421
3	8	142	138	2	0.421	1	0.421
9	5	162	158	2	0.419	2	0.821
5	5	162	158	2	0.417	2	0.810
13	5	162	158	2	0.417	2	0.810
3	6	164	158	3	0.363	2	0.527
4	6	164	158	3	0.351	86	0.527
4	3	154	148	3	0.350	81	0.352
4	4	144	138	3	0.348	76	0.352
3	3	154	148	3	0.340	2	0.351
2	7	143	138	2.5	0.332	3	0.527
2	3	154	148	3	0.321	4	0.333
3	9	144	138	3	0.318	2	0.333
3	9	144	138	3	0.318	2	0.527

Table H-20. continued

packaging Efficiencies for all cases								
Sorted in descending order of packaging efficiency for the entire LCB								
Table Number	Case Number	Lower Core Dimensions inches	Barrel Diameter	Inner Diameter	Thickness	Packaging Efficiency for entire LCB	Number of waste containers needed	Packaging Efficiency for a full waste container
2	5	162	162	158	2	0.303	3	0.333
3	4	163	163	158	2.5	0.302	2	0.527
2	9	144	144	138	3	0.300	4	0.333
2	4	163	163	158	2.5	0.285	4	0.333
2	2	152	152	148	2.2	0.284	3	0.333
3	1	153	153	148	2.5	0.283	2	0.527
2	6	164	164	158	3	0.274	5	0.333
2	1	153	153	148	2.5	0.267	4	0.333
2	8	142	142	138	2.2	0.265	3	0.333
3	7	143	143	138	2.5	0.264	2	0.527
1	1	153	153	148	2.5	0.222	12	0.225
1	1	142	142	138	2.5	0.221	9	0.225
1	1	163	163	158	2.5	0.219	13	0.225
1	4	144	144	138	3	0.214	14	0.225
1	9	154	154	148	3	0.214	15	0.225
1	3	164	164	158	3	0.213	16	0.225
1	6	152	152	148	2.5	0.213	10	0.225
1	2	143	143	138	2.5	0.208	12	0.225
1	7	162	162	158	2	0.206	11	0.225
1	5							

APPENDIX B

Description of Sealed Sources and Devices

X-ray fluorescence

Technique

Primary radiation from the radioisotope source excites atoms of the elements present in the sample, removing electrons from the sub-shells round the nucleus. X-rays characteristic of each element are emitted as electrons from the outer shells and move to fill the gaps created in inner shells. The shell from which the electron is removed determines the series of X-rays produced. The intensity of the X-ray is indicative of the concentration of the particular element in the sample. Since radioisotopes emit specific radiations, a limitation results on the range of elements whose characteristic X-rays can be excited. Thus a series of nuclides is employed in order that excitation of all elements from silicon to uranium can be achieved.

Geometry

Applications

- Alloy analysis for checking stock, scrap sorting and checking components.
- In mining, analysis of material excavated from pits, and cores, chippings and slurries from drilling operations.
- Analysis of electroplating solutions.
- General chemical analysis.

Source

Capsule dimensions and Safety performance testing

Capsule	Overall diam. 'A' mm	Active diam. 'B' mm	Window thickness 'C' mm	Safety performance testing AHS/ISO Classification	IAEA special form	NRC model no.
x 10/2	8	4.2	0.2-0.25	77C64545	GB/3/S	AMC D2
x 11	10.8	7.2	0.2-0.25	77C64444	GB/4/S	AMC D3
x 11/1	10.8	8.0	0.2-0.25	77C64444	GB/4/S	AMC D3

Moisture gauging

Technique

Fast neutrons emitted by the source are moderated by collision with hydrogen atoms in moisture contained in the material. These moderated or thermal neutrons are detected by a neutron detector (usually a boron trifluoride (BF_3) proportional counter) to give a measure of the concentration of hydrogen atoms.

Geometry

Applications

- Soil moisture content for agricultural and construction use.
- Moisture content of materials in silos.
- Continuous moisture content gauging in raw material supplies e.g. gravel, wood chips etc.

Source

Safety performance testing

Capsule	ANSI/ISO classification	IAEA special form	NRC model no.
X.2	77E66544	GB.8.S	AMN PE1
X.21	77C65545	GB.43.S	AMN PE5

Thickness gauging

Transmission thickness technique

The source and the detector are placed on opposite sides of the material to be measured. Gamma or beta radiation transmitted through the sample is then directly related to the sample thickness, provided the density of the material is constant.

Geometry

Applications

Gamma gauging

- Thickness gauging of sheet metal, glass, plastic, and rubber at thickness too large for beta sources, e.g., greater than 500mg/cm².
- Belt weighing, giving mass (kg/m²) flowing on conveyor belt.

Beta Gauging

- Thickness gauging of thinner plastics, thin sheet metal, rubbers, textiles and paper, e.g., 1 – 1000mg/cm².
- The weighing of cigarettes
- Measurement of dust and pollutant levels on filter paper samples e.g., 0.1 – 200 mg/m² dust.

Source

Capsule dimensions and Safety performance testing

Capsule	Overall diam. 'A'	Active diam. 'B'	Window thickness 'C'	Safety performance testing ANSI/ISO classification	IAEA special form	NRC model no.
X 34	36	31	0.25-0.3	77E6444	GB/107-S	AMC 30
X 35	45	40	0.25-0.3	77E6444	GB/121-S	AMC 50

Thickness gauging

Beta backscatter thickness technique

The intensity of beta radiation which is scattered back from thin samples is related to sample thickness and atomic number.

Geometry

Gamma backscatter thickness technique

The intensity of backscattered radiation from the sample is measured to give sample thickness or mean atomic number. Used for the measurement of substances of low Z for which transmission measurements are not sufficiently sensitive.

Geometry

Applications

- The thickness gauging of paper, plastic and rubber on steel rolls.
- The measurement of a coating thickness on a substrate, virtually any coating/substrate combination providing there is sufficient difference in density or atomic number. Coating range < 1–100μm depending on source and materials.

Source

Applications

Thickness gauging

- Measurement of light alloys, glass, plastics, rubbers for which beta sources are not suitable e.g., greater than 500 mg/cm², and access only available from one side. e.g., tube wall thickness gauging

Capsule dimensions and Safety performance testing

Capsule	Overall diam. 'A' mm	Active diam. 'B' mm	Window thickness 'C' mm	Safety performance testing ANSI/ISO classification	IAEA special form model no.	NRC model no.
x 91	10.9	7.5	0.2–0.25	77C64444	GB/38/S	AMC 16
x 92	15	12	0.2–0.25	77C64444	GB/39/S	AMC 17
x 93	30	25	0.2–0.25	77C64444	GB/40/S	AMC 19
x 97	22	18	0.2–0.25	77C64444	GB/41/S	AMC 18

Level gauging

Gamma switching technique

The transmission of gamma radiation through a container is affected by the level of the contents. The intensity of the transmitted radiation is measured and used to activate switches when pre-set intensity levels are reached.

Geometry

Storage hopper level control

Can/package contents monitoring

Applications

Storage hopper level control

- Switch can be used to operate high level and low level alarms or pump switch control.

Oil well logging

Neutron porosity logging technique

Fast neutrons emitted by a neutron source are slowed down by the formation and may undergo three interactions: 1) inelastic scatter, 2) elastic scatter, 3) absorption. Therefore, by collision with hydrogen atoms in the formation the neutron will be moderated to thermal or epithermal energies where it is soon captured by hydrogen nuclei and emits a secondary gamma ray. The detection of these three interactions by using different types of neutron detectors (BF_3 (thermal), ^3He (epithermal)) can be used to determine the hydrogen content of the formation. Since the majority of hydrogen in a formation generally exists within the pore space, the neutron flux will then be related to the porosity.

Geometry

Applications

- Determination of formation hydrogen content
- Formation porosity for oil and mineral logging

Source

APPENDIX C

Description of 55-Gal Drum for Sealed/Device Source for Storage and Transport

(Design Concepts by Bill Walker, Troxler Electronics Inc.)

Clamp

Lid

Top Plug

Top View - Barrel

DOT TYPE "A"
SPEC 7A
55 Gal Drum

Shielding Detail

Note: This section made to fit
Type "A" Spec. 7A 55 gal Barrel

Top View

PVC Source Tubes

Detail of Void Space
In Shielding

Shielding

Detail PVC Source Tube

Note: ¹⁾ Like sources inserted
in tube & logged.

²⁾ Metal Canister 1' l'd
be of similar den.
Normal Form Sources

Greater-Than-Class C Low-Level Radioactive Waste Characterization

Appendix I: Impact of Concentration Averaging Low-Level Radioactive Waste Volume Projections

*Greater-Than-Class C Low-Level Waste
Management Program*

August 1991

**GREATER-THAN-CLASS C
LOW-LEVEL WASTE CHARACTERIZATION**

**Appendix I: Impact of Concentration Averaging
Low-Level Radioactive Waste
Volume Projections**

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ABSTRACT

This study provides a quantitative framework for bounding unpackaged greater-than-Class C low-level radioactive waste types as a function of concentration averaging. The study defines the three concentration averaging scenarios that lead to base, high, and low volumetric projections; identifies those waste types that could be greater-than-Class C under the high volume, or worst case, concentration averaging scenario; and quantifies the impact of these scenarios on identified waste types relative to the base case scenario. The base volume scenario was assumed to reflect current requirements at the disposal sites as well as the regulatory views. The high volume scenario was assumed to reflect the most conservative criteria as incorporated in some compact host state requirements. The low volume scenario was assumed to reflect the 10 CFR Part 61 criteria as applicable to both shallow land burial facilities and to practices that could be employed to reduce the generation of Class C waste types.

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APPENDIX I: IMPACT OF CONCENTRATION AVERAGING LOW-LEVEL RADIOACTIVE WASTE VOLUME PROJECTIONS

1. INTRODUCTION

The requirements for low-level radioactive waste classification are defined in 10 CFR Part 61.¹ This regulation, specifically Paragraph 61.55, requires that waste suitable for disposal must fall into one of three classes: A, B, or C. Waste is determined to fall into one of these three classes by comparing radionuclide concentrations within the waste to limiting concentrations for particular radionuclides. The Class C designation is the most restrictive. Waste suitable for commercial disposal must not exceed those concentrations designated for Class C waste. Waste that exceeds these concentrations is referred to as greater-than-Class C (GTCC) low-level waste (LLW). The concentrations of radionuclides within a package are determined by dividing the estimated radioactivity by the volume of the waste material within the package. This calculation assumes that the waste material within the package is homogeneous, and the calculated concentrations are averaged over the packaged waste mass. In some cases, relatively high-concentration waste materials are combined with low-concentration waste materials in the same package, and the concentration calculations are based on the combined radioactivity divided by the combined waste volumes. This practice is referred to as concentration averaging. How this practice is applied to different types of waste affects the 10 CFR Part 61 Class C classification and the amounts of GTCC LLW that will be generated.

EG&G Idaho, Inc., has contracted with WMG, Inc., to provide a quantitative framework for bounding unpackaged GTCC low-level types as a function of the concentration averaging scenario. This information will be used as input to the U.S. Department of Energy (DOE) GTCC projection computer model. Three objectives were specified in order to perform this evaluation:

- Define the three concentration averaging scenarios that lead to base, high, and low volumetric projections.
- Identify those waste types that could be GTCC under the high volume, or worst case, concentration averaging scenario.
- Quantify the impact of these scenarios on identified waste types relative to the base case scenario.

To reasonably bound the volumes of GTCC LLW that could be generated over the 1990 to 2055 time span, both current and prospective LLW practices were considered. The base volume scenario was assumed to reflect current requirements at the LLW disposal sites as well as the current views of the DOE and the U.S. Nuclear Regulatory Commission (NRC) pertaining to the subject. The high volume scenario was assumed to reflect the most conservative criteria as incorporated in some compact host state requirements. The low volume scenario was assumed to reflect the 10 CFR Part 61 criteria as applicable to both shallow land burial facilities and to practices that could be employed to reduce the generation of Class C, hence GTCC, waste types.

A master list of potential GTCC low-level types was prepared under the high volume scenario. Those waste types are already included in the DOE data base; the current basis for their consideration was reviewed and revised to reflect current waste type characteristics and practices.

The basis for consideration of whether a waste type was to be included or excluded from the GTCC category was subsequently developed.

The master GTCC LLW type list provided the basis for consideration of volumetric variances as a function of concentration averaging scenarios. Initially, each waste type was evaluated to determine the extent to which it would be GTCC LLW under the different concentration averaging scenarios. The basis for these relative volumes was defined, and such volumes were normalized relative to the base volume case values. These normalized values could then be applied to the DOE model volumes to account for the various concentration averaging scenarios.

Section 2 presents the summary and conclusions of this evaluation. The three concentration averaging scenarios considered and their bases are described in Section 3. Those waste types identified as potential GTCC and the basis for such designation are presented in Section 4. The extent to which potential GTCC LLW types will be GTCC LLW for the three concentration averaging scenarios considered is presented in Section 5. The study conclusions are also presented in Section 5. Section 6 presents the study references.

2. EVALUATION SUMMARY AND CONCLUSIONS

To reflect current and prospective practices, wastes were divided into the following four categories before concentration averaging scenarios were defined:

- **Activated Metals**--Activated metals consist of the nonfuel-bearing components that are activated through exposure to neutron flux. Two types were considered: (a) components that are periodically replaced during routine operations, and (b) structural components in the reactor vessel that are removed at decommissioning.
- **Process Wastes**--Process wastes consist of the wet wastes generated from cleanup of liquids containing soluble and insoluble radioactive constituents.
- **Contaminated Solids**--Contaminated solids consist of plastic, paper, cloth, glass, wood, and metal that are contaminated.
- **Sealed Sources**--Sealed sources consist of small capsules, usually stainless steel, which encapsulate relatively high concentrations of a single radionuclide.

Three different concentration averaging scenarios were considered for these categories. The base volume case reflects current practices at the Barnwell, South Carolina, disposal site. The low volume case primarily reflects current practices at the Richland, Washington, disposal site since it will remain open beyond 1992. The high volume case is based on those compact regions that might prohibit any form of concentration averaging.

As expected, the results indicate that volumes of GTCC LLW will vary significantly with each concentration averaging scenario. However, even under the high volume scenario, the extent to which GTCC LLW could be generated is less than potential GTCC LLW types previously estimated. Thus, this evaluation showed fewer GTCC LLW types and does not support the implicit assumption in previous estimates that all wastes generated from a particular type of GTCC waste stream will be GTCC. In many cases, only a portion of a particular waste stream is potential GTCC, even under the high volume scenario.

Table I-1 lists the GTCC LLW types identified by this study and presents estimates of that portion, as a percentage of the total amount generated, which could be GTCC for each concentration averaging scenario.

As shown in Table I-1, the relative variances between the high volume scenario and the base case scenario covered different ranges for each waste category and type. This range varied from unity, meaning no change between scenarios, to a factor of 100, meaning the high volume case would generate 100 times the volume of the base case. The relative variances between the low volume scenario and the base case scenario ranged from unity to zero. Except for some decommissioning activated metals and three types of Transuranic-bearing sealed sources, no GTCC LLW would be generated for the low volume case.

These results have varying degrees of confidence. The information used for routinely generated activated metals is based on a significant data base of actual components with a high confidence level. This is not the case for decommissioning activated metals. Few such components have been actually characterized and disposed of to date, yet some of this work is now being initiated at power reactors

Table I-1. Waste summary--relative GTCC waste volumes

WASTE TYPE SUMMARY RELATIVE GTCC WASTE VOLUMES			
	Percent GTCC For High Case	Percent GTCC For Base Case	Percent GTCC For Low Case
ACTIVATED METALS			
1.BWR Operations			
- Control Rod Blades--Control Cell	25	10	0
- Control Rod Blades--Normal	10	1	0
- LPRM Strings	75	10	0
- Dry Tubes	100	10	0
2. PWR Operations			
- Thimble Plug Assemblies	25	1	0
- Instrument Strings	75	10	0
- Primary Sources	100	100	100
3.BWR Decommissioning			
- Core Shroud	100	100	100
- Orificed Fuel Supports	10	1	0
4.PWR Decommissioning			
- Core Shroud	100	100	100
- Lower Core Barrel	25	1	0
- Thermal Shield	25	1	0
- Lower Grid Plate	100	100	75
5.Other Reactor internals			
-Research	100	100	0
-Test	100	1	0
PROCESS WASTES			
1. BWR & PWR Decon Resins			
- Full RCS with Fuel In	25	1	0
- Full RCS with Fuel Out	10	1	0
2.Cartridge Filters			
- BWR Control Rod Drive	50	10	0
- Other BWR & PWR	25	1	0
CONTAMINATED SOLIDS			
	25	10	1
SEALED SOURCES			
1.TRU Bearing Sources			
- Gamma Gauges	100	100	50
- X-Ray Fluorescent	100	100	0
- Well Logging	100	100	100
- Moisture Gauges	100	100	0
- Smoke Detectors	100	100	0
- Pacemakers	100	100	0
2.Other Sources			
- Med. Teletherapy	100	100	0
- Gamma Gauges	50	1	0
- Beta Gauges	50	1	0
- Well Logging	50	1	0
- Calibration	50	25	0
- Gamma Irradiators I,II,III,IV	100	100	0
- Gas Chromatography	100	1	0
- Beta Eye Applicators	100	1	0

and research facilities. While there are more extensive technical data sources available, detailed analyses of these data sources was beyond the scope of this study. Thus, the confidence level for the estimates on decommissioning activated metals is much lower than that for other metals.

The information used for spent resins and filter cartridges is also based on actual waste data. Except for the full reactor coolant system (RCS) decontamination resins that have not as yet been generated, this information has a high confidence level. The information used for contaminated solids has the lowest degree of confidence. The readily available DOE information was minimal.

For sealed sources, NRC reference materials were used almost exclusively. The information used to identify source characteristics has a high degree of confidence. However, data on the numbers of sources by type and the distribution of activities for source types is not readily available. Moreover, the assumption that sources would be encapsulated in 55-gal drum-size containers with approved stabilization media is overly simplistic. In addition to ignoring economic considerations, this also assumes that typical source users are aware of the approved stabilization methods and can readily employ them. Since sealed sources are the most widely used and likely GTCC LLW types, a more detailed evaluation of this waste category appears warranted.

The dependence of GTCC LLW generation on concentration averaging scenarios will also lead to some "orphan" wastes. In a compact-based regulatory environment, there is a high degree of certainty that waste could be GTCC in one compact region, meet Class C limits in another, and not meet the DOE definition of GTCC. Steps should be taken to address these types of issues to provide some degree of certainty to potential GTCC LLW type management practices.

3. CONCENTRATION AVERAGING

The concentrations of radionuclides within a package are determined by dividing the estimated radioactivity by the volume of the waste material within the package. This calculation assumes that the waste material within the package is homogeneous, and the calculated concentrations are averaged over the packaged waste mass. In some cases, relatively high concentration wastes are combined with low concentration wastes in the same package, and the related concentration calculations are based on the combined radioactivity divided by the combined waste volumes. This practice is referred to as concentration averaging. The application of this practice to different types of waste affects the 10 CFR Part 61 waste classification. For the purposes of this study, concentration averaging significantly affects which wastes will meet or exceed NRC waste form Class C limits.

To reflect current practices and disposal site criteria, potential GTCC LLW were divided into four major categories:

- **Activated Metals**--Activated metals consist of components comprised (primarily stainless steel) that are activated by neutrons within a reactor core. These wastes mostly contain activation product radionuclides.
- **Process Wastes**--Process wastes consist of wet wastes produced during cleanup of radioactive liquids in the form of ion-exchange resins and filter cartridges. These wastes primarily contain fission product and transuranic radionuclides.
- **Contaminated Solids**--Contaminated solids consist of dry wastes that are radioactive due to surface contamination. These wastes are produced during cleanup at special facilities and primarily contain transuranic radionuclides (TRU).
- **Sealed Sources**--Sealed sources consist of radioactive wastes with high radioactivity encapsulated in small volumes. Some of these wastes contain long-lived fission products, while others contain transuranic radionuclides.

To provide a framework for DOE volume projections, three concentration averaging scenarios were defined:

- **High Volume Case**--This covers disposal criteria that reflect prospective practices leading to high GTCC LLW volumes.
- **Base Volume Case**--This covers disposal criteria that reflect current practices leading to moderate GTCC LLW volumes.
- **Low Volume Case**--This covers disposal criteria that reflect reasonable practices leading to low GTCC LLW volumes.

This section describes the basis for integrating the disposal requirements with the different waste categories in order to define practicable and realistic concentration averaging scenarios. These scenarios would provide the concentration averaging bounding framework for DOE's GTCC volume projection model.

3.1 General NRC Guidelines

The documents published by the NRC allude to concentration averaging, but they do not specifically address the concept. The regulation in 10 CFR 61.55(a)(8) provides that "the concentration of a radionuclide may be averaged over the volume of the waste." When this statement is read in conjunction with the premise that 10 CFR Part 61 deals with Class A, B, and C packaged wastes, concentration averaging is permissible. The NRC Branch Technical Position (BTP)² on waste classification expands on 61.55(a)(8), indicating that the principal consideration is whether or not the distribution of radionuclides is reasonably homogeneous. In one example, the BTP indicates that a concentrated source being combined with trash can be averaged over the trash volume. In another example, the use of stabilization media as part of the concentration calculation for filters and sealed sources is discussed. Limits on concentration averaging are not defined, but the BTP recognizes that, for hard-to-detect radionuclides, which are the primary drivers of classification status, the concentration must be accurate to within a factor of 10.

Additional guidance is provided in a draft regulatory guide on waste classification^a that was never published. This draft indicates that concentration averaging, or blending, is acceptable provided the concentrations of the blended wastes are within several orders of magnitude.

While each of the commercial disposal sites has applied both these 10 FR Part 61 rules and the NRC guidelines differently, concentration averaging of packaged waste is routinely done.

3.2 Current Practices

Current practices arise from the requirements in 10 CFR Part 61, the NRC Branch Technical Position on waste classification (Reference 2), together with the currently licensed disposal site application of these requirements to specific commercial disposal sites. Current practices by waste category include:

- Activated Metals--Different types of components and different pieces from different types of components are usually packaged in the same disposal liner. In some cases, activated metals can also be encapsulated with a media like cement. The classification status of stabilized metal can sometimes be determined by dividing the metal activity by the total volume plus the stabilization media.
- Process Wastes--Different batches of resin are combined in the same package. Different types of filter cartridges are also combined in the same package. In some cases, spent filters and spent resins, or spent filters and activated metals, are combined in the same package. Both spent resins and filter cartridges can be stabilized with a media like cement. The classification status of the stabilized waste can usually be determined by dividing the waste activity by the total volume, including the stabilization media.
- Contaminated Solids--Current practices vary widely for these dry wastes and can include both incineration and compaction to reduce disposal volumes. Compaction can increase concentrations by a factor of 10, while incineration can increase concentrations by a factor of 100. Wastes are segregated primarily by physical form, with no differentiation between

a. U. S. Nuclear Regulatory Commission, "Regulatory Guide on Waste Classification and Manifest Reporting," draft, February 1986.

waste classes. Where these wastes are incinerated, the incinerator ash is sometimes stabilized. The classification status of the stabilized ash can usually be determined by dividing the ash activity by the total volume, including the stabilization media.

- **Sealed Sources**--By design, sealed sources contain high radioactivity in small volumes. This usually leads to very high concentrations of 10 CFR Part 61 radionuclides. Under NRC guidelines, these sealed sources can be stabilized or encapsulated in media like cement. When encapsulated, the classification status is determined by dividing the activity of the sources by the volume, including the stabilization media.

Practices vary with waste category. Moreover concentration averaging criteria, vary between both the waste categories and among the commercial disposal sites. Current criteria at the commercial disposal facilities are discussed below.

3.2.1 Barnwell Disposal Facility

The Barnwell Disposal Facility received about 68% of the commercial low-level radioactive waste disposed of in 1989.³ Hence, its application of 10 CFR Part 61 requirements to concentration averaging has a significant impact on how wastes are currently packaged for disposal. The criteria applied to commingling wastes varies with each waste category as follows:

- **Activated Metals**--Packages containing activated metals must satisfy what is commonly referred to as the "rule of 10." Application of this rule allows concentration averaging among the same types of components but limits concentration averaging to a factor of 10 among individual components. For example, local power range monitor (LPRM) strings with relatively high concentrations of 10 CFR Part 61 radionuclides cannot be combined with control rod blades of lower concentrations unless the concentrations of both 10 CFR Part 61 Table 1 and Table 2 radionuclides for the two components are within a factor of 10 of each other. Once the rule of 10 is satisfied, the classification status can be determined by dividing the total activity of all the components by the total volume.
- **Process Wastes**--Where different batches of resins or filters are placed in the same package, the classification status is determined by dividing the total activity of the waste by the total volume of the waste. Where different types of waste, (e.g., resins and filters) are combined in the same package, the classification status of each waste type is determined separately. The radioactivity of each waste type is divided by the volume for each waste type and documented separately.

The addition of cement increases the volume of the waste and correspondingly reduces the concentration of 10 CFR Part 61 radionuclides. Under NRC guidelines, the classification status of the stabilized waste can properly be determined by dividing the total activity by the total volume plus the stabilization media. Where spent resins are stabilized, the total volume can be applied. Where filter cartridges are stabilized, classification status must be determined by dividing the total activity by the volume of the filter cartridges only.

- **Contaminated Solids**--Packaged waste both with and without volume reduction is considered homogeneous, and the entire volume is used for classification determination. Incinerator ash may require stabilization, and the criteria applicable to resins would apply.

- Sealed Sources--Requests for disposal of encapsulated sources are addressed on a case-by-case basis. Generally, sources containing non-TRU radionuclides with activities up to 10 Ci and sources containing TRU radionuclides with activities up to 50 mCi are approved for encapsulation.

3.2.2 Richland Disposal Facility

The Richland Disposal Facility received about 25% of the commercial low level radioactive waste disposed in 1989 (Reference 3). Its application of 10 CFR Part 61 requirements to concentration averaging is similar to the NRC guidelines. The criteria applied to commingling waste varies with each waste category as follows:

- Activated Metals--The classification status of packages containing activated metals can be determined by dividing the total activity of all the components by the total volume of the components in the package.
- Process Wastes--In all cases of process wastes, the classification status of spent resins and filter cartridges is determined by dividing the total activity of the waste by its total volume.

Both spent resins and filter cartridges can also be stabilized. The classification status of the stabilized waste can be determined by dividing the total activity by the total volume, including the stabilization media.

- Contaminated Solids--The packaged wastes are considered homogeneous, and the entire volume is used for this classification determination.
- Sealed Sources--Non-TRU-bearing sources can be encapsulated, and the encapsulation media volume is included in the concentration calculation. Prior approval on the encapsulation media and method is required for sources containing TRU radionuclides.

3.2.3 Beatty Disposal Facility

The Beatty Disposal Facility in Nevada received about 7% of the commercial low-level radioactive waste disposed in 1989 (Reference 3). Its application of 10 CFR Part 61 requirements to concentration averaging is identical to that applied by the Richland facility, except that prior approval is not required for encapsulated TRU sources.

3.3 Prospective Practices

Beyond 1992, concentration averaging practices will be based on the criteria in effect within the different compact regions and sited states. Therefore, it is likely that disposal criteria will vary to a much greater extent than at present. The Richland site will continue to be the disposal site for the Northwest Compact, and, absent any new NRC criteria, current practices will remain essentially unchanged. However, the Appalachian Compact Region, hosted by Pennsylvania, intends to prohibit any form of concentration averaging. Some of the other compact regions and states that represent relatively large volumes of low-level radioactive waste, such as the Southeast, Midwest, and New York, have not yet stated their positions on concentration averaging. It is unlikely, however, that

these regulatory positions will differ from those already defined and bounded by the Northwest Compact Region and the Appalachian Compact Region.

The "no concentration averaging" position taken by Pennsylvania is the most restrictive. Applying this position to the different waste categories discussed above would lead to the following criteria:

- **Activated Metals**--Each activated metal component would be classified under 10 CFR Part 61 as if it were the only component in the disposal package. Radionuclide concentrations would be determined by dividing the component activity by the component volume.
- **Process Solids**--Where resins and filter cartridges are stabilized, the volume of the stabilization media would be excluded from the classification determination.
- **Contaminated Solids**--Commingling of contaminated solids would be prohibited. Use of the stabilization media volume to classify incinerator ash would probably be unacceptable.
- **Sealed Sources**--Encapsulation of sealed sources in a stabilization media practice would probably be acceptable. However, radionuclide concentrations would be determined by dividing the source activity by the source volume.

3.4 GTCC Projection Scenarios

The bounding of the volumes of GTCC LLW that could be generated requires some assumptions concerning prospective disposal criteria. For the purposes of determining the impact of alternative concentration averaging criteria, the following bounding conditions were assumed:

- **Base Volume Case**--The criteria currently in effect at the Barnwell site as described in Section 3.2.1 above represent the current practices for most waste (68% by volume in 1989) that could be GTCC. These criteria were selected to represent the base volume case. They provide for use of the rule of 10 for activated metals, use of the stabilization media volume for resins and incinerator ash, commingling of contaminated solids, and encapsulation of non-TRU-bearing sources with activities up to 10 Ci.
- **Low Volume Case**--The criteria currently in effect at the Richland site as described in Section 3.2.2 above are consistent with the current NRC reading of 10 CFR Part 61, representing current practices for some waste (25% by volume in 1989) that could be GTCC. These criteria were selected to represent the low volume case. They provide for determination of Class C status based on disposal liner activity and volume, including the stabilization media, and encapsulation of all sources.
- **High Volume Case**--The prospective criteria for compacts that are similar to the Appalachian Compact Region, as described in Section 3.3 above, represent a very restrictive reading of 10 CFR Part 61 requirements. These criteria were selected to represent the high volume case. They provide for no concentration averaging. Contaminated solid commingling, use of stabilization media volumes, and encapsulation would all be prohibited.

Table I-2 summarizes these cases and criteria as they apply to the four different categories of waste discussed.

Table I-2. GTCC volume projection scenarios

GTCC VOLUME PROJECTION SCENARIO 3			
WASTE CATEGORY	LOW VOLUME CASE	BASE VOLUME CASE	HIGH VOLUME CASE
1. ACTIVATED METALS	Activity in liner divided by waste volume in liner Stabilization media volume INCLUDED	Activity in liner divided by waste volume in liner Concentrations of components/ types within a FACTOR of 10	Activity in component divided by component volume
2. PROCESS WASTE	Activity in liner divided by waste volume in liner Stabilization media volume INCLUDED for Resins and Filters	Activity in liner divided by waste volume in liner Stabilization media volume INCLUDED for Resins NOT for Filters	Activity in liner divided by waste volume in liner Stabilization media volume NOT INCLUDED
3. CONTAMINATED SOLIDS	Activity in liner divided by waste volume in liner Commingling with non TRU Solids	Activity in liner divided by waste volume in liner Commingling with lower activity TRU Solids	TRU bearing solids activity divided by solids volume No commingling
4. SEALED SOURCES	Activity in liner divided by waste volume in liner Encapsulation media volume INCLUDED for ALL	Activity in source divided by source volume Encapsulation media volume INCLUDED for non TRU sources < 10Ci	Activity in source divided by source volume Encapsulation media volume EXCLUDED

4. POTENTIAL GTCC LLW TYPES AND BASIS FOR CLASSIFICATION

Any facility that produces LLW containing 10 CFR Part 61 Class C radionuclides can theoretically generate GTCC LLW. Fortunately, the types of facilities that routinely generate LLW containing these radionuclides also operate in environments that preclude or limit the potential to generate GTCC LLW. As discussed in Section 3, current and prospective disposal criteria have evolved to the extent that wastes can be divided into four main categories: (a) activated metals, (b) process wastes, (c) contaminated solids, and (d) sealed sources. Each of these categories has dominant 10 CFR Part 61 Class C radionuclides, the concentrations of which determine the extent to which wastes in the category approach or exceed Class C limits. The 10 CFR Part 61 Class C radionuclides of interest for each waste category are as follows:

- Activated Metals--The activation product radionuclides Ni-59, Ni-63, Nb-94, and C-14.
- Process Wastes--The fission product radionuclides Cs-137 and Sr-90, the activation product radionuclide Ni-63, and transuranics.
- Contaminated Solids--Transuranics.
- Sealed Sources--Two distinct groups, dependent on source type: the transuranics Am-241, CM-244, Pu-238, and Pu-239; and the radionuclides Cs-137, Sr-90, and Ni-63.

The DOE conducted a survey of LLW generators to identify waste types that could be GTCC LLW. The results of this survey are summarized in a draft DOE report.^b Table I-3 lists the wastes that the DOE indicated could exceed Class C limits and the radionuclide, or set of radionuclides, of interest. The listing in Table I-3 indicates that the waste could potential exceed Class C limits under the high volume, or most restrictive, concentration averaging case defined previously in Section 3.4. This table also indicates whether these wastes were considered GTCC for this evaluation.

The wastes identified on Table I-3 and the basis for their inclusion in this table as potential GTCC LLW are discussed by category in this section. These discussions also address the other wastes considered and the basis for their exclusion.

4.1 Activated Metals

Activated metals consist of the nonfuel-bearing components that are activated through exposure to neutron flux within light water, research, and test reactors. Emphasis is placed on light water reactors (LWR). Two sources were considered: (a) those components that are periodically replaced during normal operations, and (b) the structural components in the reactor vessel that are removed at decommissioning. These components are manufactured of stainless steels, high nickel bearing inconels, zircalloys, or a combination of all these materials. Some research reactors have aluminum components. Exposure to neutrons activates certain base metals that produce the 10 CFR Part 61 activation product radionuclides. Of particular interest in metals are nickel, which produces the Table 1 radionuclide Ni-59 and the Table 2 radionuclide Ni-63; niobium impurities in the base metals, which

b. U.S. Department of Energy, *Projected Volumes, Activities, and Other Waste Characteristics of Commercial Greater Than Class C Low-Level Radioactive Waste*, draft, Idaho National Engineering Laboratory, September 1989.

Table I-3. Potential GTCC LLW and radionuclides of interest

POTENTIALLY GTCC WASTES AND RADIONUCLIDES OF INTEREST		
Waste Category and Type	GTCC Potential	Part 61 Class C Nuclides of Interest
1. ACTIVATED METALS		
a. BWR Operations	YES	Ni-59,Nb-94,C-14,Ni-63
b. PWR Operations	YES	Ni-59,Nb-94,C-14,Ni-63
c. BWR Decommissioning	YES	Ni-59,Nb-94,C-14,Ni-63
d. PWR Decommissioning	YES	Ni-59,Nb-94,C-14,Ni-63
e. Other Reactor Internals	YES	Ni-59,Nb-94,C-14,Ni-63
2. PROCESS WASTES		
a. Spent Resins	YES	TRU,Cs-137,Sr-90
b. Cartridge Filters	YES	TRU,Cs-137,Sr-90,Ni-63
c. Aqueous Liquids	NO	TRU,Cs-137,Sr-90
d. Organic Liquids	NO	TRU
e. Zeolite Ion Exchange Media	NO	TRU
f. Sludge	NO	TRU,Cs-137,Sr-90
3. CONTAMINATED SOLIDS		
a. Non Metals	YES	TRU
b. Metals	YES	TRU
4. SEALED SOURCES		
a. TRU Bearing	YES	TRU
b. Other	YES	Cs-137,Sr-90,Ni-63

produce the Table 1 radionuclides C-14 from nitrogen impurities in stainless steel; and Nb-94. The Table 1 radionuclide Tc-99, from activation of molybdenum, is also present but yet insignificant. 10 CFR Part 61 classification is driven by either the sum of the Table 1 radionuclide concentrations Ni-59 plus Nb-94 plus C-14, or by the concentration of Ni-63 under Table 2. The Table 2 sum of the fractions drives the 10 CFR Part 61 Class C limits for most stainless steels and inconels, while the Table 1 radionuclide Nb-94 drives the Class C limits for zircalloys. For primary sources, the activation product radionuclides are secondary to the transuranic neutron source material in the component.

Type 304 stainless steels contain 8 to 11 wt% nickel plus trace ppm quantities of niobium. Zircalloys also contain trace quantities of niobium plus trace quantities of nickel. Inconels have the greatest potential for becoming GTCC LLW since they contain 52 to 72 wt% nickel plus 0.01 to 5.5 wt% niobium.

For routinely discharged components, the physical characteristics vary with LWR plant type and the component's application. The metal activation that produces 10 CFR Part 61 long-lived activation product radionuclides is directly proportional to the integrated neutron flux received by the component. Therefore, the extent to which a discharged component approaches Class C limits depends only on its initial materials composition, its axial and radial location in the core, and the duration for which it is exposed to the core neutron flux. Components that are fully inserted in the core and exposed to a relatively constant neutron flux will approach Class C limits with less incore time than those components that are only partially inserted in the same core. For example, a pressurized water reactor (PWR) instrument string will approach Class C limits after several fuel cycles since its hot end is fully inserted, while a PWR reactor control cluster assembly will never approach Class C limits since it is either fully or partially withdrawn during full power operation.

For decommissioning activated metals, the components consist of the structural components that support the fuel assemblies in the reactor vessel. Their axial and radial locations within the reactor vessel vary, and they are not exposed to the same neutron flux as fuel assemblies. Thus, while these components are exposed to a somewhat reduced neutron flux, this exposure takes place throughout the LWR operating lifetime. The physical characteristics of these structural components vary with LWR plant type and the component application. 10 CFR Part 61 activation product radionuclide content is directly proportional to the integrated neutron flux received by the component. Therefore, the extent to which a component approaches Class C limits depends only on its initial materials composition, its axial and radial location relative to the core, and the plant's operating history.

Table I-4 identifies the components considered. All component types identified as potential GTCC LLW are discussed below.

4.1.1 BWR Routine Components

The components of interest, except poison curtains, enter the core from the bottom. With the exception of control rod blades, these components are located in the active fuel region neutron flux. Potential GTCC components are discussed below.

Table I-4. Potential GTCC activated metals

POTENTIALLY GTCC ACTIVATED METALS		
	Waste Type Volume (Ft ³)	GTCC Potential
1. BWR Operations		
a. Control Rod Blades	0.6–0.64	YES
b. LPRM Strings	0.1	YES
c. Dry Tubes	0.08	YES
d. Poison Curtains	0.08	YES
e. Secondary Sources	0.03–0.05	NO
f. Fuel Channels	0.16–0.24	NO
2. PWR Operations		
a. Burnable Poison Rod Assemblies	0.06–0.24	NO
b. Thimble Plug Assemblies	0.02–0.03	YES
c. Instrument Strings	0.01–0.10	YES
d. Primary Sources	0.02–0.10	YES
e. Secondary Sources	0.01–0.10	NO
3. BWR Decommissioning		
a. Core Shroud	120–150	YES
b. Orificed Fuel Supports	14–18	YES
c. Control Rod Guide Tubes	70–95	NO
d. Top Fuel Guide	10–12	NO
e. Jet Pump Assemblies	25–30	NO
4. PWR Decommissioning		
a. Core Shroud	50–60	YES
b. Lower Core Barrel	150–200	YES
c. Thermal Shield	40–50	YES
d. Lower Grid Plate	15–20	YES
e. Upper Core Grid Plates	15–20	NO
f. Lower Support Columns	12–15	NO
5. Other Reactor Internals		
a. Research	Variable	YES
b. Test	Variable	YES

Control Rod Blades. Each core contains from 137 to 185 control rod blades (CRB) that are distributed radially throughout the core. A CRB consists of an upper handle, four blades containing neutron-absorbing material in a cruciform configuration, and a velocity limiter section. These units are attached to control rod drives at the velocity limiter end.

Typically, CRBs are comprised of B_4C absorber material in stainless steel tubing that is clad in a stainless steel sheath. Weights are model dependent and vary between 203 and 280 lb. Displaced volumes vary between 0.6 to 0.64 ft³.

Older models also contains eight stellite bearings. The top four stellite bearings are just below the handle and reside in the core. Due to the extremely high radiation levels from the cobalt in the stellite, the top four stellite bearings are usually removed prior to CRB shipment. Newer model CRBs have replaced the stellite bearings with inconel bearings, which perform the same function without the high cobalt content and attendant radiation levels.

CRBs are discharged based on boron depletion, which is proportional to the unit neutron exposure. Control patterns vary, with some CRBs full in, others full out, and others partially inserted. Some plants also run in what is referred to as the controlled cell mode. In this mode, CRB movement is restricted to a fixed group of 25 to 37 CRBs, with the rest of the CRBs completely withdrawn during full power operation.

Thus, the extent of boron depletion depends on radial location and mode of operation. Boron depletion also varies axially due to the partial insertion of the blade along its length. Some CRBs that remain incore for 10 to 15 years can have the same exposure as those run in the controlled cell mode, which remain for one to two fuel cycles.

Individual CRBs with extensive boron depletion (high neutron exposure) are potential GTCC LLW. Where stellites have been removed from the older model CRBs, the stellites alone are always GTCC LLW.

Local Power Range Monitor Strings. Each core contains 31 local power range monitor (LPRM) strings. These units are 0.7 in.-diameter and 0.03 in.-thick stainless tubing and are 38 to 42 ft in length. Each LPRM string contains four fission chambers that monitor neutron flux in the active core region. The units are inserted through the bottom of the reactor vessel. The uppermost 13 to 14 ft, which reside in the core itself, are commonly referred to as the hot end.

The metal volume of an intact LPRM string is 0.1 ft³, and the weight is about 50 lb. Each fission chamber, which is an integral part of the string, is about one in. long and weighs about 40 g.

LPRM strings are routinely discharged based on fission chamber depletion or when strings malfunction. Chamber depletion is proportional to the neutron exposure of the string which, for each fuel cycle, varies with the string radial location.

Individual LPRM strings with extensive neutron exposure of several fuel cycles are usually GTCC LLW. Where fission chambers have been removed and segregated from the string, the fission chambers alone are always GTCC LLW.

IRM/SRM Dry Tubes. Each core contains four source range monitors (SRM) and eight intermediate range monitors (IRM). These monitors are fission chambers that are inserted into the core region through stainless steel tubes referred to as dry tubes. These dry tubes are 0.69 in.

diameter and 0.012 in.-thick stainless tubing and are 42 to 44 ft in length. The units are inserted through the bottom of the reactor vessel. The uppermost 13 to 14 ft, which reside in the core itself, are commonly referred to as the hot end. The metal volume of an intact dry tube is 0.08 ft³, and the weight is about 39 lb.

Dry tubes are not routinely discharged. They are only replaced when damaged or embrittled. Individual dry tubes with extensive neutron exposure of several fuel cycles are usually GTCC LLW.

Poison Curtains. For the older plants, each core contained 140 to 170 poison curtains to shape flux during the first fuel cycle. These poison curtains are thin, about 0.06 in. thick, and have flat borated stainless steel plates 8.5 in. wide by 13 ft long. They were located between fuel assemblies and were exposed to the active fuel region neutron flux. These units receive one cycle of exposure before discharge, but, due to their axial location and small volume of about 0.08 ft³, they usually exceed Class C limits. Several of the older power plants still have poison curtains in their fuel pools.

4.1.2 PWR Routine Components

The components of interest enter the core from the top and, with the exception of instrument strings, are located above the active fuel region neutron flux.

Unlike boiling water reactors (BWRs), there are differences in materials compositions and physical characteristics among the components for the three PWR reactor manufacturers, Westinghouse, Babcock & Wilcox, and Combustion Engineering. Potential GTCC components are discussed below.

Thimble Plug Assemblies. These components are used to restrict water flow through fuel assemblies and are positioned in the tops of selected assemblies. For Westinghouse plants, they are stainless steel with an inconel spring and weigh 10 to 12 lb. They are only discharged when damaged. Although positioned above the active fuel region, their relatively long incore residence times can result in some units exceeding Class C limits.

For Babcock & Wilcox plants, orifice rod assemblies are very similar to Westinghouse thimble plugs. Orifice rod assemblies weigh about 15 lb and are made of stainless steel with an inconel spring. Like the thimble plugs, their long exposure times can result in some units exceeding Class C limits.

Incore Instrument Strings. Instrument strings consist of flux monitor probes and inconel or stainless steel flux wires. They are inserted through instrument tubes in the fuel assemblies. Their characteristics vary with reactor vendor.

Westinghouse plants use eight fixed and four movable incore detectors that contain fission chamber detectors housed in a stainless steel sheath. Each movable detector can be inserted into any one of 60 detector thimbles in the core. The fixed detectors have a total length of about 110 ft and a metal volume of about 0.06 ft³. The top 30 ft are activated due to neutron exposure. The 12 to 14 ft hot end section, which resides in the active core region, typically exceeds Class C limits. The fission chamber detectors themselves always exceed Class C limits.

Babcock & Wilcox plants have locations to accommodate 52 incore instrument assemblies. These instrument strings use rhodium emitters with inconel lead wires and stainless steel sheaths. They have a total length of about 100 ft and metal volumes in the 0.05 to 0.1 ft³ range. About 30 ft of the instrument string is neutron activated. The 12 to 14 ft hot end section typically exceeds Class C limits.

Combustion Engineering plants use between 28 and 61 incore instrument assemblies. These instrument strings use rhodium emitters attached to inconel lead wires, with the sheathing material being either inconel or stainless steel. They have a total length of 35 to 116 ft and weigh between 7 and 34 lb. Metal volumes are in the 0.01 to 0.07 ft³ range. The top 30 ft are activated due to neutron exposure. The 12 to 14 ft hot end section typically exceeds Class C limits. The inconel assemblies are the most restrictive from a classification standpoint.

Primary Sources. Westinghouse plants utilize one to two primary source assemblies. The primary source assemblies are thimble plug assemblies with one primary source rod, one to four secondary source rods, and 0 to 16 burnable poison rods. The remainder of the 24 guide tubes contain thimble plugs. The primary source assemblies have a total weight of 19 to 52 lb. The primary source rods weigh 2.7 to 3.6 lb. and are 110 to 152 in. long. The source materials are californium, plutonium-beryllium, or polonium-beryllium. The californium and plutonium-beryllium, source rods always exceed Class C limits due to the high transuranic content. Since the rods are not easily identifiable, the entire assembly is considered to exceed Class C limits.

Babcock & Wilcox plants use two primary neutron sources, which are stainless steel-clad americium beryllium sources. Each primary source is 146 in. long and 0.4 in. in diameter, with a total weight of about 1 lb. The primary sources are expected to exceed Class C limits due to the high transuranic content.

Each combustion engineering plant uses two primary neutron sources. The source assemblies are 99 to 117 in. long and weigh between 4.5 and 10.9 lb. The source assemblies are 316 stainless steel with inconel springs. Each source assembly contains one antimony-beryllium and two plutonium-beryllium sources. Due to the high transuranic content of the plutonium sources, these source assemblies will exceed Class C limits.

4.1.3 BWR Decommissioning Internals

The NRC has supported BWR decommissioning studies⁴ to identify the activated metals that could be GTCC LLW. Although these studies are dated, they are still used as reference materials on this subject. Activated metals in a BWR include the reactor vessel and its internal structural components. The low-alloy steel reactor vessel is cylindrical with hemispherical heads. Except for the top hemispherical head, the reactor vessel is clad internally with stainless steel. The vessel itself does not approach Class C limits. Other components include the core shroud, the top fuel guide, steam separators and dryers, jet pumps, control rod guide tubes, and orificed fuel supports. Except for the steam separators and dryers, these components are exposed to neutron fluxes that could lead to components, or parts thereof, that exceed Class C limits. These components are discussed below.

Core Shroud. The shroud is a large cylindrical type 304 stainless steel structure (about 90 to 180 ft³) that surrounds the core and provides a barrier to separate the upward flow through the core from the downward flow in the annulus. A flange at the top of the shroud mates with a flange on the steam separator assembly to form the core discharge plenum. This component can readily exceed Class C limits.

Orificed Fuel Supports. Each fuel assembly that makes up the core rests on an orificed fuel support (guide) mounted on top of the control rod guide tubes. Each fuel support is stainless steel and supports one to four fuel assemblies, dependent upon the radial location in the core. Each unit is about 1 ft² and has a metal volume of about 0.1 ft³ for a total metal volume of about 14 to 18 ft³. It is unlikely that these components will exceed Class C limits.

Control Rod Guide Tubes. Each core contains 137 to 185 type 304 stainless steel control rod blade guide tubes that extend from the top of the control rod drive housings up through holes in the core support plate. Besides serving as guides for control rods, the tubes also provide support for the fuel assemblies surrounding the control rods. Each guide tube has a metal volume of 0.5 ft³ for a total metal volume of about 70 to 95 ft³. It is unlikely that these components will exceed Class C limits.

Top Fuel Guide. The top fuel guide is a stainless steel structure that provides lateral support for the top section of the fuel assemblies. It is composed of about 10 to 12 ft³ of stainless steel beams that are joined at right angles, forming square openings for the fuel assemblies. The beams are fastened to a peripheral rim that rests on the core shroud. Dependent upon the core design, the beams that make up the top fuel guide can extend down into the active fuel region, which considerably increases their activation. Some portions of the top fuel guide that are toward the center of the core may exceed Class C limits, but the component as a whole is not expected to be GTCC.

Jet Pump Assemblies. Two semicircular groups of 10 jet pumps are located in the outer annulus between the core shroud and the reactor vessel wall. The jet pumps are type 304 stainless steel and have a total metal volume of 25 to 30 ft³. Each jet pump consists of an inlet riser pipe, a driving nozzle, a suction inlet, a throat or mixing section, and a diffuser with attachment braces. The driving nozzle, suction inlet, and throat are joined together as a removable unit. The diffuser, associated bracing, and riser pipe are permanently installed. This component should not exceed Class C limits.

4.1.4 PWR Decommissioning Internals

The NRC has also supported PWR decommissioning studies.⁵ These studies are dated but continue to be used as reference materials. Activated metals in a PWR include the reactor vessel and its internal structural components. The reactor vessel is cylindrical, with a hemispherical bottom head and a flanged and gasketed removable upper head. The body of the vessel is low alloy carbon steel. Inside surfaces in contact with coolant are clad with a minimum of 1/8 in. of stainless steel. The vessel itself does not approach Class C limits.

PWR internals include the core shroud and support structures, thermal shield, and coolant circulating channels. These components are exposed to neutron fluxes that could lead to components, or parts thereof, exceeding Class C limits. These components are discussed below.

Core Baffle (Shroud). The baffle, or shroud, is a type 304 stainless steel structure that surrounds the core and provides for a limited water flow between the baffle and the core barrel in order to provide additional cooling of the barrel. The core shroud has a metal volume of about 40 to 70 ft³. This component can readily exceed Class C limits.

Lower Core Barrel. The lower core barrel is a type 304 stainless steel cylinder that provides lateral support for the core and directs the flow of coolant water. The lower core barrel has a metal volume of about 80 to 130 ft³. The lower core barrel can exceed Class C limits.

Thermal Shield. The thermal shield is attached to the lower core barrel in the high flux region and protects the reactor vessel by attenuating gamma radiation and fast neutrons that escape the core. The heat energy deposited in the thermal shield is removed by the reactor coolant. The thermal shield is type 304 stainless steel and has a metal volume of about 50 to 110 ft³. It is unlikely that these components will exceed Class C limits.

Lower Grid Plate. The lower grid plate is type 304 stainless steel and has a metal volume of 15 to 20 ft³. It is positioned at the bottom level of the core and provides support and orientation for the fuel assemblies. The lower grid plate is perforated and contains the locating pins for the fuel assemblies. This component should not exceed Class C limits.

Upper Core Grid Plates. The upper core grid plate contains about 15 to 20 ft³ of 304 stainless steel. It is perforated and provides orientation for the upper core support assembly. Fuel assembly locating pins protrude from the bottom of the upper core grid plate and engage the fuel assemblies as the upper assembly is lowered into place. Some parts of the lower core barrel could exceed Class C limits. The component, as a whole, should not exceed Class C limits.

Lower Support Columns. Columns made from type 304 stainless steel are placed between the lower grid plate and the bottom support casting of the core barrel in order to provide stiffness to this plate and transmit the core load to the bottom support casting. The lower support columns have a metal volume of about 12 to 15 ft³. This component should not exceed Class C limits.

4.1.5 Research and Test Reactor Components

There is a diversity in types and sizes of nuclear research and test reactors and in their operating schedules and lifetimes. This significantly affects the generation of neutron activated materials that can potential exceed Class C limits. The neutron flux levels in these reactors are low compared to power reactors, and the irradiation histories are very different due to intermittent operation for relatively short lifetimes. The facilities considered are:

- TRIGA research reactor with a postulated 40-year operational lifetime at about 5% operating time utilization.
- 60 MWt Materials Test Reactor (MTR), light water moderated and cooled, used in testing materials for space flight applications. This reactor was operated over a 12-year period for a total of 4.476 effective full power years (EFPY).

These reactors do not use routinely replaced stainless steel components, as power reactors do. The relatively low fluxes mitigate against routine GTCC components, and, while isolated cases can be found, these are not a likely source of GTCC LLW.

At the decommissioning of these units, some GTCC LLW could be generated. The NRC studies⁶ on this subject, which are dated, indicate that these units generate small volumes of GTCC LLW at decommissioning.

The core assembly of the TRIGA reactor is a right-circular cylinder 3.6 ft in diameter and 2 ft high. It consists of a compact concentric array of cylindrical fuel elements with a central thimble, a neutron source, and control rods. The stainless components in the core include rotary specimen rack hardware, grid plate inserts and hardware, and three control rods. The aluminum components of the core do not approach Class C limits. The stainless steel components are located close to the core centerline and can potential exceed Class C limits.

The MTR vessel is 31 ft tall and 8.8 ft O.D. It is fabricated of type A-201 steel with a type 304 stainless steel cladding. The major internal components are the core (including fuel, control rods, and incore nuclear instrumentation), the core support structure (including the upper and lower flow guides, control rod drive box, and orifice plate), the thermal shields, the horizontal and vertical test holes, the horizontal beam holes, the thermal column, and the rabbit tubes. Some of these components are potential GTCC LLW.

4.2 Process Wastes

Process wastes consist of the wet wastes that are generated from cleanup of liquids containing soluble and insoluble radioactive constituents.

At power plants, these wastes are in the form of ion-exchange media, filter media, or cartridge filters. These waste forms are potential GTCC due to the concentrations of the Table 2 radionuclides Cs-137 and Sr-90, which have limits of 4600 and 7000 $\mu\text{Ci/cc}$ respectively. The Class C status of some cartridge filters is also affected by the presence of Ni-63, another Table 2 radionuclide with a Class C limit of 700 $\mu\text{Ci/cc}$. Where plants have operated with failed fuel, the concentrations of transuranics, with a Class C limit of 100 nCi/g, can also lead to GTCC LLW.

The other facilities originally considered by DOE included fuel development and fabrication facilities and academic institutions.

Table I-5 identifies the process wastes considered for LWRs, as well as the other types of facilities that could generate GTCC process wastes. Not all process wastes identified are potential GTCC LLW due to the facility type, specific activity, or Class C radionuclide content, or a combination of these factors. All process wastes shown as potential GTCC LLW are discussed below. Where a waste type is not potential GTCC, the basis for this conclusion is also discussed.

4.2.1 BWR Resins--Normal Operations

The BWR waste types of interest use deep bed resins, powdered resins, and other types of filter media. While these waste types can be Class C, in the absence of extensive fuel failures they will not exceed Class C limits.

- Reactor Water Cleanup Resins (RWCU)- These are generated by deep-bed demineralizers consisting of powdered resins. Beds of about 150 ft³ are depleted and transferred every 12 to 18 months.
- Fuel Pool Cleanup Resins--These are generated by relatively small ion-exchange vessels consisting of bead resins, usually two beds with 30 ft³ capacity. Beds are depleted and transferred every 12 to 18 months.

Table I-5. Potential GTCC process wastes

POTENTIALLY GTCC PROCESS WASTES		
	Waste Type Volume (Ft ³)	GTCC Potential
1.BWR Resins – Normal Operations		
a.Reactor Water Cleanup	150	NO
b.Fuel Pool Cleanup	30	NO
c.Fuel Pool Filter Precoat	20–30	NO
2.PWR Resins – Normal Operations		
a.Chemical and Volume Control System	30	NO
b.Fuel Pool Cleanup	30	NO
3.BWR Decontamination Resins		
a.Recirculation/RWCU	100–150	NO
b.Full System – Fuel Out	600–800	YES
c.Full System – Fuel In	1000–1200	YES
4.PWR Decontamination Resins		
a.Steam Generator Channel Head	200–300	NO
b.Full System – Fuel Out	500–4000	YES
c.Full System – Fuel In	600–4000	YES
5.BWR Cartridge Filters		
a.Control Rod Drive	0.01	YES
b.Fuel Pool	0.5	YES
c.Underwater Vacuum	0.5	YES
6.PWR Cartridge Filters		
a.Seal Water Injection	0.1–0.5	YES
b.Seal Water Return	0.5	YES
c.Reactor Coolant system	0.5	YES
d.Letdown	2	YES
e.Fuel Pool	0.5	YES
7.Aqueous Liquids		
	Variable	NO
8.Organic Liquids		
	Variable	NO
9.Zeolite Ion Exchange		
	Variable	NO
10.Sludge		
	Variable	NO

- Fuel Pool Filter Precoat (older BWRs only)--These are fuel pool filter septums (>200 micron elements) coated with a back-flushable mixture of powdered resin and cellulose. Volumes range from 20 to 30 ft³ per charge.

4.2.2 PWR Resins--Normal Operations

The PWR waste types of interest use deep bed cation and anion resins. While these waste types can be Class C, in the absence of extensive fuel failures, they will not exceed Class C limits.

- Chemical and Volume Control System Resins (CVCS)--These bead resins are generated from cation and mixed bed demineralizers with capacities of 30 ft³. Four beds are usually discharged after normal refueling cycles (prior to starting up again) every 12 to 18 months.
- Fuel Pool Cleanup Resins--These bead resins are generated from cation and mixed bed demineralizers with bed capacities of 30 ft³. Beds are discharged every 12 to 24 months.

4.2.3 Decontamination Resins

Two decontamination processes have been employed to date at BWRs, referred to as the LOMI and CANDECON processes. Although the primary systems differ, the same two decontamination processes have been employed at PWRs. A third process, referred to as CANDEREM, is also being evaluated for the full reactor coolant system decontamination at PWRs. These processes deposit the crud and scale removed from the reactor internal surfaces on deep bed resins.

Each of these decontamination processes involves several steps and uses different types of resin that include cation, mixed, and anion beds. The chelation step generates the cation resins, which can contain up to 75% of the total activity removed during the decontamination. The cation resins have the highest curie loading and could exceed Class C limits.

At BWRs, these processes have only been employed on the recirculation reactor water cleanup (RWCU) system, that represents less than 20% of a typical BWR system surface area with fuel out. At PWRs, these processes have only been employed on the steam generator channel heads. Channel heads represent less than 1% of the surface area of a PWR primary system.

The power industry is also considering what is referred to as a full RCS decontamination with and without fuel assemblies in the system. These decontamination operations are not routine operations and might be employed once per decade at a plant for as low as is reasonable achievable (ALARA) reasons.

A recent study⁷ examined the GTCC potential for the resins generated by the CANDECON and LOMI processes. The data were more recently supplemented by an ongoing evaluation of full RCS system decontamination^c for PWRs. Some of the results of these studies are summarized below.

BWR Recirculation/RWCU Piping System. This decontamination process covers a surface area of about 1800 m² and generates 100 to 150 ft³ of resin, dependent upon the decontamination process employed (Reference 5) It is highly unlikely that these resins will be GTCC LLW.

c. Westinghouse presentation to U. S. Nuclear Regulatory Commission, December 1990.

BWR Reactor Coolant System (Fuel Out). This decontamination process covers an RCS surface area of about 11,000 m². It can generate 600 to 800 ft³ of resin, dependent on the decontamination process employed (Reference 5). It is unlikely that these resins can be GTCC LLW.

BWR Reactor Coolant System (Fuel In). This decontamination process covers the same RCS surface area plus the fuel assemblies, which represent an additional 12,000 m², (including fuel channels) for a total area of about 23,000 m². It can generate 1,000 to 1,200 ft³ of resin, dependent upon the decontamination process employed (Reference 5). Cation resin volumes could range from 200 to 275 ft³. Due to the presence of the fuel assemblies, a source of TRU radionuclides, about half of these cation resins can be GTCC LLW.

PWR Steam Generator Channel Head. This decontamination process covers a surface area of about 1.5 m² and generates 200 to 300 ft³ of resin (Reference 5), dependent upon the decontamination process employed. It is highly unlikely that these resins will be GTCC LLW.

PWR Reactor Coolant System (Fuel Out). This decontamination process covers an RCS surface area of about 200 m². One study (Reference 5) estimates it can generate 1,500 to 1,700 ft³ of resin, dependent upon the decontamination process employed. Other sources (Footnote a) indicated larger volumes, ranging from 3,000 to 4,000 ft³. About 75% of the activity is on 300 ft³ of cation resin. It is unlikely that these cation resins will be GTCC LLW.

PWR Reactor Coolant System (Fuel In). This decontamination process covers the same RCS surface area plus the fuel assemblies, which represent an additional 50 m², for a total area of about 250 m². One study (Reference 5) indicated 1,600 to 1,700 ft³ of resin, dependent upon the decontamination process employed. The other study (Footnote a) indicated volumes ranging from 3,000 to 4,000 ft³. About 75% of the activity is on 300 ft³ of cation resin. About half of these cation resins could be GTCC LLW.

4.2.4 BWR Cartridge Filters

The BWR filter cartridges of interest remove particulates from primary coolant and in the spent fuel pool. Due to their relatively small volumes and high activities, these filters are usually Class C. Individual filters can exceed Class C limits.

- **Control Rod Drive Strainers**--Each control rod drive contains three strainers that remove particulates. The volumes of these strainers are very small at 0.006 to 0.06 ft³, and individual strainers, particularly the small unit at the spud end, could be GTCC LLW. These strainers are replaced when the control rod drives are refurbished.
- **Fuel Pool Filters**--These units are located upstream or downstream of fuel pool purification demineralizers and consist of cartridge-type filter elements with metal retention screens. Their sizes vary but are usually about 6 in. in diameter and 30 in. long. The volume is about 0.5 ft³.
- **Underwater Vacuum Filters**--These units consist of large and small micron cartridge filters with metal retention screens. They are generated in the fuel pool during refueling. The volume is about 0.5 ft³.

4.2.5 PWR Cartridge Filters

PWRs use cartridge filters throughout the primary and secondary systems. Because of their relatively small volumes and high activities, these filters are usually Class C. Individual filters can exceed Class C limits.

- Seal Water Injection (SWI) Filters--These units consist of very small micron metal or fiber retention elements to purify water from the volume control tank. Reactor coolant pump SWI filters are normally changed during refueling operations and after major power transients. Filter cartridge sizes vary from 2 in. in diameter by 6 in. long to 6 in. in diameter by 30 in. long.
- Seal Water Return (SWR) Filters--These units consist of medium micron metal or fiber retention elements. Reactor coolant pump SWR filters are normally changed during refueling operations and after major power transients. Typically these units are about 0.5 ft³.
- Reactor Coolant System (RCS) Filters--These units consist of small to medium micron size cartridge filters with metal retention screens that purify water returning to the volume control or make-up tank. These filters are normally changed during normal refueling operations and after major power transients. Typically, these units are about 0.5 ft³.
- Letdown Filters--Some PWRs use back-flushable letdown filters downstream of the nonregenerative heat exchanger. When these filters are back-flushed, the liquids and solids pass through medium to large micron cartridge filters (crud filters) with metal retention screens prior to being transferred to a waste holdup tank. These units could be as large as 12 in. in diameter and 30 in. long, or about 2 ft³.
- Fuel Pool Filters--These units are located upstream or downstream of fuel pool purification demineralizers and have cartridge type-filter elements with metal retention screens. Their sizes vary but are usually about 6 in. in diameter and 30 in. long, or about 0.5 ft³.

4.2.6 Aqueous Liquids

The DOE survey (Footnote b) indicated that one respondent, an academic institution, had TRU-bearing aqueous liquids that were GTCC LLW. These wastes must be stabilized for disposal, and it is unlikely that the stabilized liquids will exceed Class C limits.

4.2.7 Organic Liquids

The DOE survey (Footnote b) indicated that three respondents, all industrial firms, had organic liquids that were GTCC LLW. These wastes are mixed wastes and are thus excluded from consideration.

4.2.8 Zeolite Ion-Exchange Media

The DOE survey (Footnote b) indicated that some utility respondents had zeolite ion-exchange media that were GTCC LLW. Most of these wastes were produced during cleanup of Three-Mile Island reactor TMI-2, and were disposed of at a DOE facility. A form of zeolites is currently used

in the power industry, but the liquid waste types processed are not those that could lead to GTCC LLW.

4.2.9 Filter Sludge

The DOE survey (Footnote b) indicated that a utility respondent had TRU-bearing sludge in the fuel pool that was GTCC LLW. TRU-bearing sludges from fuel pools are usually deposited on cartridge filters. They do not warrant a separate waste type designation since they are addressed under cartridge filters.

4.3 Dry Contaminated Solids

Dry wastes are routinely generated at most facilities in the nuclear power industry and by other commercial users of radioactive materials that produce LLW. In fact, these dry materials, which are in the form of contaminated plastic, paper, cloth, glass, metals, and wood, have historically represented the largest volume of LLW generated.

Since these materials become LLW due to contamination, their specific activity is usually very low, and they rarely exceed Class A limits, much less Class C limits. Accordingly, the routine generation of GTCC LLW based on exceeding the Class C limits for Table 2 radionuclides like Cs-137, Sr-90 and Ni-63 is highly unlikely. For most facilities, it is also unlikely that the Class C Table 1 limits for transuranics will be exceeded. There are, however, some fuel development and hot cell facilities that could generate GTCC LLW due to TRU contamination. These waste types are discussed below.

4.3.1 Contaminated Nonmetals

These wastes are compactible and, in most cases, combustible. Dry wastes that are produced in areas that contain TRU radionuclides are usually segregated and evaluated separately before commingling with other dry wastes or volume reduction processing. The DOE survey (Footnote b) indicated volumes for these wastes. They are unique to a few facilities.

4.3.2 Contaminated Metals

These wastes are generated by the same types of facilities as the nonmetals, but the metals are neither readily compactible nor combustible. The DOE survey (Footnote b) indicated volumes for these wastes. They are limited to a few facilities.

4.4 Sealed Sources

Sealed sources consist of small capsules, usually stainless steel, which encapsulate relatively high concentrations of a single radionuclide. Sources are used for a wide range of industrial and medical applications and become waste when they are no longer usable. The NRC has conducted a survey on sealed sources.^d

Criteria for disposal of sealed sources has evolved to the point where the sources can be divided into two distinct groups: (a) those containing transuranics, and (b) other radionuclides. As noted

d. U. S. Nuclear Regulatory Commission, Sealed Source Data Survey, December 1990.

in Section 3, the current and prospective criteria for disposal of TRU sources is more stringent than that for non-TRU sources.

Transuranics are 10 CFR Part 61 Table 1 radionuclides with a Class C limit of 100 nCi/g. The transuranic radionuclides of interest include Am-241, Cm-244, Pu-238, and Pu-239. For other sealed sources, the radionuclides of interest are Cs-137, Sr-90, and Ni-63. These are all 10 CFR Part 61 Table 2 radionuclides with Class C concentration limits of 4600, 7000, and 700 μ Ci/cc, respectively.

Table I-6 identifies the types of sealed sources considered, their volumes and activity ranges, and radionuclides of interest. All source types identified are discussed below by group.

4.4.1 Transuranic-Bearing Sources

As shown in Table I-6, most TRU sealed sources have activities in the millicurie to curie range and volumes in the 1 cc to 1000 cc range. Assuming the density for stainless steel, most sealed source TRU concentrations are orders of magnitude above the 100 nCi/g, which is the Class C limit. A description of these sources and their characteristics is presented below.

TRU Gamma Gauges. These are used throughout industry to measure or control processes, component thickness, density, and other industrial parameters. Both Am-241 and Cm-244 sources are used. For Am-241 gauges, source strength may vary from 0.1 μ Ci to 10 Ci. For Cm-244 gauges, the strength may vary from 1 mCi to 1 Ci.

Typically the volumes of these sources are less than 1 cc and, assuming a stainless steel density, weigh about 8 g. All sources with strengths above 1 μ Ci exceed Class C limits.

X-Ray Fluorescent Sources. These are used to determine material composition. Both Am-241 and Cm-244 sources are used. For Am-241 sources, source strength may vary from 2 to 200 mCi. For Cm-244 sources, the strength may vary from 10 to 100 mCi. Typically, the volumes of these sources are less than 1 cc and, assuming a stainless steel density, weigh about 8 g. All these sources exceed Class C limits.

TRU Well Logging Sources. These are used to determine geological properties of rock formations. Each licensee uses both a gamma source and a neutron source. The neutron sources are either Am-241/Be or Pu-239/Be. Source strengths can vary from 5 to 20 Ci. Typically, the volumes of these sources are less than 1 cc and, assuming a stainless steel density, weigh about 8 g. All these sources exceed Class C limits.

TRU Moisture Density Gauges. These are portable devices used to measure the moisture content of soils and other materials. The TRU type primarily consists of 50 mCi Am-241 sources with a source volume of about 500 cc. Assuming a stainless steel density, these sources weigh about 4000 g for a concentration of 12,500 nCi/g. All these sources exceed Class C limits.

Smoke Detectors. These devices are used in homes to detect combustion gases and particles. They employ Am-241 sources in the 1 to 180 μ Ci range. The sources themselves are very small, about 0.1 cc, and even at the 1 μ Ci level, they exceed Class C limits. Since they are consumer products, they are not regulated as LLW. However, the manufacturers of these devices are regulated and do produce them as LLW.

Table I-6. Potential GTCC sealed sources

POTENTIALLY GTCC SEALED SOURCES					
	Part 61 Nuclide Of Interest	Typical Volume cc	Minimum Activity Ci	Maximum Activity Ci	GTCC Potential
TRU Bearing Sources					
1. Gamma Gauges	Am-241	1	1.00E-07	1.00E+01	YES
	Cm-244	1	1.00E-03	1.00E+00	YES
2. X-Ray Fluorescent	Am-241	1	2.00E-03	2.00E-01	YES
	Cm-244	1	1.00E-03	1.00E-01	YES
3. Well Logging	Am-241	1	5.00E+00	2.00E+01	YES
	Pu-239	1	5.00E+00	5.00E+00	YES
4. Moisture Gauges	Am-241	500	5.00E-02	5.00E-02	YES
5. Smoke Detectors	Am-241	1	1.00E-06	1.80E-04	YES
6. Pacemakers	Pu-238	100	4.00E+00	4.00E+00	YES
Other Sources					
1. Med. Teletherapy	Cs-137	50	1.00E+02	1.50E+03	YES
2. Gamma Gauges	Cs-137	1	1.00E-07	1.00E+01	YES
3. Beta Gauges	Sr-90	3	1.00E-08	1.00E-01	YES
4. Well Logging	Cs-137	50	2.00E+00	2.00E+00	YES
5. Moisture Gauges	Cs-137	100	1.00E-02	1.00E-02	NO
6. Calibration	Cs-137	2	1.00E-04	3.00E+01	YES
7. Gamma Irradiators I	Cs-137	50	3.00E+01	3.00E+04	YES
8. Gamma Irradiators II-IV	Cs-137	2000	1.00E+03	1.00E+04	YES
9. Gas Chromatography	Ni-63	0.02	1.00E-02	2.00E-02	YES
10. Beta Eye Applicators	Sr-90	1	7.50E-02	1.50E-01	YES

Heart Pacemakers. These devices are implanted in humans to regulate heartbeat. These devices contain about 4 Ci of Pu-238 and contain about 50 g of material. All these sources exceed Class C limits.

4.4.2 Nontransuranic Sealed Sources

As shown in Table I-6, other sealed sources have activities from microcuries to thousands of curies and volumes that vary from less than 1 cc to 2000 cc. While some of these sources do not exceed Class C limits, most of them do. A description of these sources and their characteristics is presented below.

Medical Teletherapy Sources. These sources are used to directly treat diseases with gamma radiation. Most are Co-60 sources that cannot exceed Class C limits, but some are also Cs-137 sources. Due to their applications, these are high activity sources containing up to 1500 Ci. They also have relatively small volumes in the 30-50 cc range. At a limit of 4600 $\mu\text{Ci/cc}$, all these Cs-137 sources exceed Class C limits.

Other Gamma Gauges. These are used throughout industry to measure or control processes, component thickness, density, and other industrial parameters. Cs-137 sources are used with source strengths in the 0.1 μCi to 10 Ci range. Typically, the volumes of these sources are less than 1 cc. Accordingly, at a limit of 4600 $\mu\text{Ci/cc}$, all Cs-137 sources with strengths above 4.6 mCi exceed Class C limits.

Well Logging Sources. These are used to determine geological properties of rock formations. Each licensee uses both a gamma source and a neutron source. The gamma source is usually 2 Ci of Cs-137. Typically, the volumes of these sources are 30 to 50 cc. At a limit of 4600 $\mu\text{Ci/cc}$, all these Cs-137 sources exceed Class C limits.

Beta Gauges. Beta-emitting sources are also used throughout industry to measure or control processes, component thickness, density, and other industrial parameters. Sr-90 sources are used with source strengths in the 0.01 μCi to 100 mCi range. Typically, the volumes of these sources are less than 5 cc. At a limit of 7000 $\mu\text{Ci/cc}$, all Sr-90 sources with strengths above 30 mCi exceed Class C limits.

Moisture Density Gauges. These are portable devices used to measure the moisture content of soils and other materials. The non-TRU type primarily consists of 10 mCi Cs-137 sources, with a source volume of about 100 cc. At a limit of 4600 $\mu\text{Ci/cc}$, all these Cs-137 sources are below Class C limits.

Calibration Sources. These are used to determine the accuracy of instruments. These are typically Cs-137 sources with strengths in the 100 μCi to 30 Ci range. The volumes are relatively small, about 2 cc. At a limit of 4600 $\mu\text{Ci/cc}$, all Cs-137 sources with strengths of about 70 mCi exceed Class C limits.

Category I Gamma Irradiators. These are self-shielded devices used for food and materials irradiation. They are Cs-137 sources with activities in the 30 to 30,000 Ci range, with volumes in the 30-50 cc range. At a limit of 4600 $\mu\text{Ci/cc}$, all these Cs-137 sources exceed Class C limits.

Category II, III, and IV Gamma Irradiators. These are very high activity irradiators that are themselves composed of multiple sources. They are Cs-137 sources with activities up to 15 million Ci.

Source volumes are relatively large, about 2000 cc. Assuming individual source strengths in the 10,000 Ci range, all these Cs-137 sources exceed Class C limits.

Gas Chromatography Sources. These are used in the chemical analysis of materials. These sources are primarily Ni-63 and have strengths in the 10-20 mCi range. Volumes are very small, less than 0.1 cc. At a limit of 700 μ Ci/cc, all these Ni-63 sources exceed Class C limits.

Beta Eye Applicators. These are Sr-90 sources used to treat eye cancers. Typically, source strengths are in the 75 to 150 mCi range, and volumes are small, about 1 cc. At a limit of 7000 μ Ci/cc, all these Sr-90 sources exceed Class C limits.

5. CONCENTRATION AVERAGING AND RELATIVE VOLUME PROJECTIONS

Section 4 identified the waste types that could potential exceed Class C limits under the most conservative concentration averaging scenario--the high volume case. This section analyzes each potential GTCC LLW type and quantifies its volume relative to the base and low volume scenarios defined in Section 3. These relative volume factors can be incorporated into the DOE projection model to project volumes as a function of concentration averaging scenario.

5.1 Methods of Analysis

For discussion purposes, the concentration averaging scenarios defined in Section 3 were redefined. The high volume case is referred to as worst, the base volume case is referred to as average, and the low volume case is referred to as best.

As described in Section 4, a wide range of routinely generated waste types could exceed Class C limits and become GTCC LLW. In some cases, the waste type clearly exceeds the limits, and all the waste produced is GTCC. This applies to some sealed sources. In other cases the waste type only exceeds the limits under worst case assumptions, the high volume scenario. This applies to activated metals in the form of BWR dry tubes. In still other cases, while the waste type exceeds the limits under the worst case scenario, only a portion of the amount generated does so. This applies to activated metals in the form of BWR control rod blades where position and mode of operation impact classification status relative to Class C limits.

To analyze the impact of the various concentration averaging scenarios on volume projections, the high volume case was used as the baseline. Each potential GTCC LLW type under this scenario was reviewed, and the amounts of GTCC LLW in the waste type were estimated. To quantify these amounts as percentages, current practices were used in conjunction with the following qualitative descriptors.

- 100%--Essentially all of the waste generated by the waste type will be GTCC.
- 75%--Most of the waste generated by the waste type will be GTCC.
- 50%--About half of the waste generated by the waste type will be GTCC.
- 25%--Some of the waste generated by the waste type will be GTCC.
- 10%--A small portion of the waste generated by the waste type will be GTCC.
- 1%--Essentially none of the waste generated by the waste type will be GTCC. The minimum used for the base case.

Once these estimates were applied to the high volume case by waste type, they were either left intact or adjusted downward to reflect the estimated volumes generated under the less conservative average and best cases. After each waste type in each waste category was quantified as to relative volume in this manner, all quantitative estimates were normalized to the average volume case.

5.2 Activated Metals

The concentration averaging criteria used for both routinely generated and decommissioning activated metals were assumed to be the same. These included:

- Worst Case--No concentration averaging.
- Average Case--Concentration averaging within a factor of 10 among components in the same disposal liner.
- Best Case--Concentration averaging of all activated metals in the disposal liner.

Table I-7 presents the estimates of the portion of each waste type that could be GTCC. The basis for these estimates is discussed below.

The routinely discharged irradiated components described below may exceed Class C limits. However, these irradiated component types, as well as others, may be included in the standard DOE fuels agreement and accompany spent fuel to the DOE repository.

5.2.1 BWR Components

Control rod blades are the largest volume component and are typically discharged in batches when boron is almost depleted. Their characteristics at discharge depend on their position in the core, their irradiation history, and the mode of operation. Records for over 600 control rod blades shipped for disposal were reviewed. About 80 of these blades were run in the controlled cell mode. Most blades that have not been operated in the controlled cell mode are not GTCC at discharge, although at times some of these blades have occupied several different positions and have been in cores for 10 to 15 years. Generally, control rod blades that received over seven cycles of exposure tended to exceed Class C limits. These were a small portion of the blades operated in the normal mode and are GTCC under the worst case conditions.

Blades operated in the controlled cell mode are discharged after one to two fuel cycles and have been operated almost fully inserted. Some of these blades, particularly the two cycle blades, can be GTCC at discharge. In neither case do these blades exceed Class C limits by factors of more than 2. Thus, the concentrations of these and others discharged are usually within a factor of 10, permitting disposal under the average case. Occasionally, a controlled cell mode blade can fall outside the factor of 10 relative to other blades available and be GTCC under the average case. Thus a small portion of the controlled cell mode blades can be GTCC under average conditions. Under best case conditions, there is always another activated metal that can be packaged with GTCC blades.

Both LPRM strings and dry tubes are typically installed for several fuel cycles. Over 1000 LPRM string and about 80 dry tube records were reviewed. Dry tubes are almost always GTCC, and most LPRM strings are GTCC under the worst case. Generally, only those LPRM strings removed early, after one or two cycles, did not exceed Class C limits. Both types of components have very low volumes, which makes it relatively easy to blend these components with the larger volume control rod blades. LPRM strings and dry tubes can usually meet the factor of 10 criteria. Occasionally, some units cannot meet the factor of 10 criteria and become GTCC LLW under the average case. Under best case conditions, there is always another activated metal that can be packaged with GTCC LPRMs and dry tubes.

Table I-7. Relative impact assessment--activated metals

RELATIVE IMPACT ASSESSMENT ACTIVATED METALS			
	Percent GTCC High Case	Percent GTCC Base Case	Percent GTCC Low Case
1. BWR Operations			
a. Control Rod Blades			
-Not Control Cell	25	10	0
-Control Cell	10	1	0
b. LPRM Strings	75	10	0
c. Dry Tubes	100	10	0
2. PWR Operations			
a. Thimble Plug Assemblies	25	0	0
b. Instrument Strings	75	10	0
c. Primary Sources	100	100	100
3. BWR Decommissioning			
a. Core Shroud	100	100	100
b. Orificed Fuel Supports	10	1	0
4. PWR Decommissioning			
a. Core Shroud	100	100	100
b. Lower Core Barrel	25	1	0
c. Thermal Shield	25	1	0
d. Lower Grid Plate	100	100	75
5. Other Reactor Internals			
a. Research	100	100	0
b. Test	100	1	0

Poison curtains are limited to early BWRs and all in existence were discharged in the 1970s. Several hundred poison curtain records were reviewed. At discharge, essentially all these units were GTCC due to their locations and low volumes. Some units still remain in several fuel pools. While they met the factor of 10 for the average case, there wasn't enough other hardware to blend with them for shipment. These units should be disposed before the 1993 deadline, and this component type should not be included in the DOE projection model.

5.2.2 PWR Components

Thimble plug assemblies are not routinely discharged. About 180 records from both Westinghouse and Babcock & Wilcox plants were reviewed. Of the Babcock & Wilcox components, none exceeded Class C limits. The Westinghouse units generally had lower volumes and had seen more fuel cycles than the Babcock & Wilcox units. Some of these Westinghouse units exceeded Class C limits. A small portion of Westinghouse thimble plug assemblies are GTCC at discharge, primarily due to their low volume and the presence of the inconel spring. Under average case conditions, they can be readily combined with other PWR components like burnable poison rod assemblies and meet the factor of 10 criteria for disposal. Hence, these units are not GTCC under average or best case conditions.

Instrument strings vary by manufacturer. Over 800 instrument string records from all three manufacturers were reviewed. About 85% of these units were from combustion engineering plants that regularly replace the strings. About 50% of these units just exceeded Class C limits. A similar situation existed for the Babcock & Wilcox instrument strings. The records reviewed for the Westinghouse plants were for thimble tubes, the stainless tubes into which instrument strings are inserted. All of these units exceeded Class C limits. Although the thimble tubes are removed infrequently, all these units could exceed Class C limits under worst case conditions.

5.2.3 BWR Decommissioning

The core shroud in a BWR should greatly exceed Class C limits due to its proximity to the active fuel region and the long exposure time over the lifetime of a BWR. All of the core shroud will be GTCC even under best case conditions.

5.2.4 PWR Decommissioning

The core baffle (shroud) in a PWR will greatly exceed Class C limits due to its proximity to the active fuel region and the long exposure time. All core shrouds will be GTCC, even under best case conditions.

The lower core barrel will also be GTCC under worst case conditions. However, blending these parts with other internals reduces this to the point where only some core barrels will exceed Class C limits under average case conditions.

There should be no core barrels above Class C limits under best case conditions.

5.2.5 Research and Test Reactor Components

For the TRIGA research reactor, the control rods and the grid plate inserts and hardware will be GTCC under worst case conditions. A small portion of the rotary specimen rack hardware will be GTCC. Blending is limited for the base or average case as there is not much stainless in the core

that can be used for blending. However, the best or low volume case includes aluminum for blending, which reduces the GTCC LLW to essentially zero for the TRIGA research reactor.

For the test reactor, the stainless components that will be GTCC under the worst case include the control rod (upper rollers), miscellaneous bolts, instrumentation thimbles, and shim rod section. Some of the stainless metering plate and flow guide will also be GTCC. Blending these wastes with other activated metals will result in essentially no GTCC LLW from these components for the average case.

5.3 Process Wastes

The concentration averaging criteria used for process wastes included:

- Worst Case--No concentration averaging.
- Average Case--Concentration averaging in that stabilization media volume can be included for resins but not filters.
- Best Case--Concentration averaging in that stabilization media volume can be included for resins and filters.

Table I-8 presents the estimates of the portion of each waste type that could be GTCC. The basis for these estimates is discussed below.

5.3.1 Decontamination Resins

Decontamination resins generated from BWR and PWR full reactor coolant system decontaminations performed about every 10 years could exceed Class C limits. Full RCS decontaminations have not yet been performed, but it is likely that about half of the resins generated with fuel in could exceed Class C limits under worst case conditions. Under average conditions, when these resins are stabilized, essentially none of these resins should be GTCC LLW. The same applies to these resins under the best case conditions.

5.3.2 Cartridge Filters

Operationally, filters can be changed out on radiation level to preclude generation of GTCC LLW in most cases. However, routine practices that rely on pressure differential for filter change-out were assumed for this evaluation.

Filter cartridges are expected to exceed Class C limits when characterized as individual units under the worst case conditions. For PWRs, some filters would be GTCC. For BWRs, some for the fuel pool and vacuum filters would be GTCC, and about half of the control rod drive (CRD) filters would be GTCC under the worst case. When these filters are commingled with other filters and characterized and classified as a group, all filter types except the CRD units should meet Class C limits. In the average case, some CRD units could still exceed the limits due to low volumes and TRU content. Under best case conditions with encapsulation, all cartridge filters should be below Class C limits.

Table I-8. Relative impact assessment--process wastes

RELATIVE IMPACT ASSESSMENT PROCESS WASTES			
	Percent GTCC High Case	Percent GTCC Base Case	Percent GTCC Low Case
1.BWR Decontamination Resins			
a. Full System – Fuel In	25	1	0
b. Full RCS System – Fuel Out	10	1	0
2.PWR Decontamination Resins			
a. Full System – Fuel In	25	1	0
b. Full RCS System – Fuel Out	10	1	0
3.BWR Cartridge Filters			
a.Control Rod Drive	50	10	0
b.Fuel Pool	25	1	0
c.Underwater Vacuum	25	1	0
4.PWR Cartridge Filters			
a.Seal Water Injection	25	1	0
b.Seal Water Return	25	1	0
c.Reactor Coolant System	25	1	0
d.Letdown	25	1	0
e.Fuel Pool	25	1	0

5.4 Contaminated Solids

The basis for consideration of these wastes as GTCC is the presence of TRU radionuclides from a select group of industrial facilities. The concentration averaging scenarios and criteria used for contaminated solids included:

- Worst Case --No concentration averaging or commingling with other lower activity TRU-bearing dry solids and no use of stabilization media volume for incinerator ash.
- Average Case--Commingling with other lower activity TRU-bearing dry solids plus inclusion of the stabilization media volume for incinerator ash.
- Best Case--Commingling with non-TRU-bearing dry solids.

For some unique fuel development and evaluation facilities, some of the contaminated solids generated could exceed Class C limits under the worst case. With commingling under the average case, a small portion of these materials could still be GTCC. With extensive commingling with other contaminated solids under the best case, essentially none of these wastes would be GTCC.

5.5 Sealed Sources

Sealed sources were considered in two groups: TRU-bearing and other. The concentration averaging criteria used for both groups included:

- Worst Case --No encapsulation to reduce concentrations.
- Average Case--Encapsulation allowed for non-TRU sources with activities below 10 curies, no encapsulation for TRU sources.
- Best Case--Encapsulation allowed for TRU and non-TRU sources.

Table I-9 presents the range of radionuclide concentrations for each source type and compares them to the 10 CFR Part 61 Class C limit for the relevant radionuclide. These values represent conditions for the worst case scenario.

Table I-10 takes the values in Table I-9 and considers encapsulation. It was assumed that sealed sources would be encapsulated in a 55-gal drum-size container with a volume of 200,000 cc. It was further assumed that the encapsulation media would be cement at a density of 1.5 g/cc for a weight of 300,000 g. Accordingly, encapsulation will reduce TRU source concentrations by a factor of 300,000 and non-TRU source concentrations by a factor of 200,000. When TRU sources are encapsulated, they still exceed Class C limits for both the average and best cases. When non-TRU sources are encapsulated, most of them meet Class C limits under both the average and the best cases. However, some of these source types could not be encapsulated under the average case because their activity exceeds 10 Ci. Thus, some non-TRU sources would also be GTCC under the average case. Under best case conditions, TRU-bearing gamma gauges (Item 1) and TRU-bearing well logging sources (Item 3) still exceed Class C limits.

Table I-11 presents estimates for the portion of each source type that could be GTCC under each scenario where the source activity ranges varied significantly and resultant concentrations were both above and below the Class C limit. It was assumed that half of the sources were GTCC.

Table I-9. Sealed sources--10 CFR Part 61 radionuclide concentrations

SEALED SOURCES PART 61 RADIONUCLIDE CONCENTRATIONS				
	Part 61 Nuclide	Part 61 Class C Limit	Minimum Specific Activity	Maximum Specific Activity
TRU Bearing Sources		nCi/gm	nCi/gm	nCi/gm
1. Gamma Gauges	Am-241	1.00E+02	1.25E+01	1.25E+09
	Cm-244	1.00E+02	1.25E+05	1.25E+08
2. X-Ray Fluorescent	Am-241	1.00E+02	2.50E+05	2.50E+07
	Cm-244	1.00E+02	1.25E+05	1.25E+07
3. Well Logging	Am-241	1.00E+02	1.04E+09	4.17E+09
	Pu-239	1.00E+02	1.04E+09	1.04E+09
4. Moisture Gauges	Am-241	1.00E+02	1.25E+04	1.25E+04
5. Smoke Detectors	Am-241	1.00E+02	1.25E+02	2.25E+04
6. Pacemakers	Pu-238	1.00E+02	5.00E+06	5.00E+06
Other Sources		uCi/cc	uCi/cc	uCi/cc
1. Med. Teletherapy	Cs-137	4.60E+03	2.00E+06	3.00E+07
2. Gamma Gauges	Cs-137	4.60E+03	1.67E-01	1.67E+07
3. Beta Gauges	Sr-90	9.00E+03	2.94E-03	2.94E+04
4. Well Logging	Cs-137	4.60E+03	4.00E+04	4.00E+04
5. Calibration	Cs-137	4.60E+03	6.25E+01	1.88E+07
6. Gamma Irradiators I	Cs-137	4.60E+03	6.00E+05	6.00E+08
7. Gamma Irradiators II-IV	Cs-137	4.60E+03	5.00E+05	5.00E+06
8. Gas Chromatography	Ni-63	7.00E+02	6.67E+05	1.33E+06
9. Beta Eye Applicators	Sr-90	7.00E+03	7.89E+04	1.58E+05

Table I-10. Sealed sources--10 CFR part 61 radionuclide concentrations after encapsulation

SEALED SOURCES PART 61 RADIONUCLIDE CONCENTRATIONS AFTER ENCAPSULATION				
	Part 61 Nuclide	Part 61 Class C Limit	Minimum Specific Activity	Maximum Specific Activity
TRU Bearing Sources		nCi/gm	nCi/gm	nCi/gm
1. Gamma Gauges	Am-241	1.00E+02	4.17E-05	4.17E+03
	Cm-244	1.00E+02	4.17E-01	4.17E+02
2. X-Ray Fluorescent	Am-241	1.00E+02	8.33E-01	8.33E+01
	Cm-244	1.00E+02	4.17E-01	4.17E+01
3. Well Logging	Am-241	1.00E+02	3.47E+03	1.39E+04
	Pu-239	1.00E+02	3.47E+03	3.47E+03
4. Moisture Gauges	Am-241	1.00E+02	4.17E-02	4.17E-02
5. Smoke Detectors	Am-241	1.00E+02	4.17E-04	7.50E-02
6. Pacemakers	Pu-238	1.00E+02	1.67E+01	1.67E+01
Other Sources		uCi/cc	uCi/cc	uCi/cc
1. Med. Teletherapy	Cs-137	4.60E+03	1.00E+01	1.50E+02
2. Gamma Gauges	Cs-137	4.60E+03	8.33E-07	8.33E+01
3. Beta Gauges	Sr-90	9.00E+03	1.47E-08	1.47E-01
4. Well Logging	Cs-137	4.60E+03	2.00E-01	2.00E-01
5. Calibration	Cs-137	4.60E+03	3.13E-04	9.38E+01
6. Gamma Irradiators I	Cs-137	4.60E+03	3.00E+00	3.00E+03
7. Gamma Irradiators II-IV	Cs-137	4.60E+03	2.50E+00	2.50E+01
8. Gas Chromatography	Ni-63	7.00E+02	3.33E+00	6.67E+00
9. Beta Eye Applicators	Sr-90	7.00E+03	0.00E+00	7.89E-01

Table I-11. Scaled sources--relative GTCC waste volumes

SEALED SOURCES RELATIVE GTCC WASTE VOLUMES			
	Percent GTCC For High Case	Percent GTCC For Base Case	Percent GTCC For Low Case
TRU Bearing Sources			
1. Gamma Gauges	100	100	50
2. X-Ray Fluorescent	100	100	0
3. Well Logging	100	100	100
4. Moisture Gauges	100	100	0
5. Smoke Detectors	100	100	0
6. Pacemakers	100	100	0
Other Sources			
1. Med. Teletherapy	100	100	0
2. Gamma Gauges	50	1	0
3. Beta Gauges	50	1	0
4. Well Logging	50	1	0
5. Calibration	50	25	0
6. Gamma Irradiators I	100	100	0
7. Gamma Irradiators II-IV	100	100	0
8. Gas Chromatography	100	1	0
9. Beta Eye Applicators	100	1	0

5.6 Summary and Conclusions

As expected, the different concentration averaging scenarios have a significant impact on GTCC LLW volumes. Table I-12 presents the estimated portions of each waste type that are GTCC, as presented in Tables I-7, I-8 and I-11, normalized to the base volume case. Note that the minimum allocated to the base or average case was 1%, or essentially none. This enables normalization of the high and low volume estimates to the base volume case.

The results indicate that the extent to which GTCC LLW could be generated, even under the worst case high volume scenario, is less than previous estimates of the types of LLW waste that are potential GTCC. Thus, this evaluation showed fewer potential GTCC LLW types. Additionally, the implicit assumption in previous estimates that if a waste type is GTCC, all wastes generated for that particular type will be GTCC, is not supported. In many cases, only a portion of a particular waste type is potential GTCC, even under the worst case scenario. Accordingly, when these factors are included in the DOE projection model, projections of GTCC volumes will be lower than previous estimates, regardless of the scenario.

The portions of activated metals that could be GTCC under the three scenarios were presented in Table I-7. For routinely generated activated metals, the high volume case varies from 2.5 to 10 times the base case amount for BWR components and up to 10 times the base case amount for PWR components. Except for the PWR primary sources that contain TRU, these amounts are reduced to zero for the low volume or best case. The information used for routinely generated activated metals is based on a significant data base of actual components and has a high confidence level. For decommissioning activated metals, the high volume case for BWR is unity and up to four times the base case amount for PWR components. However, unlike routinely generated components, these amounts are not reduced to zero for the low volume case. This is due to the size of the components and their activity at discharge. The same applies to research and test reactor decommissioning components. The confidence level for the estimates on decommissioning activated metals is much lower than those for other metals. Few components have been actually characterized and disposed. Some of this work is just now being initiated at power reactors and research facilities. Additionally, while there are more extensive technical data sources available than those used, detailed analyses of these data were not performed.

The portions of process wastes that could be GTCC under the three scenarios were presented in Table I-7. For RCS decontamination resins, the high volume case is 50 times the base case amount and is reduced to zero for the low volume case. The use of the stabilization media volume effectively precludes the generation of GTCC resins in the absence of economic factors. For cartridge filters, the high volume case varies from 5 to 25 times the base case amount and is also reduced to zero for the low volume case. Again, the use of the stabilization media volume effectively precludes the generation of GTCC cartridge filters. The information used for spent resins and filter cartridges is based on actual waste data. Except for the full RCS decontamination resins, which have not yet been generated, this information has a high confidence level.

The estimates for contaminated solids are not well-founded since the information available on the few facilities that can reasonably be expected to generate these wastes is limited. However, it was assumed that even under best case conditions there will always be some of these wastes that are GTCC. The readily available DOE information was minimal. As shown, the high volume case is estimated at 25 times the base case amount, and the low volume case is 10% of the base case

Table I-12. Relative GTCC waste volumes normalized to base volume case

RELATIVE GTCC WASTE VOLUMES NORMALIZED TO BASE VOLUME CASE		
	High Volume Scenario	Low Volume Scenario
ACTIVATED METALS		
1. BWR Operations		
- Control Rod Blades-Control Cell	2.5	0
- Control Rod Blades-Normal	10	0
- LPRM Strings	7.5	0
- Dry Tubes	10	0
2. PWR Operations		
- Thimble Plug Assemblies	25	0
- Instrument Strings	7.5	0
- Primary Sources	1	1
3. BWR Decommissioning		
- Core Shroud	1	1
- Orificed Fuel Supports	10	0
4. PWR Decommissioning		
- Core Shroud	1	1
- Lower Core Barrel	25	0
- Thermal Shield	25	0
- Lower Grid Plate	1	0.75
5. Other Reactor internals		
- Research	1	0
- Test	100	0
PROCESS WASTES		
1. BWR & PWR Decon Resins		
- Full RCS with Fuel In	25	0
- Full RCS with Fuel Out	10	0
2. Cartridge Filters		
- BWR Control Rod Drive	5	0
- Other BWR & PWR	25	0
CONTAMINATED SOLIDS	2.5	0.1
SEALED SOURCES		
1. TRU Bearing Sources		
- Gamma Gauges	1	0.5
- X-Ray Fluorescent	1	0
- Well Logging	1	1
- Moisture Gauges	1	0
- Smoke Detectors	1	0
- Pacemakers	1	0
2. Other Sources		
- Med. Teletherapy	1	0
- Gamma Gauges	50	0
- Beta Gauges	50	0
- Well Logging	50	0
- Calibration	2	0
- Gamma Irradiators I,II,III,IV	1	0
- Gas Chromatography	100	0
- Beta Eye Applicators	100	0

amount. The absolute volumes of these waste categories, although not considered here, are expected to be relatively small. The information used for contaminated solids had perhaps the lowest degree of confidence.

Sealed sources have the greatest GTCC potential other than activated metals from LWR decommissioning. The portions of sealed source waste that could be GTCC under the three scenarios were presented in Table I-11. As shown on that table, almost all of the TRU-bearing sources and most of the non-TRU-bearing sources were GTCC for the high volume case. Encapsulation of the non-TRU sources for the average case reduced the base case amounts for these sources, but the TRU sources, which could not be encapsulated, remained unchanged for the base case. Removing the limits on curie content for non-TRU sources and the encapsulation of TRU sources had a significant impact on the relative quantities of GTCC LLW. This impact is reflected in Table I-12, where the high volume case is either equal to or 100 times greater than the base case amount. Conversely, the low volume case essentially eliminates all sealed sources as GTCC LLW except for high activity non-TRU gamma gauges and TRU-based well logging sources.

For sealed sources, NRC reference materials were used almost exclusively. Thus, the information used to identify source characteristics has a high degree of confidence. However, data on the numbers of sources by type and the distribution of activities for source types is not readily available. Moreover, the assumption that sources would be encapsulated in 55-gal drum-size containers with approved stabilization media was overly simplistic. In addition to ignoring economic considerations, it assumes that typical source users are aware of the approved stabilization methods and can readily employ them. Since sealed sources are the most widely used and likely GTCC LLW category, a more detailed evaluation appears warranted.

Finally, the dependence of GTCC LLW generation on the concentration averaging scenarios may lead to some orphan wastes. In a compact based regulatory environment, there is a high degree of certainty that waste could be GTCC in one compact region, meet Class C limits in another, and not meet the DOE definition of GTCC. Steps should be taken to address these types of issues to provide some degree of certainty to potential GTCC LLW type management practices.

6. REFERENCES

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