RECEIVED

DOE/LLW-114A-3

NOV 21 1995

OSTI

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

Appendix A-3: Basis for Greater-Than-Class C Low-Level Radioactive Waste Light Water Reactor Projections

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

September 1994

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DOE/LLW-114A-3

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

Appendix A-3: Basis for Greater-Than-Class C Low-Level Radioactive Waste Light Water Reactor Projections

**A. Mancini
P. Tuite
K. Tuite
S. Woodberry**

Published September 1994

**Idaho National Engineering Laboratory
Idaho Falls, Idaho 83415**

**Prepared for the
U.S. Department of Energy
Office of Environmental Restoration and Waste Management
Under DOE Idaho Operations Office
Contract DE-AC07-76ID01570**

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DL

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

ABSTRACT

This study characterizes low-level radioactive waste types that may exceed Class C limits at light water reactors, estimates the amounts of waste generated, and estimates radionuclide content and distribution within the waste. Waste types that may exceed Class C limits include metal components that become activated during operations, process wastes such as cartridge filters and decontamination resins, and activated metals from decommissioning activities. Operating parameters and current management practices at operating plants are reviewed and used to estimate the amounts of low-level waste exceeding Class C limits that is generated per fuel cycle, including amounts of routinely generated activated metal components and process waste. Radionuclide content is calculated for specific activated metals components. Empirical data from actual low-level radioactive waste are used to estimate radionuclide content for process wastes. Volumes and activities are also estimated for decommissioning activated metals that exceed Class C limits. To estimate activation levels of decommissioning waste, six typical light water reactors are modeled and analyzed. This study does not consider concentration averaging.

SUMMARY

This study evaluates three categories of light water reactor (LWR) waste that may exceed Class C limits: activated metals resulting from operations, process wastes such as cartridge filters and resins, and activated metals resulting from decommissioning. The study characterizes the waste found in each category, estimates the amount that will exceed Class C limits, and estimates the radionuclide content and distribution. This report does not consider concentration averaging. Hence, estimates given herein may not represent amounts of LWR waste that will be classified as greater-than-Class C low-level radioactive waste (GTCC LLW). These estimates are used only as input for a data base that factors in concentration averaging assumptions to calculate actual amounts of GTCC LLW from light water reactors. See Appendix E-5 for GTCC LLW estimates after concentration averaging.

The amount of LWR waste that exceeds Class C limits depends upon plant type and fuel cycle. Management practices at various types of operating plants were reviewed to (a) estimate the amount of activated metal and process LLW generated per fuel cycle that could exceed Class C limits and (b) determine routine generation rates. Component-specific radionuclide content and distribution were calculated for activated metals components. Empirical data from actual LWR waste were used to estimate radionuclide content and distribution for process wastes.

This study found that activated metals and process wastes exceeding Class C limits represent a relatively small portion of the LLW routinely generated during LWR operations. The estimated generation rates at boiling water reactors (BWRs) ranged from a low of 0.13 m^3 (4.5 ft^3) to a high of 0.21 m^3 (7.4 ft^3) per plant per fuel cycle. The estimated generation rates at pressurized water reactors (PWRs) ranged from a low of 0.050 m^3 (1.7 ft^3) to a high of 0.090 m^3 (3.2 ft^3) per plant per fuel cycle.

Full reactor coolant system decontamination with fuel assemblies remaining in the vessel will probably result in some deep bed cation resins that exceed Class C limits. The estimated volume of this decontamination resin waste for a typical BWR was 2.83 m^3 (100 ft^3) and for a PWR was 4.25 m^3 (150 ft^3), assumed to be generated at each plant once during its operating life.

Volumes and activities were also estimated for decommissioning activated metals that will exceed Class C limits. Six typical LWR plants were selected for detailed analysis. Plant-specific drawings and data were used to model the reactor vessel internals and conditions for each plant. Component-specific radionuclide content and distributions were calculated, and volumes from decommissioning were estimated.

Activated metals from nuclear reactor decommissioning represent the largest (in terms of volume and radioactivity) potential source of nuclear utility LLW that exceeds Class C limits. The estimated volume of activated metals that exceed Class C limits immediately after decommissioning range from 3.14 m^3 (111 ft^3) to 10.9 m^3 (386 ft^3) per reactor, depending upon the reactor model analyzed. The results from the reactor models indicate that plants should be modeled and analyzed individually due to variances in geometry and operating history, which can significantly influence the results.

FOREWORD

In 1990, WMG Inc. was engaged by EG&G Idaho to incorporate WMG's extensive experience with nuclear power plant waste characterization and shipping into the Department of Energy Greater-Than-Class C Low-Level Waste (GTCC LLW) Management Program. Those efforts led to two reports in support of the GTCC LLW Management Program, issued in 1991: DOE/LLW-114F, *Greater-Than-Class C Low-Level Radioactive Waste Light Water Reactor Projections*, and DOE/LLW-114I, *Impact of Concentration Averaging Low-Level Radioactive Waste Volume Projections*. To a large extent, both reports relied on data representative of LLW that was characterized, packaged, and shipped for disposal.

DOE/LLW-114F, initially prepared by P. Tuite, K. Tuite, A. Levin, and M. O'Kelley of WMG, has since been changed and updated to support Revision 1 of DOE/LLW-114; Section 5 has been rewritten to reflect new information on decommissioning waste, and volume estimates for some waste types have been updated based on more recent experience. This document has been renumbered as DOE/LLW-114A-3.

CONTENTS

1. INTRODUCTION	1
1.1 Background	2
2. TECHNICAL APPROACH	3
2.1 LWR Operations Parameters	3
2.2 Study Parameters	9
2.2.1 Operations LLW Exceeding Class C Limits	9
2.2.2 Decommissioning LLW Exceeding Class C Limits	14
3. GENERATION AND ACTIVITY OF OPERATIONS ACTIVATED METALS	15
3.1 Activated Metals from BWR Operations	15
3.1.1 Control Rod Blades	17
3.1.2 Local Power Range Monitor Strings	19
3.1.3 IRM/SRM Dry Tubes	20
3.2 Activated Metals from PWR Operations	20
3.2.1 Thimble Plug Assemblies (Westinghouse)	21
3.2.2 In-Core Instrument Strings	23
3.2.3 Primary Sources	25
3.3 Activated Metals Summary	26
4. GENERATION AND ACTIVITY OF PROCESS WASTE	28
4.1 Cartridge Filters	28
4.1.1 BWR Operations	28
4.1.2 PWR Operations	30
4.1.3 Radionuclide Content and Distribution	32
4.2 Full RCS Decontamination Resins	33
4.2.1 Resin Generation Processes	33
4.2.2 Projections for Decontamination Resins	35
4.3 Process Waste Summary	36
5. GENERATION AND ACTIVITY OF DECOMMISSIONING ACTIVATED METALS .	39

5.1 LWR Components	39
5.1.1 BWR Decommissioning Internals	39
5.1.2 PWR Decommissioning Internals	41
5.2 Methodology Used to Determine Component Activities and Classification Status ..	42
5.2.1 Reference Plants	44
5.2.2 One-Dimensional Transport Calculations	44
5.2.3 Point Neutron Activation Calculations	45
5.2.4 Operating Histories	47
5.2.5 Radial ANISN Models	47
5.2.6 Radial ANISN Model Calculations	48
5.2.7 Axial ANISN Models	48
5.2.8 Axial ANISN Model Calculations	49
5.3 Description of a Typical Reference Plant Case	49
5.3.1 Radial ANISN Model	49
5.3.2 Axial ANISN Model	54
5.4 Results of Reference Plant Cases	54
5.4.1 BWR Case Results	54
5.4.2 Westinghouse PWR Case Results	57
5.4.3 Other PWR Case Results	57
5.4.4 Time Dependence of Case Results	57
6. SUMMARY AND CONCLUSIONS	66
7. REFERENCES	71

FIGURES

1. GTCC LLW is a fraction of all low-level radioactive waste	1
2. Diagram showing the location of some BWR decommissioning components	12
3. Diagram showing the location of some PWR components	13
4. B&W radial ANISN case thermal flux results—use of SAILOR vs CASK-81 cross-section libraries	46
5. Arrangement of surveillance capsules in the large Westinghouse reactor vessel	52
6. Westinghouse large PWR thermal neutron flux—surveillance capsule normalized	53

TABLES

1. BWR reactors	4
2. PWR reactors	6
3. Master list of LWR waste types evaluated.	10
4. BWR operations components (activated metals) that individually exceed Class C limits—volume and activity estimates.	16
5. Control rod blade generation per BWR.	18
6. PWR operations activated metals components that individually exceed Class C limits—volume and activity estimates	22
7. Summary of operations activated metals components that individually exceed Class C limits—volumes.	27
8. Cartridge filters that can individually exceed Class C limits—volumes and weights	29
9. Cartridge filters that individually exceed Class C limits—volume and activity estimates	32
10. Full PWR RCS decontamination resin LLW volume estimates (for comparison purposes only)	35
11. Fuel in/full RCS decontamination resin exceeding Class C limits.	37
12. Summary of process LLW exceeding Class C limits—volumes	38
13. Reactor decommissioning components evaluated in this study	40
14. Summary of reference reactors analyzed to predict activation levels of decommissioning components	44
15. Large Westinghouse PWR—radial ANISN model input	50
16. Westinghouse large reference Case D axial ANISN model input summary	51
17. General Electric BWR-4 reference Case A—results of activation analysis of decommissioning components, after 40 years operation, immediately after shutdown ...	55
18. General Electric BWR-6 reference Case B—results of activation analysis of decommissioning components, after 40 years operation, immediately after shutdown ...	56
19. Westinghouse intermediate reference Case C—results of activation analysis of decommissioning components, after 40 years operation, immediately after shutdown ...	58

20. Westinghouse large reference Case D—results of activation analysis of decommissioning components, after 40 years operation, immediately after shutdown	59
21. Combustion Engineering reference Case E—results of activation analysis of decommissioning components, after 40 years operation, immediately after shutdown . . .	60
22. Babcock & Wilcox reference Case F—results of activation analysis of decommissioning components, after 40 years operation, immediately after shutdown	61
23. BWR-4 core shroud activation as a function of time	63
24. Large Westinghouse lower core support barrel activation as a function of time	64
25. GTCC classification status for decommissioning components considered in each reference LWR as a function of plant operating time	65
26. Percentages of LLW components that individually exceed Class C limits	67
27. Operations activated metals and process wastes that exceed Class C limits	68
28. Decommissioning activated metals that exceed Class C limits	70

Appendix A-3: Basis for Greater-than-Class C Low-Level Radioactive Waste Light Water Reactor Projections

1. INTRODUCTION

This study evaluates three categories of light water reactor (LWR) waste that may exceed Class C limits: activated metals resulting from operations, process wastes such as cartridge filters and resins, and activated metals resulting from decommissioning. The objective of the study is to (a) identify and characterize the components in each waste category that individually exceed Class C limits, (b) estimate the amounts of these components, and (c) estimate the radionuclide content and distribution within that waste.

This report does not consider concentration averaging (the practice of packaging components with a higher radionuclide concentration under 10 CFR 61 with components that have a lower concentration to produce a waste package with a combined concentration that is suitable for disposal in a low-level waste disposal facility). As such, the estimates given in this report may not represent final estimates of greater-than-Class C low-level radioactive waste (GTCC LLW) from light water reactors; instead, these estimates serve as a baseline upon which concentration averaging calculations can be computed. These estimates are used only as input to a data base that factors in concentration averaging assumptions to calculate amounts of GTCC LLW from light water reactors. Figure 1 illustrates how applying concentration averaging factors to these estimates yields the actual amount of LWR waste that will be classified as GTCC LLW.

Section 2 presents the technical approach to the study and the LWR fuel cycle information used. Sections 3, 4, and 5 address generation and activity of activated metals from operations, process wastes, and activated metals from decommissioning. Section 6 discusses the summary and conclusions for this study, and Section 7 presents the study references.

Figure 1. GTCC LLW is a fraction of all low-level radioactive waste.

1.1 Background

A study conducted in 1991¹ evaluated three types of light water reactor wastes that may exceed Class C limits: activated metals resulting from operations, process wastes such as cartridge filters and resins, and activated metals resulting from decommissioning. A range of reactor models was considered for both boiling water reactors (BWRs) and pressurized water reactors (PWRs). That report described the various LLW components found in each of the three waste type categories, their physical characteristics, and the portions that individually exceed Class C limits. This report updates the decommissioning data, percentages, and volumes reported in the 1991 study.

The amount of GTCC LLW generated by LWRs depends upon plant type and fuel cycle. To define LWR operating parameters, a list of LWRs was prepared. This list includes startup dates, plant type, reactor model, and information on plant-specific fuel cycles and refueling outage times.^{2,3,4,5} Management practices at operating plants were then reviewed to estimate the amount of LLW exceeding Class C limits that would be generated per fuel cycle and to determine routine waste generation rates as a result of operations. These estimates included the amounts of such LLW from operations activated metals and process LLW. Component-specific radionuclide content and distribution were calculated for activated metals. Empirical data from actual LWR waste were used to estimate radionuclide content and distribution for process waste.

Volumes and activities were also estimated for decommissioning activated metals components that individually exceed Class C limits. The methods used were similar to those described in the Nuclear Regulatory Commission (NRC) reports^{6,7} on decommissioning waste. Six typical LWR plants—four PWRs and two 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 from four of the six plants were used to normalize the calculated neutron fluxes. The normalized fluxes were then used to perform activation calculations for vessel internal components in each plant-specific reactor model. Specific radionuclide content and distributions were calculated for each component. The six plant-specific reactor models were analyzed to identify which components will exceed Class C limits at the time of decommissioning.

As noted earlier, these estimates are used only as input to a data base that factors in concentration averaging and other considerations to calculate amounts of LWR waste that will be classified as GTCC LLW. See Appendix E-5^a for a discussion of concentration averaging.⁸

a. References to appendices in this report are to the appendices in *Greater-Than-Class C Low-Level Radioactive Waste Characterization: Estimated Volumes, Radionuclide Activities, and Other Characteristics* (DOE/LLW-114, Revision 1).

2. TECHNICAL APPROACH

The study objective is to provide quantitative estimates of volumes and radioactivity for LLW exceeding Class C limits that is generated by current-technology LWRs. To achieve this objective, the study considers parameters that describe LWR plant performance and the impact of this performance on generation of LLW components that individually exceed Class C limits. These parameters, identified based on References 2-5, are discussed below.

2.1 LWR Operations Parameters

There are 37 General Electric BWR plants ranging in size from the 504 MWe Vermont Yankee plant to the 1205 MWe Perry plant. Table 1 summarizes the overall BWR plant parameters relevant to this study. Table 2 illustrates similar information for PWR stations broken down by reactor vendor. As shown, there are 7 Babcock & Wilcox plants, 15 Combustion Engineering plants, and 51 Westinghouse plants.

The parameters shown on these tables include

- Reactor Type — For BWRs: Models 1 through 6; for Westinghouse PWRs: Models 2, 3, or 4
- Plant Name, Unit, and MWe
- Startup Date — Actual date of commercial operation
- License Expiration Date — Year that plant's 40-year license will expire
- Cycle Months — Length of the plant's fuel cycle
- Outage Start and Finish Dates — Outages planned for 1993 and 1994
- Days Shut Down — The elapsed time between start and finish dates of the 1993-1994 outages
- Capacity Factors — The plant's values for capacity factors for the periods 1988-1990 and 1991-1993; 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 1988-1990 or the 1991-1993 time period.

These operations parameters provided the overall framework for defining LWR fuel cycles and the basis for determining generation rates of LLW components that individually exceed Class C limits.

Table 1. BWR reactors.^a

Number	Type	Plant name	Unit	MWe (net)	Commercial Operation Startup (month/year)	License Expiration Date (year) ^b	Fuel cycle (mos)	Outage start	Outage finish	Days shut down	Cap. factor '88-'90 (percent)	Cap. factor '91-'93 (percent)	Best cap. factor (percent)
1	1	Big Rock Point	—	67	3/63	2002	12.0	9/10/94	—	40	64.73	62.79	64.73
2	2	Nine Mile Point	1	610	12/69	2009	22.2	2/19/93	—	55	7.55	68.30	68.30
3	2	Oyster Creek	1	610	12/69	2009	22.0	9/10/94	—	77	59.89	71.00	71.00
4	3	Dresden	2	772	6/70	2010	16.0	1/16/93	4/18/93	92	62.95	48.86	62.95
5	3	Millstone	1	660	3/71	2010	19.7	1/15/94	—	70	87.97	61.28	87.97
6	3	Monticello	—	536	6/71	2011	16.7	9/15/94	—	40	81.80	83.09	83.09
7	3	Dresden	3	773	11/71	2011	15.1	3/12/94	6/12/94	92	69.03	50.74	69.03
8	4	Vermont Yankee	—	504	11/72	2012	16.3	8/28/93	10/6/93	53	83.85	82.95	83.85
9	3	Pilgrim	1	670	12/72	2012	24.0	4/3/93	5/30/93	57	34.54	72.59	72.59
10	3	Quad Cities	1	769	2/73	2012	14.9	3/12/94	6/26/94	105	73.58	61.40	61.40
11	3	Quad Cities	2	769	3/73	2012	15.2	3/6/93	5/29/93	84	68.76	59.33	68.76
12	4	Cooper	—	764	7/74	2014	17.0	3/7/93	5/1/93	56	68.91	72.05	72.05
13	4	Peach Bottom	2	1100	7/74	2014	22.0	9/10/94	—	60	37.70	65.81	65.81
14	4	Browns Ferry	1	1065	8/74	2013	—	3/19/85	—	—	0	0	0
15	4	Peach Bottom	3	1100	12/74	2014	24.0	9/11/93	—	—	27.57	66.39	66.39
16	4	Duane Arnold	—	538	2/75	2014	17.0	7/29/93	—	60	65.70	76.42	76.42
17	4	Browns Ferry	2	1065	3/75	2014	18.0	10/1/94	—	35	0	63.99	63.99
18	4	J. A. FitzPatrick	—	757	7/75	2015	—	9/93	—	28	70.42	37.85	70.42
19	4	Brunswick	2	754	11/75	2015	18.0	3/94	—	91	56.29	41.10	56.29
20	4	Edwin I. Hatch	1	741	12/75	2014	18.0	9/21/94	11/19/94	58	71.82	77.46	77.46
21	4	Browns Ferry	3	1065	3/77	2016	—	3/9/85	—	—	0	0	0
22	4	Brunswick	1	767	3/77	2016	15.0	Fall 1993	—	84	60.00	28.78	60.00
23	4	Edwin I. Hatch	2	761	9/79	2018	18.0	3/16/94	5/14/94	59	72.36	70.85	72.36
24	4/5	Susquehanna	1	1032	6/83	2022	16.2	9/11/93	—	55	76.93	73.90	76.93
25	5	LaSalle County	1	1036	1/84	2022	16.1	3/5/94	6/5/94	92	71.37	72.22	72.22

Table 1. (continued).

Number	Type	Plant name	Unit	MWe (net)	Commercial Operation Startup (month/year)	License Expiration Date (year) ^b	Fuel cycle (mos)	Outage start	Outage finish	Days shut down	Cap. factor '88-'90 (percent)	Cap. factor '91-'93 (percent)	Best cap. factor (percent)
26	5	LaSalle County	2	1036	10/84	2024	15.5	9/4/93	—	77	64.67	71.72	71.72
27	5	WNP-2	2	1100	12/84	2024	10.0	4/16/94	6/15/94	60	61.73	58.60	61.73
28	4/5	Susquehanna	2	1038	2/85	2024	16.2	3/12/94	—	55	75.65	81.66	81.66
29	6	Grand Gulf	—	1142	7/85	2024	16.5	10/2/93	—	43	75.55	76.62	76.62
30	4	Limerick	1	1055	2/86	2025	22.5	1/28/94	3/13/94	45	63.10	83.27	83.27
31	6	River Bend	—	936	6/86	2025	21.0	4/16/94	6/8/94	53	71.59	59.74	71.59
32	4	Hope Creek	—	1031	12/86	2026	16.8	3/5/94	4/22/94	41	77.34	82.91	82.91
33	6	Clinton	1	930	4/87	2027	15.0	9/26/93	12/15/93	80	50.14	68.71	68.71
34	6	Perry	1	1205	11/87	2026	17.0	2/5/94	—	85	60.64	64.22	64.22
35	4	Fermi	2	1075	1/88	2025	16.0	3/12/94	—	56	—	—	—
36	5	Nine Mile Point	2	1080	4/88	2027	16.0	9/11/93	—	60	—	—	—
37	4	Limerick	2	1055	1/90	2029	22.0	1/23/93	3/23/93	59	—	—	—
Averages				864			17.6			63	58.06	61.37	65.50

a. Blank spaces in Tables 1 and 2 indicate areas of information that were not available from the references.

b. Assumes renewal of licenses that were issued before 1982, but does not reflect any license extension.

Source: This table was compiled from information found in References 2-5.

Table 2. PWR reactors.^a

Number	Type	Plant name	Unit	MWe (net)	Commercial Operation Startup (month/year)	License Expiration date (year) ^b	Fuel Cycle (mos)	Outage start	Outage finish	Days shut down	Cap. factor '88-'90 (percent)	Cap. factor '91-'93 (percent)	Best cap. factor (percent)
B&W Plants													
1		Oconee	1	846	7/73	2013	16.4	4/28/94	6/22/94	55	83.97	80.72	83.97
2		Oconee	2	846	9/74	2013	16.4	9/29/94	11/23/94	55	76.39	84.12	84.12
3		Three Mile Island	1	786	9/74	2014	22.2	9/17/93	—	38	83.48	86.18	86.18
4		Arkansas Nuclear One	1	836	12/74	2014	18.0	9/1/93	—	52	51.19	82.71	82.71
5		Oconee	3	846	12/74	2014	15.4	12/28/93	2/21/94	56	84.60	79.07	84.60
6		Crystal River	3	821	3/77	2017	22.0	4/7/94	6/6/94	60	59.17	77.58	77.58
7		Davis-Besse	—	877	7/78	2017	16.2	10/1/94	11/23/94	53	53.07	82.15	82.15
Averages							18.1			53	70.267	81.790	83.044
CE Plants													
1		Palisades	—	768	12/71	2011	15.4	5/7/93	6/26/93	50	47.61	62.74	62.74
2		Maine Yankee	—	840	12/72	2012	15.0	8/28/93	—	—	77.04	76.45	76.45
3		Fort Calhoun	—	478	9/73	2013	12.0	9/18/93	11/13/93	56	66.34	70.69	70.69
4		Calvert Cliffs	1	825	5/75	2015	—	10/1/93	10/30/93	30	35.34	76.10	76.10
5		Millstone	2	862	12/75	2015	18.0	6/18/94	8/25/94	68	69.59	56.57	69.59
6		St. Lucie	1	839	12/76	2016	18.0	10/1/94	—	53	81.03	84.12	84.12
7		Calvert Cliffs	2	825	4/77	2016	21.0	2/4/94	5/4/94	90	36.22	68.39	68.39
8		Arkansas Nuclear One	2	858	3/80	2018	18.0	3/12/94	4/22/94	41	73.14	79.06	79.06
9		San Onofre	2	1070	8/83	2022	18.0	6/93	8/93	61	80.06	78.88	80.06
10		St. Lucie	2	839	8/83	2023	18.0	2/15/94	—	53	83.21	80.46	83.21
11		San Onofre	3	1080	4/84	2023	18.0	9/93	12/93	91	75.88	79.69	79.69
12		Waterford	3	1075	9/85	2025	18.0	3/94	—	52	78.34	82.77	82.77
13		Palo Verde	1	1221	1/86	2025	14.0	10/93	—	70	39.47	71.68	71.68
14		Palo Verde	2	1221	9/86	2026	—	1/8/94	2/20/94	43	52.95	70.33	70.33
15		Palo Verde	3	1221	1/88	2027	17.0	3/12/94	05/31/94	80	—	—	—
Averages							17.0			60	64.02	74.14	75.35

Table 2. (continued).

Number	Type	Plant name	Unit	MWe (net)	Commercial Operation Startup (month/year)	License Expiration date (year) ^b	Fuel Cycle (mos)	Outage start	Outage finish	Days shut down	Cap. factor '88-'90 (percent)	Cap. factor '91-'93 (percent)	Best cap. factor (percent)
Westinghouse Plants													
1	2	Haddam Neck	—	590	1/68	2007	15.7	11/12/94	1/5/95	54	48.46	73.97	73.97
2	2	R. E. Ginna	—	470	7/70	2009	10.8	3/4/94	—	40	81.36	84.66	84.66
3	2	Point Beach	1	485	12/70	2010	10.6	4/2/94	—	36	83.89	84.41	84.41
4	2	Robinson	2	683	3/71	2010	15.0	9/93	—	63	50.35	70.80	70.80
5	2	Point Beach	2	485	10/72	2012	10.6	10/8/94	—	38	84.12	85.66	85.66
6	3	Surry	1	781	12/72	2012	16.5	1/21/94	3/25/94	63	51.27	87.05	87.05
7	3	Turkey Point	3	666	12/72	2012	18.0	4/4/94	5/19/94	38	57.05	57.00	57.05
8	3	Surry	2	781	5/73	2013	16.0	11/1/94	12/30/94	60	49.70	72.14	72.14
9	3	Turkey Point	4	666	9/73	2013	18.0	10/18/94	12/3/94	46	53.40	55.85	55.85
10	2	Prairie Island	1	503	12/73	2013	18.0	6/7/94	—	44	86.40	85.05	86.40
11	4	Zion	1	1040	12/73	2013	11.0	9/25/93	3/27/94	182	57.69	56.27	57.69
12	2	Kewaunee	—	503	6/74	2014	11.2	3/26/94	5/6/94	41	82.13	81.22	82.13
13	4	Indian Point	2	970	8/74	2013	19.5	1/30/93	—	75	66.22	68.14	68.14
14	4	Zion	2	1040	9/74	2013	—	11/14/92	2/25/93	104	62.10	57.72	62.10
15	2	Prairie Island	2	500	12/74	2014	18.0	10/6/93	—	42	83.08	82.08	83.08
16	4	Donald C. Cook	1	1020	8/75	2015	15.0	2/5/94	—	—	70.88	78.47	78.47
17	4	Indian Point	3	965	8/76	2016	—	5/1/93	—	28	65.84	52.21	65.84
18	3	Beaver Valley	1	810	10/76	2016	14.0	9/16/94	—	70	68.01	65.36	68.01
19	4	Salem	1	1106	6/77	2016	13.7	10/2/93	12/12/93	71	66.40	61.27	66.40
20	3	Joseph M. Farley	1	814	12/77	2017	18.0	3/4/94	4/28/94	55	86.37	82.26	86.37
21	3	North Anna	1	911	6/78	2018	15.0	9/1/94	10/18/94	47	77.27	69.91	77.27
22	4	Donald C. Cook	2	1060	7/78	2018	16.0	8/6/94	—	—	47.69	59.75	59.75
23	3	North Anna	2	909	12/80	2020	16.5	9/4/93	11/16/93	74	82.81	84.81	84.81
24	3	Joseph M. Farley	2	824	7/81	2021	18.0	Fall 93	—	—	82.74	79.71	82.74
25	4	Sequoyah	1	1148	7/81	2020	17.0	4/2/93	4/6/93	65	54.45	55.86	55.86
26	4	Salem	2	1106	10/81	2021	16.0	9/24/94	11/22/94	59	65.48	61.10	65.48
27	4	W. B. McGuire	1	1129	8/81	2021	12.9	7/18/94	9/21/94	58	64.19	63.94	64.19
28	4	Sequoyah	2	1148	6/82	2021	16.0	4/1/94	6/5/94	65	56.69	61.77	61.77
29	3	Virgil C. Summer	—	885	1/84	2022	15.0	9/9/94	—	100	70.00	80.09	80.09
30	4	W. B. McGuire	2	1129	3/84	2023	13.8	11/21/93	1/25/94	65	70.65	74.43	74.43
31	4	Callaway	—	1125	4/85	2024	18.0	9/30/93	—	56	82.12	85.92	85.92

Table 2. (continued).

Number	Type	Plant name	Unit	MWe (net)	Commercial Operation Startup (month/year)	License Expiration date (year) ^b	Fuel Cycle (mos)	Outage start	Outage finish	Days shut down	Cap. factor '88-'90 (percent)	Cap. factor '91-'93 (percent)	Best cap. factor (percent)
32	4	Diablo Canyon	1	1073	5/85	2024	16.0	3/12/94	5/7/94	56	74.10	83.44	83.44
33	4	Catawba	1	1129	6/85	2025	13.1	10/30/93	1/8/94	70	73.93	70.65	73.93
34	4	Byron	1	1105	9/85	2025	15.8	9/10/94	11/13/94	64	75.26	76.89	76.89
35	4	Wolf Creek	—	1135	9/85	2025	15.4	9/94	—	—	78.76	72.29	78.76
36	4	Diablo Canyon	2	1087	3/86	2025	16.0	9/10/94	11/4/94	55	76.13	84.09	84.09
37	4	Millstone	3	1146	4/86	2026	22.0	7/93	—	56	75.67	52.36	75.67
38	4	Catawba	2	1129	8/86	2026	14.4	5/5/94	7/12/94	68	63.47	82.04	82.04
39	3	Shearon Harris	—	860	5/87	2027	18.0	3/94	—	70	73.07	79.65	79.65
40	4	Alvin W. Vogtle	1	1100	6/87	2027	16.5	9/17/94	10/31/94	44	78.87	86.99	86.99
41	4	Byron	2	1105	8/87	2027	15.0	9/4/93	11/7/93	64	62.57	79.36	79.36
42	3	Beaver Valley	2	833	11/87	2027	14.6	9/17/93	—	60	69.60	80.07	80.07
43	4	Braidwood	1	1120	7/88	2027	17.2	3/5/94	5/1/94	57	—	—	—
44	4	South Texas Project	1	1250	8/88	2028	—	2/5/93	1/31/94	360	—	—	—
45	4	Braidwood	2	1120	10/88	2028	15.8	9/10/94	11/13/94	64	—	—	—
46	4	Alvin W. Vogtle	2	1097	5/89	2029	16.5	9/10/93	10/26/93	46	—	—	—
47	4	South Texas Project	2	1250	6/89	2029	19.0	2/27/93	—	67	—	—	—
48	4	Seabrook	—	1150	7/90	2030	16.0	3/26/94	5/20/94	55	—	—	—
49	4	Comanche Peak	1	1150	8/90	2030	15.7	10/1/93	—	70	—	—	—
50	4	Comanche Peak	2	1150	8/93	2033	16.0	—	—	—	—	—	—
51	4	Watts Bar	1	1177	6/94 ^c	2034	—	9/15/94	11/24/94	70	—	—	—
Averages							15.7			68.0	69.29	73.02	74.51

a. Blank spaces in Tables 1 and 2 indicate areas of information that were not available from the references.

b. Assumes renewal of licenses that were issued before 1982, but does not reflect any license extension.

c. Reference 2 predicted that Watts Bar 1 would begin commercial operation in June 1994. As of September 1994, commercial operation had not yet begun.

Source: This table was compiled from information found in References 2–5.

2.2 Study Parameters

A master list was prepared of LWR waste types that might individually exceed Class C limits. Waste types that were already included in the DOE data base, and the reason for their inclusion, were reviewed and revised. The waste types were characterized, based on the authors' experience, to determine what components may individually exceed Class C limits. Table 3 lists all LWR components that were evaluated as possible sources of LLW components exceeding Class C limits. The remainder of this report discusses only those that were determined to have amounts of such LLW of significant amount or activity to warrant inclusion in estimates by the GTCC LLW Program. Figures 2 and 3 illustrate the typical locations of some of the components in BWRs and PWRs.

Reactors components were categorized into three waste types: operations activated metals, process waste, and decommissioning activated metals. Of these waste types, operations activated metals and cartridge filters (process waste) are routinely generated during ongoing operations. Decontamination resins are generated intermittently. Decommissioning metals are generated at plant closure. The following section discusses how operations parameters such as fuel cycles, capacity factors, etc., are used to make projections for each waste generation category (i.e., operations activated metals, process waste, and decommissioning activated metals).

2.2.1 Operations LLW Exceeding Class C Limits

The generation of operations waste that exceeds Class C limits is largely dependent on individual plant operation. The information shown on Tables 1 and 2 was used to define the following parameters relevant to generation of LLW that exceeds Class C limits.

- **Operating Cycles**—The information in Table 1 indicated that the average BWR fuel cycle was over 17 months, with 63 days of downtime between cycles for refueling. A 20-month operating cycle, which includes three months for outages, was assumed.

As shown in Table 2, fuel cycle lengths varied among PWR reactor vendors. For PWRs, two different operating 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 operating cycles include three months for outages.

- **Exposure per Cycle**—For activated metals, the effective full-power days (EFPD) per cycle determines a component's exposure and its radioactivity. The EFPD per cycle was calculated for each general reactor type as follows: The capacity factor was multiplied by the operating cycle and divided by the fuel cycle to define an average activation factor. This factor was multiplied by the average fuel cycle duration for each plant type 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 fuel cycle at a 76% activation factor. Two different cycle exposures were used for PWRs: 400 EFPD for Babcock & Wilcox and Westinghouse plants, which corresponds to a 16-month fuel cycle at an 83% activation factor; and 430 EFPD for Combustion Engineering plants, which corresponds to a 17-month fuel cycle at an 83% activation factor.

Table 3. Master list of LWR waste types evaluated.

	Potentially include GTCC LLW?
Activated metals from operations	
BWR operations	
Control rod blades—control cell mode of plant operation	Yes
Control rod blades—normal or conventional mode of plant operation	Yes
Local Power Range Monitor (LPRM) strings	Yes
Intermediate Range Monitor/Source Range Monitor (IRM/SRM) dry tubes	Yes
Poison curtains ^a	Yes
Fuel flow channels ^a	No
Control rod blade stellite bearings ^a	Yes
PWR operations	
Thimble plug assemblies	
Westinghouse thimble plug assemblies	Yes
B&W orifice rod assemblies ^a	Yes
Combustion Engineering control element assemblies ^a	Yes
Instrument strings	
Westinghouse thimble tubes	Yes
B&W in-core detectors	Yes
Combustion Engineering instrument strings	Yes
Primary sources	Yes
Secondary sources ^a	No
Fission chambers ^a	No
Burnable poison rod assemblies (BPRAs) ^a	No
Rod control cluster assemblies ^a	No
Control rod flow plugs ^a	No
Control rod flow channels ^a	No
Control rod drives ^a	No
Process wastes	
BWR & PWR decontamination resins	
Full RCS with fuel out	No
Full RCS with fuel in - anion	No
Full RCS with fuel in - cation	Yes
Cartridge filters	
BWR	
Control rod drive filter - not spud end	No
Control rod drive filter - two at spud end	Yes
Fuel pool filters	Yes
Vacuum filters	Yes
PWR	
Reactor coolant system filters	Yes
Seal water injection filters	Yes
Cavity drain (seal water return) filters	Yes
Fuel pool filters	Yes
Letdown filters	No
B&W crud tank filters	Yes
Sludge from reactor cavity ^a	Yes
Solidified liquid	No

Table 3. (continued).

	Potentially include GTCC LLW?
Activated metals from decommissioning	
BWR decommissioning	
Top fuel guide ^b	Yes
Core shroud	Yes
Fuel support castings (orificed fuel supports) ^b	No
Core support plate ^b	No
PWR decommissioning	Yes
Upper core support plate ^b	Yes
Core baffle (shroud)	No
Thermal shield	Yes
Lower core barrel	Yes
Lower core support plate ^b	Yes
Lower support columns ^a	No
Contaminated solids ^a	
Fuel assembly hardware—end fitting/spacer grids ^a	Yes
Shim rods	No
Scrap bolts and small parts	No
Reactor vessel maintenance scrap metal	No
Glassware	No
Contaminated equipment ^a	Yes
Trash—compactible and noncompactible ^a	Yes
Flux wire ^a	Yes
Wire	No
HTG metal-clad blocks	No
Research/test reactors	No
Cartridge filters from decommissioning	No
Demolition rubble	No

a. These components were listed in 1991 as waste types that may exceed Class C limits; when reevaluated in 1994 it was determined that they contribute such a small amount (if any) that they can be neglected for this report.

b. These components were not considered in 1991, but were evaluated in 1994.

Figure 2. Diagram showing the location of some BWR decommissioning components.

Figure 3. Diagram showing the location of some PWR decommissioning components.

These parameters, summarized below, were used to estimate volumes and activities of LLW exceeding Class C limits that is generated during operations, as described in Sections 3 and 4.

	Fuel cycle (months)	Operating cycle (months)	Capacity factor	Activation factor	EFPD (days)
BWR					
General Electric	17	20	65%	76%	390
PWR					
Westinghouse	16	19	70%	83%	400
Babcock & Wilcox	16	19	70%	83%	400
Combustion Engineering	17	20	70%	83%	430

2.2.2 Decommissioning LLW Exceeding Class C Limits

This study assumed that one full Reactor Coolant System (RCS) decontamination project will be performed with fuel assemblies in the vessel for all reactor plants shutting down after the year 2013. The resulting resin wastes that exceed Class C limits are assumed to be generated just once during each plant's life, and so are reported as decommissioning wastes in DOE/LLW-114 Revision 1.

Decommissioning LLW components that individually exceed Class C limits consist primarily of activated metals; how much activated metals waste will exceed Class C depends on the integrated flux to which vessel internals are exposed during the plant's operating lifetime. The variables that define integrated exposure are the plant's power level and capacity factor. A 40-year operating lifetime is used as a fixed parameter for all plants to predict activation levels of decommissioning waste metals.^b

For BWRs, the average for the best capacity factors, 65%, was used in conjunction with the 40-year lifetime assumption to estimate the radionuclide activities at the time of BWR decommissioning. For PWRs, the average capacity factors varied among vendors. For this study, an overall average capacity factor of 70% was used in conjunction with the 40-year lifetime assumption to estimate the radionuclide activities at the time of PWR decommissioning. These parameters were used to estimate volumes and activities of decommissioning LLW that individually exceeds Class C limits, as described in Section 5.

b. The DOE/LLW-114 Revision 1 report makes future projections using reactor shutdown timing assumptions in Appendix A-2, which predicts early shutdown or life extension of some reactors. However, the activity of decommissioning activated metal components used in this report is based on the analysis described here, which assumes a 40-year life for each reactor.

3. GENERATION AND ACTIVITY OF OPERATIONS ACTIVATED METALS

Activated metals from nuclear utility operations consist of the nonfuel-bearing components that are activated through exposure to neutron flux within LWRs; these components are periodically replaced during normal operations. The generation of activated metals depends on the plant's type, size, fuel cycle length, and average exposure during each fuel cycle. The LLW quantities that exceed Class C limits are a fraction of the volume discharged.

After removal from the core, the LLW components are stored in the spent fuel pool until sufficient quantities are available for shipment. This study assumes that current practices will continue. Thus, activities of operations components were estimated after two cycles of decay in the spent fuel pool. This decay does not affect 10 CFR 61⁹ 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 activated metal components routinely generated during LWR operations. It presents the estimated volumes and curie content for activated metal components that individually exceed Class C limits, generated from the operation of BWRs and PWRs, based on the operating cycles discussed in Section 2. Section 5 discusses decommissioning components.

3.1 Activated Metals from BWR Operations

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 (i.e., the activation factor).

These assumed average conditions were compared to the actual operating cycle histories of 10 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 10 plants. Since this average included unplanned outages, the 17-month typical fuel cycle assumption is reasonable. The actual average total exposure per plant 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 2. For conservative purposes, the study 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 estimated based on experience in characterizing actual BWR components for disposal. Characterization methods include taking radiation profile measurements and multiplying the results by scaling factors based on ratios of concentrations of critical material elements relative to cobalt concentrations. The BWR materials compositions used are based on composition data from manufacturer heat data on actual material used to manufacture the components considered.

BWR activated metal components that may exceed Class C limits are discussed below. Table 4 presents the estimated activities of the major radionuclides for these BWR components at the time of shipment, and their status relative to 10 CFR 61 Tables 1 and 2 sums of fractions. The activities are based on 40 months of decay after discharge. For local power range monitor (LPRM) strings and intermediate range monitor/source range monitor (IRM/SRM) dry tubes, the classification status is shown for the entire component as well as for the "hot" (higher radiological activity) end by itself.

The following section presents the basis for projecting volumes and activities of activated metals exceeding Class C limits, generated from routine BWR operations.

Table 4. BWR operations components (activated metals) that individually exceed Class C limits—volume and activity estimates.

Component type	Control rod blade (conventional) ^a	Control rod blade (controlled cell) ^a	LPRM string	IRM/SRM dry tube
Volume per component				
m ³	0.017	0.017	0.0028	0.0023
(ft ³)	(0.6)	(0.6)	(0.1)	(0.08)
Weight (lb)	205	205	50	39
Mass (kg)	93	93	23	18
Number of cycles	6	2	3	6
Exposure—EFPD (390 EFPD/cycle)	2,340	780	1,170	2,340
10 CFR 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
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

a. CRB activity levels for controlled cell CRBs were used for the data base calculations, because they exhibit higher activity levels than conventional mode CRBs.

3.1.1 Control Rod Blades

This study terms a "midsize" BWR as either Model 1, 2, 3, or 4, while a "large" BWR is either Model 5 or 6. A midsize BWR contains up to 137 control rod blades (CRBs), and a large BWR contains 185 CRBs. The CRBs are distributed radially throughout the core. The boron material in CRBs helps control the rate of fission within the reactor by absorbing neutrons. Each CRB volume is assumed to be 0.017 m^3 (0.6 ft^3). Weights are model-dependent and vary between 203 and 280 lb.

Control rod operating patterns vary by reactor plant, and CRBs are typically discharged in batches based on boron depletion. Most plants operate in a conventional control rod pattern, where alternating banks of deeply and shallowly inserted CRBs are slowly withdrawn from the core. Generally, CRBs that run in the conventional mode of operation receive six to seven cycles of exposure, which could cause the CRBs to individually exceed Class C limits.

Some plants also run in what is referred to as the controlled cell mode. In this mode, a 137-CRB core contains a group of 25 movable CRBs in a 5×5 array, or a 185-CRB core contains a group of 37 movable CRBs in a 5×5 array with three additional CRBs on each side. These movable CRBs are termed "controlled cell" CRBs, or "CRBs operating in the controlled cell mode." CRBs in the controlled cell mode are typically more fully inserted while the rest of the CRBs are often completely withdrawn; therefore, controlled cell CRBs receive more exposure than conventional mode CRBs and must be changed out more frequently. Half of the 25 to 37 CRBs that operate in the controlled cell mode are discharged each fuel cycle.

Waste generation rates for CRBs depend on the core size (137 vs 185 CRBs) and the mode of CRB operation. This study assumed that 10% of conventional mode CRBs and 25% of controlled cell CRBs may exceed Class C limits. Table 5 summarizes the CRB waste generation rates for each category and presents the estimated CRB waste volumes that exceed Class C limits per plant per fuel cycle. A weighted average of 2.9 such CRBs are expected per cycle for the midsize BWRs, and a weighted average of 3.5 such CRBs per cycle are expected for the large BWRs. These quantities were used to calculate input for the data base; results are reported in DOE/LLW-114, Revision 1.

CRB activity levels used for calculations were conservatively chosen to be that of controlled cell CRBs, which have higher activity than that of conventional mode CRBs. Table 4 indicates that activity estimates for controlled cell CRBs were based on two cycles, or 780 EFPD, of exposure. These CRBs contained 3420 Ci and had 10 CFR 61 Table 1 sum of fractions of 1.3. For CRBs run in the conventional mode of operation, curie estimates were based on CRBs with six fuel cycles of operation, or 2340 EFPD, of exposure. These CRBs contained 2570 Ci and had a 10 CFR 61 Table 1 sum of fractions of 1.4.

Table 5. Control rod blade generation per BWR.

	Midsize BWR		Large BWR	
	Conventional mode	Controlled cell mode ^a	Conventional mode	Controlled cell mode ^a
Number of plants	14	15	5	3
Total number of CRBs in core	137	137	185	185
Conventional mode CRBs				
Number of conventional CRBs in core	137	112	185	148
Number of waste conventional CRBs generated per cycle	14	11	19	15
Percent conventional CRBs that exceed Class C limits	10	10	10	10
Number of conventional CRBs that exceed Class C limits, generated per cycle	1.4	1.1	1.9	1.5
Controlled cell CRBs				
Number of controlled cell CRBs in core	0	25	0	37
Number of waste controlled cell CRBs generated per cycle	0	13	NA	19
Percent controlled cell CRBs that exceed Class C limits	NA	25	NA	25
Number of conventional CRBs that exceed Class C limits, generated per cycle	NA	3.25	NA	4.75
Total number that exceed Class C limits, generated per cycle	1.4	4.35	1.9	6.25
Total CRBs that exceed Class C limits per cycle m ³ (ft ³)	0.024 (0.84)	0.074 (2.6)	0.032 (1.14)	0.106 (3.75)

a. "Controlled cell mode" plants actually operate only 25 to 37 of their CRBs in a controlled cell mode. The remaining 112 to 148 CRBs still operate in the conventional mode. These terms are further explained in Section 3.1.1 of the text.

Older-model CRBs contain four upper stellite bearings (rollers), which were not included in the analyses. The above curie estimates exclude the CRB stellite bearings because it is assumed that the bearings are removed before waste classification of the CRBs. Stellite bearing volumes average approximately 0.0001 m^3 (0.002 to 0.003 ft^3) per CRB, depending on the removal method. Due to their high cobalt content, stellite bearings become highly activated; 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 because of their small volume. However, most plants that have stellite bearings on their CRBs are gradually replacing them with inconel bearings. These inconel bearings become activated to a much lower level than the stellite bearings. Because the inconel bearings are not highly activated, as are the stellite bearings, they are left in place in the CRB and their activity is averaged over the entire volume of the CRB. Since stellite bearing replacement should be completed by about 1996, stellite bearings are not a concern for the GTCC LLW Program, and were not included in the volume and activity estimates.

The estimated CRB volumes and curie contents conservatively assume that CRBs which exceed Class C limits 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. Plants encountering CRBs above Class C limits would employ customary practices to reduce GTCC LLW volumes. This would, as a minimum, entail removal of the velocity limiter, reducing CRB metal volume from 0.017 to 0.015 m^3 (0.6 to 0.53 ft^3). If CRBs had to be processed for shipment, those axial sections below Class C limits would also be segregated from the portion of the CRB that is above Class C limits. For those CRBs that run in the conventional mode of operation, the 0.015 m^3 (0.53 ft^3) of the cruciform could be reduced by half, to about 0.0074 m^3 (0.26 ft^3). For those CRBs that run in the controlled cell mode, the cruciform volume could be reduced by 25%, to about 0.0113 m^3 (0.40 ft^3). Nevertheless, volume estimates given in this report are conservatively based on intact CRBs with a volume of 0.017 m^3 (0.6 ft^3).

3.1.2 Local Power Range Monitor Strings

A BWR core contains 31 LPRM strings, each with 0.7-in. diameter, 0.03-in.-thick stainless steel tubing, 38 to 42 ft long, and containing four fission chamber detectors. The metal volume of an intact LPRM string is 0.0028 m^3 (0.1 ft^3), and the weight is about 50 lb. When sectioned, the 13.5-ft "hot" (high activity) end has a waste volume of 0.0005 m^3 (0.017 ft^3) and weighs 8.5 lb. The "cold" (low activity) end has a waste volume of 0.0024 m^3 (0.083 ft^3) and weighs 41 lb.

LPRM strings are routinely discharged based on fission chamber depletion or when strings malfunction. The normal operating life for an LPRM string is about three cycles. Generally, all LPRM strings exceed Class C limits except those that are removed early, after one or two cycles. Historical data indicate that 12 LPRM strings are discharged each fuel cycle. The study assumed that 75% of the LPRMs discharged would exceed Class C limits. This yields a generation rate of 0.0255 m^3 (0.9 ft^3) per BWR per fuel cycle.

To estimate typical activities, the study assumed that the LPRMs received three cycles, or 1,170 EFPD, of exposure. These LPRMs contained 930 Ci and had 10 CFR 61 Tables 1 and 2 sums of fractions of 1.6 and 1.4, respectively.

The estimated LPRM string volumes conservatively assume that LPRM strings will remain intact. Over 99% of the activity in the LPRM strings is contained in the hot end. Removal and shipment of those cold end sections that are under Class C limits could substantially reduce GTCC LLW volumes from 0.0255 m^3 (0.9 ft^3) to about 0.0042 m^3 (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 having 10 CFR 61 Tables 1 and 2 sums of fractions of 9.2 and 8.1, respectively. Although this method of GTCC LLW volume reduction is possible, volume estimates given in this report are conservatively based on intact LPRM strings with a volume of 0.0028 m^3 (0.1 ft^3) each.

3.1.3 IRM/SRM Dry Tubes

Each core contains eight IRM dry tubes and four SRM dry tubes. These dry tubes are 0.69-in. diameter, 0.012-in.-thick stainless steel tubing, and are 42 to 44 ft long. The metal volume of an intact dry tube is 0.00227 m^3 (0.08 ft^3), and the weight is about 39 lb.

Dry tubes are not routinely discharged. They are replaced only when damaged or embrittled. Historical data indicate that all 12 dry tubes are replaced at intervals of five to six fuel cycles, and all typically exceed Class C limits. An average rate of 2 units or 0.0045 m^3 (0.16 ft^3) per plant per fuel cycle was assumed.

To estimate typical activities, the study assumed that the dry tubes received six cycles, or 2340 EFPD, of exposure. These dry tubes contained 1690 Ci and had 10 CFR 61 Tables 1 and 2 sums of fractions of 4.6 and 4.0, respectively.

The estimated volumes and curie contents for dry tubes conservatively assume that the tubes will remain intact. However, over 99% of the activity in the dry tubes is contained in the hot end. When sectioned, the 13.5 ft hot end has a waste volume of 0.00028 m^3 (0.01 ft^3) and weighs 5 lb. The cold end has a waste volume of 0.002 m^3 (0.07 ft^3) and weighs 34 lb. The removal and shipment of those cold end sections that are below Class C limits could substantially reduce GTCC LLW volumes. For dry tubes, the generation rate would be reduced from 0.0045 m^3 (0.16 ft^3) to about 0.0006 m^3 (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 having 10 CFR 61 Tables 1 and 2 sums of fractions of about 37 and 32, respectively. Although this method of GTCC LLW volume reduction is possible, volume estimates given in this report are conservatively based on intact dry tubes with a volume of 0.00227 m^3 (0.08 ft^3) each.

3.2 Activated Metals from PWR Operations

As discussed in Section 2, current data indicate that typical PWRs operate on 19- to 20-month cycles, including three months allowed for refueling. This study used the 20-month cycle for calculations for Combustion Engineering plants. For Babcock & Wilcox and Westinghouse calculations, the 19-month cycle was used. The average PWR 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. The study 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. The study assumed that Westinghouse and Babcock & Wilcox plants had 38 months of decay time, and that Combustion Engineering plants had 40 months of decay time.

Component activities were estimated based on experience in characterizing actual PWR components for disposal. Characterization methods include taking radiation profile measurements and multiplying the results by scaling factors based on ratios for concentrations of critical material elements relative to cobalt concentrations. The PWR materials compositions used are based on actual data furnished by clients for disposal projects.

Table 6 presents the estimated activities of the major radionuclides for these PWR 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.

The following section presents the basis for projecting volumes and activity of activated metals exceeding Class C limits generated from routine PWR operations.

3.2.1 Thimble Plug Assemblies (Westinghouse)

These components are used to restrict water flow through fuel assemblies and are positioned in the top of selected assemblies. For Westinghouse plants, the thimble plug assemblies are stainless steel with an inconel spring and weigh 10 to 12 lb. They are discharged only when damaged. Although positioned above the active fuel region, their relatively long in-core residence times can result in some units exceeding Class C limits.

For Westinghouse plants that discharged thimble plug assemblies in the five years prior to 1991, the average discharge rate was 16 units per plant per fuel cycle. Since a small portion (about 25%) of these were above Class C limits, the generation rate is assumed to be 4 units, or 0.0028 m^3 (0.1 ft^3), per plant per fuel cycle.

Comparable components in other plants are the Babcock and Wilcox (B&W) orifice rod assemblies and Combustion Engineering control element assembly plugs. However, only a negligible fraction of these B&W and Combustion Engineering components exceed Class C limits, and should not be a concern for the GTCC LLW Program. Therefore, these B&W and Combustion Engineering components were not included in the volume and activity estimates.

To estimate activities, the study assumed that the Westinghouse units received five fuel cycles of exposure (2000 EFPD). These units contained 150 Ci and have 10 CFR Tables 1 and 2 sums of fractions of 1.0 and 1.3, respectively.

Table 6. PWR operations activated metals components that individually exceed Class C limits—volume and activity estimates.

PWR vendor	Westinghouse			CE		B&W	
Component type	Thimble plugs	Instrument strings (thimble tubes)	Primary sources	Instrument strings	Primary sources	Instrument strings (in-core detectors)	Primary sources
Volume per component m ³ (ft ³)	0.0007 (0.025)	0.0017 (0.06)	0.0002 (0.007)	0.002 (0.07)	0.0006 (0.02)	0.0028 (0.1)	0.0006 (0.02)
Weight (lb)	12	30	3.5	34	10	50	1
Mass (kg)	5.4	14	1.6	15	4.5	23	0.5
Number of cycles	5	7	5	3	5	3	5
Exposure-EFPD	2,000	2,800	2,000	1,290	2,150	1,200	2,000
10 CFR 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
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-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

3.2.2 In-Core 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. The common names of the instrument strings and their characteristics vary with each reactor vendor. This study uses the general term "instrument string" to refer to the following components, listed by their reactor-specific name.

- Thimble tubes (Westinghouse)
- In-core detectors (Babcock & Wilcox)
- Instrument strings (Combustion Engineering)

3.2.2.1 Thimble Tubes (Westinghouse). Each Westinghouse plant uses eight fixed and four movable in-core detectors, which contain fission chambers housed in a stainless steel sheath. These are inserted into about 50 thimble tubes located throughout the core. Each thimble tube has a total length of about 110 ft and a metal volume of about 0.0017 m^3 (0.06 ft^3).

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 about 75% of these exceed Class C limits, the generation rate is 1.5 units, or 0.0025 m^3 (0.09 ft^3), per plant per fuel cycle.

To estimate activities, the study assumed that the thimble tubes received seven fuel cycles, or 2800 EFPD, of exposure. These thimble tubes contained 1,060 Ci and had 10 CFR 61 Tables 1 and 2 sums of fractions of 3.3 and 4.3, respectively.

The estimated volumes and curie contents of thimble tubes that individually exceed Class C limits conservatively assume that the thimble tubes will remain intact. However, most of the activity in the thimble tubes is contained in the hot end section, which has a volume of about 0.0002 m^3 (0.007 ft^3). The removal and shipment of those cold end sections that are below Class C limits could substantially reduce GTCC LLW volumes. For thimble tubes, the generation rate would be reduced from 0.0025 m^3 (0.09 ft^3) to about 0.0003 m^3 (0.011 ft^3) 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 having a 10 CFR 61 Table 1 sum of fractions of about 37. Although this method of reducing GTCC LLW volumes is possible, this report's conservative volume estimates, as well as classification status, are based on intact thimble tubes with a volume of 0.0017 m^3 (0.06 ft^3) each.

3.2.2.2 In-Core Detectors (Babcock & Wilcox). Babcock & Wilcox plants have locations to accommodate 52 in-core 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 a metal volume of about 0.0028 m^3 (0.1 ft^3).

For plants that have disposed of in-core detectors, the average rate has been 16 detectors per plant per fuel cycle. Since about 75% exceed Class C limits, the generation rate is 12 detectors, or 0.034 m^3 (1.2 ft^3), per plant per fuel cycle.

To estimate activities, the study assumed that the detectors received three fuel cycles, or 1200 EFPD, of exposure. These detectors contained 360 Ci and had 10 CFR 61 Tables 1 and 2 sums of fractions of 0.8 and 2.1, respectively.

The estimated volumes and curie contents of in-core detectors that individually exceed Class C limits conservatively assume that the in-core detectors will remain intact. However, over 99% of the activity is contained in the 12- to 14-ft hot end section, with a metal volume of about 0.0004 m^3 (0.013 ft^3). The removal and shipment of those cold end sections that are below Class C limits would substantially reduce GTCC LLW generation rates. For these in-core detectors, the generation rate could be reduced from 0.034 m^3 (1.2 ft^3) to about 0.0044 m^3 (0.16 ft^3) 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 having a 10 CFR 61 Table 2 sum of fractions of about 16. Although this method of reducing GTCC LLW volumes is possible, this report's conservative volume estimates, as well as classification status, are based on intact in-core detectors with a volume of 0.0028 m^3 (0.1 ft^3) each.

3.2.2.3 Instrument Strings (Combustion Engineering). Combustion Engineering plants use between 28 and 61 in-core 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 yield the higher 10 CFR 61 characterization due to high nickel content. Both types have a total length of 35 to 116 ft and weigh between 7 and 34 lb. The typical volume is 0.002 m^3 (0.07 ft^3).

For the five years prior to 1991, these instrument strings were changed out at an average rate of 24 units per plant per fuel cycle. Since about 75% of these exceed Class C limits, the generation rate is assumed to be 18 instrument strings, or 0.036 m^3 (1.26 ft^3), per plant per fuel cycle.

To estimate activities, the study assumed that the instrument strings received three fuel cycles, or 1290 EFPD, of exposure. These instrument strings contained 385 Ci and had 10 CFR 61 Tables 1 and 2 sums of fractions of 3.3 and 4.1, respectively.

The estimated volumes and curie contents of instrument strings that individually exceed Class C limits conservatively assume that the instrument strings will remain intact. However, over 99% of the activity is contained in the 12- to 14-ft hot end, with a metal volume of about 0.0002 m^3 (0.008 ft^3). The removal and shipment of those cold end sections that are below Class C limits could substantially reduce GTCC LLW generation rates. For these instrument strings, the generation rate would be reduced from 0.036 m^3 (1.26 ft^3) to about 0.0041 m^3 (0.14 ft^3) 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 having a 10 CFR 61 Table 2 sum of fractions of about 36. Although this method of volume reduction of GTCC LLW is possible, this report's conservative volume estimates, as well as classification status, are based on intact instrument strings with a volume of 0.002 m^3 (0.07 ft^3) each.

3.2.3 Primary Sources

Primary sources greatly exceed Class C limits. Usually they are used only in the first fuel cycle to start a reactor, then removed. Only a few BWR primary sources exist, which are used to start a reactor and then stored until needed to start another reactor; these sources are not considered waste. At PWRs, however, current practice is to store primary sources as waste in the spent fuel pool after they have been used to start a reactor. This current waste inventory of PWR primary sources is considered by the GTCC LLW Program to be the only significant 1993 inventory of GTCC LLW from nuclear utilities. When DOE provides for disposal of GTCC LLW, the primary sources that have accumulated will be shipped with any other GTCC LLW.

No additional primary sources, nor any other GTCC sources, are normally generated during the life of a reactor. Continued reactor operation uses only secondary sources, which are not GTCC. Therefore, this study assumes no ongoing generation of GTCC LLW from primary sources.

PWR primary source strength and volume vary by reactor vendor:

Westinghouse plants use two primary source rods in assemblies that also contain burnable poison rods. The primary source assemblies are thimble plug assemblies with one primary source rod, 1 to 4 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 themselves only weigh 2.7 to 3.6 lb, and are 110 to 152 in. long. The source materials are californium (Cf-252), plutonium-beryllium (Pu-Be), or polonium-beryllium (Po-Be). The Cf and Po-Be sources are not a GTCC LLW concern because they do not contain radionuclides regulated by 10 CFR 61. However, the Pu-Be sources are of concern because of the plutonium. The study assumed that the source rods are separated from the assembly and that the two rods have a waste volume of 0.0002 m^3 (0.007 ft^3). Typical activities are 50 Ci for the Pu-Be sources.

Babcock & Wilcox plants use two primary neutron sources that are stainless steel-clad americium-beryllium. Each primary source is 146 in. long and 0.4 in. in diameter, with a total weight of about 1 lb. The waste volume of each unit is 0.0006 m^3 (0.02 ft^3). Typical activities are 25 Ci.

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. Each source assembly contains one secondary antimony-beryllium and two primary plutonium-beryllium sources. All the sources remain in the assembly for this waste form. The waste volume is 0.0006 m^3 (0.02 ft^3) per unit. Typical activities are 340 Ci.

3.3 Activated Metals Summary

Table 7 summarizes the generation rates for BWR and PWR activated metals components that individually exceed Class C limits during routine operations. Two sets of values are presented: a maximum set that addresses intact components, and a minimum set which 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. The maximum set was used to project waste quantities.

Table 7. Summary of operations activated metals components that individually exceed Class C limits—volumes.

	Number of plants	Unit volume m ³ (ft ³)	Units per cycle	Percent above Class C	Units above Class C, per cycle	Volume above Class C, per cycle m ³ (ft ³)
Maximum estimates—assume intact components						
1. BWR operations	37					
a. Control rod blades (CRBs)						
- Midsize plant conventional CRBs	14	0.017 (0.6)	14	10	1.4	0.024 (0.84)
- Midsize plant controlled cell CRBs	15	0.017 (0.6)	14/11 ^a	25/10 ^a	4.4	0.074 (2.6)
- Large plant conventional CRBs	5	0.017 (0.6)	19	10	1.9	0.032 (1.1)
- Large plant controlled cell CRBs	3	0.017 (0.6)	19/15 ^a	25/10 ^a	6.2	0.106 (3.8)
b. LPRM strings	37	0.0028 (0.1)	12	75	9	0.0255 (0.9)
c. IRM/SRM dry tubes	37	0.0023 (0.08)	2	100	2	0.0045 (0.16)
2. Westinghouse PWR operations	51					
a. Thimble plug assemblies		0.0007 (0.024)	16	25	4	0.0028 (0.1)
b. Thimble tubes		0.0017 (0.06)	2	75	1.5	0.0025 (0.09)
c. Primary source rods ^b		0.0002 (0.007)	NA	100	NA	0.0002 (0.007)
3. CE PWR operations	15					
a. Instrument strings		0.002 (0.07)	24	75	18	0.036 (1.26)
b. Primary sources ^b		0.0006 (0.02)	NA	100	NA	0.0006 (0.02)
4. B&W PWR operations	7					
a. In-core detectors		0.0028 (0.1)	16	75	12	0.034 (1.2)
b. Primary sources ^b		0.001 (0.02)	NA	100	NA	0.001 (0.02)
Minimum estimates—assume only hot portions of components						
1. BWR operations	37					
a. Control rod blades						
- Midsize plant conventional CRBs	14	0.0074 (0.26)	14	10	1.4	0.0104 (0.36)
- Midsize plant controlled cell CRBs	15	0.0105 (0.37)	14/11 ^a	25/10 ^a	4.4	0.0462 (1.68)
- Large plant conventional CRBs	5	0.0074 (0.26)	19	10	1.9	0.0141 (0.49)
- Large plant controlled cell CRBs	3	0.0105 (0.37)	19/15 ^a	25/10 ^a	6.2	0.0651 (2.29)
b. LPRM strings	37	0.0005 (0.017)	12	75	9	0.0043 (0.153)
c. IRM/SRM dry tubes	37	0.0003 (0.012)	2	100	2	0.0007 (0.024)
2. Westinghouse PWR operations	51					
a. Thimble plug assemblies		0.0007 (0.024)	16	25	4	0.0028 (0.1)
b. Thimble tubes		0.0002 (0.007)	2	75	1.5	0.0003 (0.011)
c. Primary source rods ^b		0.0002 (0.007)	NA	100	NA	0.0002 (0.007)
3. CE PWR operations	15					
a. Instrument strings		0.0002 (0.008)	24	75	18	0.004 (0.14)
b. Primary sources ^b		0.0006 (0.02)	NA	100	NA	0.0006 (0.02)
4. B&W PWR operations	7					
a. In-core detectors		0.0004 (0.013)	16	75	12	0.0045 (0.16)
b. Primary sources ^b		0.001 (0.02)	NA	100	NA	0.001 (0.02)
<p>a. These two numbers represent the controlled cell CRBs and the conventional mode CRBs that operate together within the same plant.</p> <p>b. Although primary source data are included under the column headings for unit volume and units per cycle, that terminology is not applicable to primary sources. Since primary sources are used only once at the startup of a reactor, the reported volumes are per plant, not per cycle. However, the volumes reported are per unit.</p>						

4. GENERATION AND ACTIVITY OF PROCESS WASTE

Process wastes are generated from the cleanup of liquids containing soluble and insoluble radioactive constituents. Waste forms that can individually exceed Class C limits consist of cartridge filters and the cation bead resin produced during full RCS decontamination, performed with the fuel remaining in the reactor. 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 radiation levels and radionuclide content and distribution for the waste type. Decontamination of a full RCS has not yet been performed, and actual data for decontamination resins under such circumstances were not available. Instead, actual data for smaller decontamination projects at both BWRs and PWRs were used in conjunction with several analytical studies related to full RCS decontaminations.

4.1 Cartridge Filters

Reactors use cartridge filters throughout the primary and secondary systems to clean up contaminated liquids. The filters represent relatively small waste volumes, with each filter typically less than 0.014 m^3 (0.5 ft^3). Due to their relatively small volumes and high activities, these filters are usually Class C. Individual filters can exceed Class C limits.

In certain applications, when cartridge filters are changed out based on pressure differential, filter radiation levels can range from 50 to 200 rem/hr. At these radiation levels, some filters exceed Class C limits. Changes in plant practices to remove filters from service at preset radiation levels would essentially prevent all filters from exceeding Class C limits, except BWR control rod drive (CRD) strainers.

Table 8 illustrates dimensions of cartridge filters that can individually exceed Class C limits. These filters are discussed below.

4.1.1 BWR Operations

LLW exceeding Class C limits that is routinely generated by BWR operations includes strainers for control rod drives and cartridge filters from fuel pool and underwater vacuum operations.

- **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. One strainer has a volume of 0.0017 m^3 (0.06 ft^3), the other a volume of 0.002 m^3 (0.006 ft^3); total volume is about 0.0019 m^3 (0.066 ft^3). About 50% of the strainers generated from this source are individually above Class C limits.

Table 8. Cartridge filters that can individually exceed Class C limits—volumes and weights.

		Diameter (in.)	Length (in.)	Waste volume		Weight (lb)	Mass (kg)
				m ³	(ft ³)		
BWR filters							
a.	Control rod drive (inner)	2.1	3	0.0002	(0.006)	0.5	0.23
b.	Control rod drive (outer)	3.75	9	0.0017	(0.06)	2.5	1.1
c.	Spent fuel pool/vacuum	6	30	0.014	(0.5)	10	4.5
PWR filters							
a.	Reactor coolant system	6.25	17.5	0.009	(0.3)	9	4.1
b.	Seal water injection	2.75	20	0.0028	(0.1)	2.5	1.1
c.	Reactor cavity drain	2	30	0.0028	(0.1)	4	1.8
d.	Spent fuel pool	6	30	0.014	(0.5)	10	4.5
e.	Crud tank (B&W only)	10	17	0.022	(0.8)	40	18

Control rod drive strainers are replaced when control rod drives are refurbished or replaced. Drives are replaced infrequently, but about 15% of the control rod drives in a plant are refurbished during each refueling outage. For projection purposes, the study assumed all control rod drives would be replaced or refurbished over 6 fuel cycles (10 years). This corresponds to a per-cycle generation rate of about 15 pairs of CRD strainers that individually exceed Class C limits, with a total volume of about 0.028 m³ (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 be 150 rem/hr on contact, contain several curies, and have a 10 CFR 61 Table 1 sum of fractions of about 1.4.

- **Fuel Pool/Vacuum Filters**—The fuel pool filters are located upstream or downstream of fuel pool purification demineralizers and consist of cartridge-type filter elements with metal retention screens. The vacuum filter units consist of large and small micron

cartridge filters with metal retention screens, and are used during vacuuming of the spent fuel pool and the reactor cavity. The typical fuel pool or underwater vacuum filter is 6 in. in diameter and 30 to 36 in. long. This corresponds to a waste volume of about 0.014 m³ (0.5 ft³).

These filters are usually generated during refueling outages. During a typical outage, a dozen filters can be generated. About 25% of these, or three units (0.042 m³ or 1.5 ft³) per plant per fuel cycle, can individually exceed Class C limits.

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 61 classification was used to estimate activities for this filter type. Typically, these filters read over 100 rem/hr on contact, contain about 4 Ci, and have a 10 CFR 61 Table 1 sum of fractions of about 1.5.

4.1.2 PWR Operations

Most of the PWR cartridge filters that can individually exceed Class C limits are common to all PWR reactor types. Generation rates depend on filter type and plant practices. The rates considered for this study are typical.

- **PWR Miscellaneous Cartridge Filters**—Most PWR cartridge filters are grouped in the DOE/LLW-114 Revision 1 report discussions under a general category called "PWR Miscellaneous Cartridge Filters." This category includes the following filter types, which are discussed below:

- Reactor coolant system filters
- Seal water injection filters
- Cavity drain filters
- Fuel pool filters.

Reactor coolant system filters are among the highest-activity filters generated by a PWR. They consist of small to medium micron-size cartridge filters with metal retention screens that purify water returning to the volume control or makeup tank. Filter sizes range from the Pall^c type, which is 6 in. in diameter and 20 in. long, to the Cuno^a array of filters, which is 14 in. in diameter and 40 in. long. In the array type, a typical filter element is 6.25 in. in diameter and 17.5 in. long. A typical volume for a single filter is about 0.009 m³ (0.3 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, may individually exceed Class C limits. This corresponds to a generation rate of 0.018 m³ (0.6 ft³) per plant per fuel cycle.

c. Pall and Cuno are filter manufacturers. Mention of specific products and/or manufacturers in this document implies neither endorsement or preference, nor disapproval by the U.S. Government, any of its agencies, or EG&G Idaho, Inc., of the use of a specific product for any purpose.

Data from a single PWR 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 has a 10 CFR 61 Table 1 sum of fractions of about 4.3.

Seal water injection filters are small micron metal or fiber retention elements used to purify water from the volume control tank. Filter sizes range from 2 in. in diameter and 27 in. long to 3 in. in diameter and 22 in. long. A typical volume for a single filter is about 0.0028 m³ (0.1 ft³). These filters are usually generated at a rate of about 16 units per fuel cycle. About 25% of these, or four units per fuel cycle, may individually exceed Class C limits. This corresponds to a generation rate of 0.0113 m³ (0.4 ft³) per plant per fuel cycle.

Data from a single PWR 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 has a 10 CFR 61 Table 1 sum of fractions of about 2.9.

Cavity drain filters, also called seal water return filters, are usually relatively low-activity filters, except in those infrequent cases when the PWR reactor cavity is decontaminated. They consist of medium micron metal or fiber retention elements. The typical filter is 2 in. in diameter and 30 in. long. This corresponds to a volume of about 0.0028 m³ (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, may individually exceed Class C limits. This corresponds to a generation rate of 0.0014 m³ (0.05 ft³) per plant per fuel cycle.

Data from a single PWR 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 has a 10 CFR 61 Table 1 sum of fractions of about 2.3.

Fuel pool filters are also relatively high-activity filters generated by a PWR. They have cartridge-type filter elements with metal retention screens. The typical filter is 6 in. in diameter and 30 in. long. This corresponds to a volume of about 0.014 m³ (0.5 ft³). These filters are located upstream or downstream of fuel pool purification demineralizers, and are usually generated during PWR refueling outages, when the spent fuel pool and reactor cavity are vacuumed. During a typical outage, four such filters can be generated. About 25% of these, or 1 unit per fuel cycle, may individually exceed Class C limits. This corresponds to a generation rate of 0.014 m³ (0.5 ft³) per plant per fuel cycle.

Data from several PWR plants showed filter radiation levels in the 100 rem/hr range. At these radiation levels, a typical filter contains about 3 Ci and has a 10 CFR 61 Table 1 sum of fractions of about 1.6.

- **Crud Tank Filter**—These filters are found only in B&W plants, and thus are listed separately in the DOE/LLW-114 Revision 1 report. They are the highest-activity filters in all PWRs. These filters collect the media backflushed from etched disk filters. The typical filter is 10 in. in diameter and 17 in. long. This corresponds to a volume of

about 0.022 m³ (0.8 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, the generation rate is 0.011 m³ (0.4 ft³) per plant per fuel cycle.

Actual activity data for these filters were not readily available. The activity estimated here is the same as for the PWR fuel pool filter because, of the PWR filters listed above, the fuel pool filter has the highest amount of high activity or long-lived radionuclides.

4.1.3 Radionuclide Content and Distribution

Cartridge filters represent very low-volume sources of LLW and exceed Class C limits because the activity is distributed over a relatively small volume. Table 9 presents a summary of the cartridge filters discussed above. As shown, radiation levels are relatively high and, in all cases, exceed Class C limits due to transuranic concentrations.

Table 9. Cartridge filters that individually exceed Class C limits—volume and activity estimates.

Reactor type	BWR		PWR				
	Control rod drive	Fuel pool vacuum	Reactor coolant	Seal water injection	Cavity drain	Fuel pool	Crud tank ^a
Volume m ³ (ft ³)	0.002 (0.07)	0.014 (0.5)	0.009 (0.3)	0.0028 (0.1)	0.0028 (0.1)	0.014 (0.5)	0.022 (0.8)
Weight (lb)	3	10	9	2.5	4.0	10	40
Mass (kg)	1.4	4.5	4.1	1.1	1.8	4.5	18
10 CFR 61 status							
- Table 1 fraction	1.35	1.52	4.27	2.90	2.25	1.63	1.63
- Table 2 fraction	0.02	0.03	0.03	0.03	0.03	0.01	0.01
R/hr on contact	150	100	200	50	50	100	100
Activity (curies)							
H-3	1.34E-04	2.55E-03	1.44E-03	3.41E-04	8.55E-04	3.24E-03	3.24E-03
C-14	3.43E-07	1.10E-03	1.06E-02	2.51E-03	5.13E-03	1.94E-02	1.94E-02
Mn-54	3.10E-02	1.35E-01	7.56E-02	1.80E-02	8.55E-03	3.24E-02	3.24E-02
Fe-55	3.03E+00	2.29E+00	1.51E+00	3.58E-01	2.37E-01	8.96E-01	8.96E-01
Co-58	—	—	9.80E-01	2.17E-01	2.85E-01	1.08E+00	1.08E+00
Co-60	5.08E-01	1.14E+00	9.12E-01	2.33E-01	2.74E-01	1.04E+00	1.04E+00
Ni-63	2.82E-02	1.16E-01	2.33E-01	5.53E-02	5.79E-02	2.19E-01	2.19E-01
Zn-65	—	2.85E-01	—	—	2.85E-03	1.08E-02	1.08E-02
Sr-90	1.30E-05	1.22E-03	8.06E-04	1.91E-04	9.50E-05	3.60E-04	3.60E-04
Cs-137	5.26E-04	1.80E-02	2.00E-03	4.75E-04	1.90E-04	7.20E-04	7.20E-04
Pu-241	1.70E-03	2.05E-03	2.13E-02	5.06E-03	4.42E-03	1.67E-02	1.67E-02
TRU	1.35E-04	3.48E-04	1.08E-03	5.61E-04	5.31E-04	9.06E-04	9.06E-04
Total curies	3.60E+00	3.99E+00	3.75E+00	8.90E-01	8.77E-01	3.32E+00	3.32E+00

a. Actual activity data for these filters were not readily available. The activity estimated here is the same as for the PWR fuel pool filter since that filter has the highest amount of high activity or long-lived radionuclides of the PWR filters.

4.2 Full RCS Decontamination Resins

Another potential source of GTCC LLW is resin from reactor coolant system (RCS) decontamination processes during routine plant operations; resin beds are used to clean up liquids containing soluble and insoluble radioactive constituents. Decontamination processing can be done with or without fuel assemblies in the system. Decontamination resins generally will not exceed Class C limits if the decontamination is performed with the fuel out. However, following the return to operation after fuel-out decontamination, the system recontaminates quickly due to the large inventory of radionuclides contained in the metal corrosion film on in-core fuel surfaces. Full system fuel-in decontaminations are being contemplated to remove the radionuclides from the fuel surfaces, thereby reducing the recontamination rate. Fuel-in decontamination resins may exceed Class C limits, primarily due to the transuranic content removed from fuel assemblies. Therefore, removing the fuel assemblies before decontamination will essentially eliminate the generation of decontamination resins above Class C limits. On the other hand, decontamination operations performed with fuel assemblies in the vessel could lead to resins above Class C limits at both BWRs and PWRs.

Full RCS decontamination is an untried process; it has never been performed and will not be until the mid-1990s. While full RCS decontamination projects 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. This study estimates that this will not happen until about 2013 (20 years in the future).

4.2.1 Resin Generation Processes

Several different decontamination processes are used to decontaminate light water reactors during routine operations. BWRs have employed two decontamination processes, referred to as the LOMI and CANDECON processes.^d Although the primary systems differ, the same two decontamination processes have also been employed at PWRs. A third process, referred to as CANDEREM,^a is also being evaluated for full RCS decontamination at PWRs. All these processes deposit the crud and scale removed from reactor internal surfaces onto 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.

d. LOMI = Low Oxidation Metal Ion, a decontamination process developed in England and licensed by the Electric Power Research Institute (EPRI)
CANDECON = A proprietary decontamination process developed by the Canadians
CANDEREM = Another proprietary decontamination process that currently is not as well known or used as the above processes

For further information concerning these decontamination processes, see Reference 10.

The 1991 DOE/LLW-114 report speculated that the weight% of chelating agents in the cation resins might cause them to be classified as mixed waste due to hazardous waste classification under the Resource Conservation and Recovery Act. However, after evaluating the possibility, Appendix E-2 indicates a very low probability of these resins being mixed waste.

Several studies have been performed to evaluate the curie loading of resin beds used for full RCS decontamination projects.

- A 1991 study by Vance and Associates^e concluded that some of the resins generated during full RCS decontamination could exceed Class C limits, but 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. Those projects involved relatively low curie loadings on resin beds, and there is some uncertainty as to whether the data were for the cation beds or for mixed beds. (Since most of the activity is collected by the cation resin beds, data for these beds are the best indicator of classification status.) That study also considered theoretical full RCS decontaminations of BWRs, indicating that LLW resins above Class C limits could be generated with fuel assemblies remaining in the vessel. The estimated total resin volumes ranged from 84 m³ (2,970 ft³) for a large BWR LOMI to 67 m³ (2,370 ft³) for a large BWR CANDECON. The maximum realistic case estimated that only 2.5% to 6% of these would be above Class C limits. Volumes for full RCS decontamination resins at PWRs were not estimated.
- Actual data available from two similar projects on cation resin beds, one BWR and one PWR, were also reviewed (Reference 1). Those data confirmed that scaling factors for the transuranics, which determine classification status, are indeed within the bounds of the sample data reported in the 1991 study by Vance and Associates. The resin curie loadings, which directly affect concentration and classification, were also relatively low for these data sets.
- A 1989 calculational study by Sejvar et al.,¹⁰ on PWR full RCS decontamination projects with fuel assemblies out, indicated relatively large curie loadings on the cation resin beds. These estimated curie loadings on the cation resin beds for 3 and 4 loop PWRs are shown on Table 10 along with the cation resin volumes. The activity concentration listed for the CANDEREM 0/1 process was used as the basis for this study's PWR cation resin activity concentration.

In estimating the radionuclide distribution of activity, the Sejvar study used transuranic scaling factors at least one order of magnitude lower than those later identified in the 1991 fuel-in decontamination resin study by Vance and Associates. When the cation bed resin activity concentrations estimated for the Sejvar calculational study are adjusted to reflect the fuel-in transuranic scaling factors found in actual data, the cation resins could exceed Class C limits.

e. Vance and Associates, *GTCC Waste Volumes of Ion Exchange Resins and Filters Projected to be Generated by the Existing U.S. Commercial Nuclear Power Plants*, draft, Vance and Associates, Ruidoso, New Mexico, October 1991.

Table 10. Full PWR RCS decontamination resin LLW volume estimates (for comparison purposes only).

	Total curies	3 loop resin		4 loop resin		3 loop ($\mu\text{Ci}/\text{cm}^3$)	4 loop ($\mu\text{Ci}/\text{cm}^3$)
		m^3	(ft^3)	m^3	(ft^3)		
CANDEREM process							
DEREM 0/1 cation	5,032	8.50	(300)	8.50	(300)	592	592
DEREM 6/1 anion	335	15.46	(546)	20.82	(735)	22	16
AP 0/1 cation	470	28.04	(990)	28.04	(990)	17	17
AP 3/11 cation	<u>872</u>	<u>19.60</u>	<u>(692)</u>	<u>35.51</u>	<u>(1,254)</u>	<u>45</u>	<u>25</u>
Total	6,709	71.60	(2,528)	92.86	(3,279)	676	650
LOMI process							
LOMI 0/1 cation	5,032	21.30	(752)	25.94	(916)	236	194
LOMI 1/0 anion	335	17.45	(616)	23.56	(832)	19	14
AP 0/1 cation	470	28.04	(990)	28.04	(990)	17	17
AP 3/11 cation	<u>872</u>	<u>19.60</u>	<u>(692)</u>	<u>35.51</u>	<u>(1,254)</u>	<u>45</u>	<u>25</u>
Total	6,709	86.38	(3,050)	113.05	(3,992)	317	250

Source: See *Full RCS Decontamination Program: Waste Management Methodology and Waste Characterization* (Reference 10) for a more complete explanation of the nomenclature used in this table.

4.2.2 Projections for Decontamination Resins

The above studies bound the problem, with high activity predicted by the Sejvar study, and high transuranic scaling factors taken from actual data.

To project full RCS fuel-in decontamination resin volumes, this study assumed that 50% of the cation resins generated from a typical project would exceed Class C limits and that one such decontamination project would be performed at each plant during its operating life. The resin wastes are thus reported as decommissioning waste quantities in the DOE/LLW-114 Revision 1 projections. It was also assumed that reactors with less than 20 years left in their 40-year life cycles (i.e., shutdown during or prior to year 2013) would not perform full RCS decontaminations with fuel assemblies in.

A full RCS BWR decontamination process cleans a surface area of about 11,000 m^2 plus the fuel assemblies, which represent an additional 12,000 m^2 , for a total area of about 23,000 m^2 . It can generate 1,000 to 1,200 ft^3 of resins depending on the decontamination process employed. Cation resin volumes could range from 200 to 275 ft^3 . About 50% of these, or about 2.83 m^3 (100 ft^3), exceed Class C limits due to the presence of the fuel assemblies, a source of transuranic radionuclides.

A full RCS PWR decontamination process cleans a surface area of about 200 m^2 , plus the fuel assemblies, which represent an additional 50 m^2 , for a total area of about 250 m^2 . It can generate 1,500 to 4,000 ft^3 of resins, depending upon the decontamination process employed. Most of the activity is on 300 ft^3 of cation resins. About 50% of these, or about 4.25 m^3

(150 ft³), exceed Class C limits due to the presence of the fuel assemblies, a source of transuranic radionuclides.

Activities were also estimated. For PWRs, the calculational estimates in the PWR full RCS decontamination study (Reference 10) were revised to increase transuranic scaling factors by a factor of 10 (i.e., to be in the same range as the actual data). The higher activities listed for PWR decontamination with the CANDEREM method were conservatively used for this study's projections. For BWRs, the plant-specific cation resin bed data evaluated for this project were used with an increased estimate for the curie loading on the cation resin bed. These estimates are shown in Table 11.

Table 11 also shows the transuranic scaling factors for the estimated radionuclide distributions used for calculations. As explained above, these scaling factors are within the boundaries of those reported from actual data on smaller decontamination projects in the 1991 decontamination study by Vance and Associates.

4.3 Process Waste Summary

Table 12 summarizes the generation rates for both BWR and PWR process wastes that exceed Class C limits. Two sets of values are presented: a maximum set that addresses all cartridge filters; and a minimum set which assumes that PWR plants will manage all filters to preclude concentrations exceeding Class C limits, and that BWR plants will similarly manage fuel pool/vacuum filters. Full RCS decontamination resin estimates are the same for both cases, and are based on estimates for fuel-in full RCS decontamination.

Table 11. Fuel in/full RCS decontamination resin exceeding Class C limits.

Plant type	PWR RCS fuel-in	PWR RCS fuel-in	BWR RCS fuel-in
Decon method	CANDEREM	LOMI	LOMI
Cation resin above Class C limits m ³ (ft ³)	4.25 (150)	4.25 (150)	2.83 (100)
Resin weight (lb)	7,500	7,500	5,000
Mass (kg)	3,400	3,400	2,300
10 CFR 61 status			
- Table 1 fraction	4.68	1.56	5.84
- Table 2 fraction	Negligible	Negligible	Negligible
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	<u>1.50E+00</u>	<u>5.00E-01</u>	<u>4.90E-01</u>
Total curies	2.52E+03	8.39E+02	3.00E+02
Activity $\mu\text{Ci}/\text{cm}^3$	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

Table 12. Summary of process LLW exceeding Class C limits—volumes.

		Number of plants	Unit volume m ³ (ft ³)	LLW units per cycle	Percent above Class C	Units above Class C, per cycle	Volume above Class C, per cycle m ³ (ft ³)	
Maximum estimates—change filters due to pressure loading								
1.	BWR cartridge filters	37						
a.	Control rod drives		0.002 (0.07)	30	50	15	0.03 (1.05)	
b.	Fuel pool/vacuums		0.014 (0.5)	12	25	3	0.042 (1.5)	
2.	PWR cartridge filters							
a.	PWR misc. cartridge filters	73						
-	Reactor coolant systems		0.009 (0.3)	8	25	2	0.018 (0.6)	
-	Seal water injections		0.0028 (0.1)	16	25	4	0.011 (0.4)	
-	Cavity drains		0.0028 (0.1)	2	25	0.5	0.0014 (0.05)	
-	Spent fuel pools		0.014 (0.5)	4	25	1	0.014 (0.5)	
b.	B&W crud tank filters	7	0.022 (0.8)	0.5	100	0.5	0.011 (0.4)	
3.	BWR RCS decon resins ^a		5.66 (200)	NA	50	NA	2.83 (100)	
4.	PWR RCS decon resins ^a		8.50 (300)	NA	50	NA	4.25 (150)	

Minimum estimates—change filters due to radiation level, keep $\leq$ Class C

1.	BWR CRD filters	37	0.002 (0.07)	30	50	15	0.03 (1.05)	
2.	BWR RCS decon resins ^a		5.66 (200)	NA	50	NA	2.83 (100)	
3.	PWR RCS decon resins ^a		8.50 (300)	NA	50	NA	4.25 (150)	

a. Although resin data are included under the column headings for unit volume and units per cycle, that terminology is not applicable to resin waste. First, resins are not quantified in separate units, as are the filters; the amounts listed reflect the total amount of resin produced during reactor decontamination operations. Second, this report assumes that a reactor undergoes full RCS fuel-in decontamination only once during its operating life; the volumes listed are total for the life of the plant, not per cycle. For this reason, the main report (DOE/LLW-114 Revision 1) includes RCS decontamination resins with decommissioning waste.

5. GENERATION AND ACTIVITY OF DECOMMISSIONING ACTIVATED METALS

The largest potential source of GTCC LLW in terms of volume and activity is activated metals within a 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.

5.1 LWR Components

These activated metal components consist of some 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 61 radionuclide concentrations due to activation are directly proportional to the integrated neutron flux received by the component during a plant's operating lifetime. Table 13 summarizes the physical characteristics of the components considered for this study. The PWR thermal shields do not exceed Class C limits but are listed due to their large mass and potential source of blending material for concentration averaging of other components that do exceed Class C limits (see Appendix E-5). The other components shown were evaluated to determine whether they will exceed Class C limits.

5.1.1 BWR Decommissioning Internals

The BWR components evaluated in this study were the top fuel guide, the core shroud, the fuel support castings, and the core support plate. These are described below.

Top Fuel Guide - The top fuel guide is a stainless steel structure that provides lateral support for the top section of the fuel assemblies (see Figure 2). In a BWR-4 it is composed of type 304 stainless steel beams joined at right angles, forming 16-in.-high square openings for the fuel assemblies. The beams are fastened to a peripheral rim that rests on the core shroud. In a BWR-6, the structure consists of a 5-in.-thick lower plate with machined holes for fuel assemblies plus a 39-in.-high cylindrical structure surrounding the plate periphery. The 5-in.-thick lower plate is the component that can exceed Class C limits in the BWR-6. In both plants, this component is also referred to as the upper core grid.

Core Shroud - The shroud is a large, cylindrical, type 304 stainless steel structure that surrounds the core (see Figure 2). It 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.

Fuel Support Castings - These type 304 stainless steel structures provide support for the lower section of the fuel assemblies; each unit supports four assemblies. A BWR-4 includes 140 of these 62-lb fuel support castings, while a BWR-6 includes 183 castings. These components are also referred to as orificed fuel support pieces.

Table 13. Reactor decommissioning components evaluated in this study.

Reference plant name		Weight (lb)	Volume (ft ³) ^a	Weight (kg) ^b	Volume (m ³) ^c
A.	BWR-4	5000	10.0	2273	0.28
	1. Top fuel guide	50400	100.8	22909	2.85
	2. Core shroud	8800	17.6	4000	0.50
	3. Fuel support castings	7800	15.6	3545	0.44
	4. Core support plate				
B.	BWR-6				
	1. Top fuel guide (plate) ^d	8200	16.4	3727	0.46
	2. Core shroud	8700	174.0	39545	4.92
	3. Fuel support castings	11350	22.7	5159	0.64
	4. Core support plate	9400	18.8	4273	0.53
C.	Westinghouse intermediate				
	1. Upper core support plate	6120	12.2	2782	0.35
	2. Core baffle	22300	44.6	10136	1.26
	3. Thermal shield	54500	109.0	24773	3.08
	4. Lower core support barrel	61700	123.4	28045	3.49
	5. Lower core support plate	6700	13.4	3045	0.38
D.	Westinghouse large				
	1. Upper core support plate	9200	18.4	4182	0.52
	2. Core baffle	35200	70.4	16000	1.99
	3. Thermal shield	22220	44.4	10100	1.26
	4. Lower core support barrel	61700	123.4	28045	3.49
	5. Lower core support plate	5900	11.8	2682	0.33
E.	Combustion Engineering				
	1. Fuel assembly alignment plate	10100	20.2	4591	0.57
	2. Core baffle	43900	87.8	19955	2.48
	3. Core barrel	143200	286.4	65091	8.11
	4. Lower forging (top plate) ^d	5800	11.6	2636	0.33
F.	Babcock & Wilcox				
	1. Upper grid assembly	7500	15.0	3409	0.42
	2. Core baffle	25600	51.2	11636	1.45
	3. Core support assembly barrel	44400	88.8	20182	2.51
	4. Thermal shield	47600	95.2	21636	2.69
	5. Lower grid	7200	14.4	3273	0.41

Notes:a. Weight (lb) divided by 500 lb/ft³

b. Weight (lb) divided by 2.2 lb/kg

c. Volume (ft³) × 0.0283 m³/ft³

d. Where a component name is followed by parentheses, the value shown is for that portion of the component found by the analysis to exceed Class C limits and is not the value for the entire named component.

Core Support Plate - This type 304 stainless steel cylindrical plate contains the fuel support castings (see Figure 2).

As shown previously in Table 1, 37 operating BWRs were considered. Ten of these plants were General Electric reactor models 1, 2, or 3; 19 were reactor model 4; four were referred to as model 5; and four were reactor model 6, General Electric's largest plants. Generally, the size of the decommissioning LLW components increases with the reactor model number. This evaluation considered two of the larger reactor models, a BWR-4 and a BWR-6. Waste from the model 5 reactors was conservatively estimated to be the same as that from model 6. Waste from each of the smaller (model 1, 2, or 3) plants was conservatively estimated to be the same as from the larger model 4 plants.

5.1.2 PWR Decommissioning Internals

The PWR components evaluated in this study were the upper core support plate, the core shroud (baffle), the lower core barrel, and the lower core support plate. These components are described below, and are schematically represented in Figure 3.

Upper Core Support Plate - The upper core support plate is a type 304 stainless steel cylindrical plate above the fuel that rests on the core barrel and supports fuel assemblies laterally. In Combustion Engineering plants this component is referred to as the fuel assembly alignment plate, and in Babcock & Wilcox plants it is referred to as the upper grid assembly.

Core Shroud (Baffle) - The baffle is a type 304 stainless steel structure that surrounds the core. In Westinghouse plants, structures called core formers maintain the baffle's form and location and provide for a limited water flow between the baffle and the core barrel for additional cooling of the barrel.

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. For Combustion Engineering plants, the lower core barrel is much larger than that in Westinghouse plants. In Babcock & Wilcox plants, this component is referred to as the core support assembly barrel.

Lower Core Support Plate - The lower core support plate is a type 304 stainless steel cylindrical plate below the fuel that attaches to the core barrel and supports fuel assemblies laterally. In Combustion Engineering plants this component is incorporated in the larger lower forging, but only the top cylindrical plate exceeds Class C limits. In Babcock & Wilcox plants this component is referred to as the lower grid.

Another PWR component considered was the thermal shield. Although this component does not exceed Class C limits, it is included because it represents the largest source of blending material for concentration averaging of the other LLW components that do exceed Class C limits (See Appendix E-5).

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 was not evaluated in this study, but was found by previous studies to be well below Class C limits (Reference 1). The Combustion Engineering units do not have thermal shields, and the large Westinghouse units have four thermal pad plates instead of a full cylindrical shield.

As shown previously in Table 2, 73 operating PWRs were considered. Seven of these plants were Babcock & Wilcox plants, 15 were Combustion Engineering plants, and 51 were Westinghouse plants. The Westinghouse plants were divided into two groups. The large plants were the 17×17 arrays (34 plants), and the midsize plants were the 15×15 arrays (11 plants). The smaller Westinghouse plants were conservatively assumed to be the same as the midsize plants. For Babcock & Wilcox, a midsize plant (15×15 array) was considered. For Combustion Engineering plants, a midsize plant (16×16 array) was considered. Four other PWR plants have recently been shut down for decommissioning; Rancho Seco, Yankee Rowe, San Onofre Unit 1, and Trojan. Based on current plans, GTCC LLW could be available from these shutdown plants in 1994.

5.2 Methodology Used to Determine Component Activities and Classification Status

The ANISN¹¹ and ORIGEN2¹² codes were employed to estimate end-of-life activities for decommissioning components. These are the same codes used previously with typical PWR and BWR plant results listed in NRC NUREG reports (References 6 and 7) on decommissioning, but they have been updated to include new cross-section libraries. For these calculations, the reactor models differed slightly from those reported in the NRC documents.

The ANISN code was used to develop radial and axial neutron transport models. Unlike the NUREG approach, the ANISN radial results were then normalized to actual surveillance capsule flux data obtained from plant-specific surveillance reports.^f This normalization tied 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 source term calculated from the normalized ANISN radial flux was also used as input to the axial ANISN cases. Thus, the axial ANISN results were also normalized to the surveillance capsule data.

The normalized ANISN flux data were then used in conjunction with the operating histories as input to ORIGEN2 to determine component activities. The ORIGEN2 cross-section libraries used were more recent versions than those used in the NUREG documents (References 6 and 7).

f. Based on the authors' working knowledge of specific plants.

The following steps were taken to complete each reactor analyses:

1. Create a model with geometric input from reactor plant drawings and with material compositions from NUREG-3474.¹³
2. Determine the appropriate core enrichment and power levels for the particular reactor model.
3. Run ORIGEN2 with core enrichments and power levels appropriate for the particular reactor model. This yields core region flux data.
4. Using the ORIGEN2 core region fluxes in conjunction with the geometric model and the material compositions, run ANISN calculations for components of interest located radially outside the core to establish
 - a. Detailed radial fluxes
 - b. The spectral fractions of thermal and total neutron fluxes.
5. Compare and normalize the ANISN radial flux profiles to surveillance capsule analysis data at the clad-vessel interface. This includes normalizing the total neutron source in the reactor core.
6. Using the normalized core region flux in conjunction with the geometric model and material compositions, run ANISN calculations for components of interest located axially outside the core to establish
 - a. Detailed normalized axial fluxes
 - b. The spectral fractions of thermal and total neutron fluxes.
7. Using the normalized ANISN results, calculate weighting factors that indicate ratios of the spectral fractions in the core region relative to the spectral fractions in regions of interest outside the core.
8. Input the normalized ANISN fluxes into ORIGEN2 and use a reactor core cross-section set to calculate the concentrations of radionuclide activity in components of interest for a given irradiation time.
9. With all the same input, except using a thermal neutron cross-section set, rerun ORIGEN2 to again calculate the concentrations of radionuclide activity in components of interest for a given irradiation time. Adjust the results for the local area temperatures.
10. Use the weighting factors to combine the results of the two ORIGEN2 runs to yield the predicted activation in components of interest outside the core.

The remainder of this section details how this methodology was employed and discusses one of the cases considered.

5.2.1 Reference Plants

The scope of this study did not warrant developing models for all of the LWRs listed in Tables 1 and 2. Instead, Table 14 summarizes the characteristics of the six representative 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 conservatively represent values for the smaller BWR-2 and BWR-3 models. For PWRs, both the Babcock & Wilcox and Combustion Engineering plants were selected, also among the largest from these vendors. Similarly, two of the larger reactor models from Westinghouse were selected. The large Westinghouse model selected was a Standardized Nuclear Unit Power Plant System (SNUPPS) model. The midsize plant evaluated will conservatively represent values for the smaller Westinghouse plants.

5.2.2 One-Dimensional Transport Calculations

The ANISN computer code (Reference 11) 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, and the planar geometry was used for the axial models. Both radial and axial models calculated real flux solutions using a P3 maximum order of scatter and an S8 order of angular quadrature to define neutron scatter assumptions.

During the analyses, weighting factors were developed from the ANISN results to indicate ratios between neutron energies in the core and neutron energies in regions of interest outside the core. These weighting factors were later used to combine results from two ORIGEN2 activation calculations performed with two different neutron cross-section sets.

A 1991 WMG study (Reference 1) on these same six reactor models used the CASK-81 cross-section library¹⁴ for neutron cross-sections for the radial models. That study also used the Oak Ridge National Laboratory's GIP computer code¹⁵ to read the nuclide-organized CASK-81 cross-section library and create a group-organized input library to be read by ANISN.

Table 14. Summary of reference reactors analyzed to predict activation levels of decommissioning components.

Reference case	A	B	C	D	E	F
Reactor vendor	GE	GE	West	West	CE	B&W
Reactor type	BWR-4	BWR-6	Midsize	Large		177
Number of fuel assemblies	560	748	173	193	217	177
Heat output (MW)	2,440	3,580	3,020	3,410	3,390	2,770
Initial enrichment	3.50%	3.50%	4.50%	4.50%	4.50%	4.50%

For the axial models, the SAILOR cross-section library for neutron cross-sections¹⁶ was used because it is more appropriate for vessel internals modeling. The SAILOR cross-section data set was specifically developed for pressure vessel fluence calculations and surveillance capsule analysis, whereas the CASK-81 data set was developed for shielding analysis of spent fuel shipping casks.

To determine the impact of the different cross-section libraries, and to evaluate whether the radial models should be changed to use the SAILOR cross-section library, a single radial case from the earlier WMG study was rerun with the SAILOR cross-sections. Although the CASK-81 data set resulted in a harder neutron spectrum^g than the SAILOR results, the total flux levels in the regions of interest were within 10% of each other. A plot of the thermal flux values as a function of radial distance is shown in Figure 4. This figure shows that, in the regions outside the core, there is good agreement between the thermal fluxes predicted with analyses using the two different cross-section libraries.

The methodology used to incorporate ANISN results into the weighting factors for core and thermal activation fractions is based on the relationship between the thermal to total flux ratio in the region of interest and the thermal to total flux ratio in the core region. The core activation fractions for the SAILOR results were within 3% of the previous CASK-81 results. The activation analysis results for the SAILOR case were within 10% of the previous results. These variances were considered acceptable for this study, and the previously run radial model results with CASK-81 were retained. It should be noted that the SAILOR cross-section data set is currently used in decommissioning studies and was used for the waste characterization of the vessel internals at Yankee Rowe^h in 1993.

5.2.3 Point Neutron Activation Calculations

The ORIGEN2 computer code (Reference 12) was used to calculate the activation and depletion of radionuclides in components exposed to neutrons. Twenty-two different one-group neutron cross-section libraries for different reactor types and spectrally averaged cores were available in the standard ORIGEN2 libraries. Only three of these libraries were required in this analyses. The first of these one-group cross-section sets is based on a collapsed multigroup cross-section set representative of a PWR reactor core with a fuel burnup of 50,000 MWD/MTU (PWR-UE).¹⁷ The second cross-section data set is representative of a BWR that has a fuel burnup of 27,500 MWD/MTU (BWR-USO, Reference 17). 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 (n/cm^2sec), 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. Both calculations were required in

g. "Harder spectrum" is a common term indicating that the thermal and low-energy epithermal flux spectra are shifted up in energy.

h. WMG Inc., *Reactor Internals Preliminary Characterization and Packaging Plan for Yankee Rowe*, WMG-9305, June 1993.

Figure 4. B&W radial ANISN case thermal flux results—use of SAILOR vs CASK-81 cross-section libraries.

order to account for the fast-to-thermal flux variation by component location. Based on the ANISN spectral results of fast-to-thermal neutron flux ratios, the two ORIGEN2 calculations were appropriately weighted to obtain location-specific neutron activation.

In the second calculation, the thermal one-group cross sections at 0.0253 eV and 293 K neutron energies were modified slightly to account for radial temperature variations for location-specific components external to the core. The activation nuclides were assumed to be perfect $1/v$ -absorbers, and a simple relation involving the square-root of the ratio of the reference temperature to the actual component temperature was used and applied to the thermal cross-sections. Activation results calculated with the thermal library were adjusted for the local area temperatures to reflect the decrease in activation at elevated temperatures.

5.2.4 Operating Histories

For the two BWR cases, the 1991 WMG study (Reference 1) assumed that the plants operated for 40 years of commercial operation at a 65% capacity factor. For this study, BWR cases were also run at 20, 30, and 60 years of operation at a 65% capacity factor. The BWR operating cycle consisted of 1.34 years at full power followed by 0.7 years of no operation.

For the four PWR cases, the 1991 WMG study (Reference 1) assumed a 70% capacity factor over the same 40-year time period. For this study, PWR cases were also run at 20, 30, and 60 years of operation at a 70% capacity factor. The PWR operating cycle consisted of 1.05 years at full power followed by 0.45 years of shutdown.

For the time frames dictated by study requirements, ORIGEN2 outputs did not correspond exactly to the end of operating cycles. A later section explains how this impacted the reporting of short-lived isotopic activities.

5.2.5 Radial ANISN Models

The modeling of the reference reactors for this activation analysis included geometry and materials considerations.

The PWR radial ANISN 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 radial 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 were used. Small components that may be found as part of the reactor internals were neglected for this evaluation. Both models included all air and water gaps.

The reactor core region of the reference reactor radial ANISN models were assumed to be composed 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 regional fuel compositions were determined by running ORIGEN2 with plant-specific parameters for combinations of 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 then used as input to the radial ANISN model.

5.2.6 Radial ANISN Model Calculations

Two radial ANISN flux calculations were performed for each reference reactor. The first calculation assumed an axially averaged, flat core power distribution, with the total neutron source equal to that calculated by ORIGEN2 for a three-region core. The results of this first calculation were compared to results obtained from the analysis of the surveillance capsules. In the second ANISN radial calculation, the neutron flux above 1 MeV was normalized to those surveillance capsule results.

5.2.7 Axial ANISN Models

The axial models of the reference reactors were considerably more complex than the corresponding radial models due to the number of components present and the significant variations in the water and metal volume fractions in the axial direction. Models of heterogeneous structural components were homogenized to contain the appropriate materials volume fractions within the core equivalent radius. The volume-averaged parameters of the three radial core regions were used, and a flat radial power distribution in the active core region was assumed.

The PWR axial ANISN models were based on detailed reviews of component physical characteristics in conjunction with their axial locations at the core centerline. The axial models typically consisted of the lower core support structures, core support plate, fuel assembly bottom nozzle, fuel spacer/water region, active core region, spacer plenum, fuel assembly top nozzle, upper core plate, and control rod guide tubes. The number of axial regions modeled varied considerably among reactor vendors, and there was significant variation in the distances from the active fuel to the components of interest due to the different fuel assembly design parameters. The control rod absorber sections were not included in the PWR models since they are fully withdrawn during full power operation and would not have a significant impact on the regions of interest. Neglecting the control rod absorbers may have resulted in slightly conservative estimates for the upper core support plates. Cycle-averaged boron concentrations in the water moderator were incorporated in the cases, and the variation in the inlet and outlet water temperatures was also factored in.

The BWR axial ANISN models consisted of the control rods and guide tubes, flow orifice, core support plate, fuel support castings and plate, fuel assembly lower tie plate, active core region, gas plenum, top fuel guide, hold down springs, fuel assembly upper tie plate, fuel assembly handle, and the steam dome. The active core region for the BWR axial cases were broken down into six separate regions to account for control rod positions and steam voiding distribution axially. The BWR-4 model included a 144-in. active fuel length, while the BWR-6 case incorporated a 150-in. active core. Differences in moderator temperature and axial variation in the moderator steam volume fraction were taken into account.

5.2.8 Axial ANISN Model Calculations

One axial ANISN calculation was performed for each reference reactor. Prior to the calculation, the total neutron source in the reactor core was normalized to the volume-averaged total flux obtained from the surveillance capsule normalized radial cases.

5.3 Description of a Typical Reference Plant Case

The characteristics of the large Westinghouse unit are summarized in Table 15. The parameters used for the axial ANISN model are presented in Table 16.

5.3.1 Radial ANISN Model

The parameters used for the radial ANISN model are presented in Table 15. The model neglected the reactor vessel thermal insulation, but did consider the primary shield wall. The biological shield was assumed to be 6 ft of standard concrete.

Two radial ANISN calculations were normally required per reactor model. However, in the case of the large Westinghouse PWR with the thermal shield configured as separate pads, four radial ANISN calculations were required. Because the thermal shield pad is intermittent (i.e., some locations have the pad and some do not), two calculations were required for the core baffle and barrel with no thermal shield pad, and two other calculations were required for the configuration including the thermal shield pads. The first set of calculations (one with pads and one without) 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 set of calculations, the ANISN-calculated neutron flux above 1 MeV was normalized to the results obtained by Westinghouse for the plant-specific surveillance capsule. The azimuthal peak location at the vessel clad interface and the capsule location reside just outside the thermal shield pads. Therefore, the normalized results were based on a weighted average of the ANISN results for the cases with and without the pads, using the fraction of the periphery covered by the pads (37.5%) as the weighting factor. The core internals configuration and locations of the surveillance capsules are illustrated in Figure 5.

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 $3.14\text{E}+10$ n/cm²sec. This value was adjusted to yield an azimuthally and axially averaged neutron flux above 1 MeV of $2.38\text{E}+10$ n/cm²sec, which was used to normalize this study's radial ANISN model. A plot for the normalized thermal flux as a function of radial location is shown on Figure 6.

The thermal and total neutron fluxes were then determined using the normalized ANISN flux results for the core baffle and core barrel. The fraction of the neutron flux that fell into the thermal energy range was calculated from the ANISN output data, as was the fraction of neutron flux above thermal energies. The resultant spectral fractions were then compared to the spectral fractions in the core region and weighting factors were determined. The total neutron flux was then input to two ORIGEN2 cases (as explained in Section 5.2.3) for each configuration of components, for a total of four ORIGEN2 cases. Finally, radionuclide activities in curies per gram for the composite core spectrum were determined by applying the appropriate weighting factors to each of the ORIGEN2-calculated activities and summing them.

Table 15. Large Westinghouse PWR—radial ANISN model input.

Reference Case D—reactor summary	
Rated thermal power (MW)	3,411
Number of fuel assemblies	193
Average moderator temperature (°F)	591.8
Average boron concentration in moderator (ppm)	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 (cm ³)	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
Core fractions	
Fuel	0.2759
Moderator	0.5896
Zircaloy-4	0.1028
Inconel-718	0.0007
Void	0.0311
Total core fractions	1.000
Moderator density at power (g/cm ³)	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	241.62
Air gap outer radius	307.34
Primary shield liner outer radius	307.98
Primary shield outer radius	490.86

Table 16. Westinghouse large reference Case D axial ANISN model input summary.

Region name	Region number	Region start (cm)	Region finish (cm)	Volume fractions				
				304 SST	Zircaloy-4	Ino-718	Void	Moderator
Water reflector below LCS	1	0.00	91.44	0.034	0.000	0.000	0.000	0.966
Lower core support	2	91.44	142.24	0.113	0.000	0.000	0.000	0.887
Core support columns	3	142.24	201.30	0.100	0.000	0.000	0.000	0.900
Lower core support plate	4	201.30	206.38	0.607	0.000	0.000	0.000	0.393
Assembly bottom nozzle	5	206.38	213.33	0.171	0.000	0.000	0.000	0.829
Core inlet water	6	213.33	217.27	0.000	0.000	0.000	0.000	1.000
Homogenized fuel	7	217.27	583.03	Refer to radial parameters, Table 15				
Gas plenum	8	583.03	600.56	0.000	0.097	0.015	0.275	0.614
Core exit water	9	600.56	604.85	0.000	0.000	0.000	0.000	1.000
Assembly top nozzle	10	604.85	614.05	0.268	0.000	0.036	0.000	0.696
Upper core plate	11	614.05	621.67	0.530	0.000	0.000	0.000	0.470
Upper core support columns	12	621.67	774.07	0.100	0.000	0.000	0.000	0.900

VG94 0011

Figure 5. Arrangement of surveillance capsules in the large Westinghouse reactor vessel.

R94 0756

Figure 6. Westinghouse large PWR thermal neutron flux—surveillance capsule normalized.

5.3.2 Axial ANISN Model

The parameters used for the axial ANISN model are presented in Table 16. The normalized thermal and total neutron fluxes were determined using the ANISN calculation results for the upper and lower core support plates. The total neutron flux was then input to two ORIGEN2 cases (as explained in Section 5.2.3) for each component, for a total of four ORIGEN2 cases. The thermal to total flux ratios in the regions of interest were used in conjunction with the thermal to total flux ratios in the core regions to determine the thermal and core activation fractions (i.e., the weighting factors); these fractions were used to weight the ORIGEN2 case results in order to obtain radionuclide activities in curies per gram for the composite core spectrum.

5.4 Results of Reference Plant Cases

The 1991 WMG study (Reference 1) analyzed components after 40 years of operation; these results represent the base case. The impacts of the shorter operating times, 20 and 30 years, and the longer operating time, 60 years, are discussed separately.

The reader should note that, although this section describes many components as exceeding Class C limits, these components may not be classified as GTCC LLW. These estimates are valid only as input to a data base that factors in concentration averaging and other such considerations to calculate actual amounts of GTCC LLW from light water reactors. See Appendix E-5 for information on concentration averaging of LWR components.

5.4.1 BWR Case Results

Table 17 presents the reference plant results for the BWR-4 case, and Table 18 presents the results for the BWR-6 case, both after 40 years of operation. This operating time period corresponded to 20 fuel cycles, and the results reflect conditions after discharge. Table 18 also shows the ORIGEN2/ANISN-calculated and capsule-normalized fluxes for the BWR-6 case. Note that the normalized flux is about 35% lower than the ORIGEN2/ANISN calculated value. Similar flux reductions would be expected if surveillance capsule data were available for the BWR-4 plant.

As shown in the tables, the top fuel guides for both plants are well above Class C limits after 40 years of operation. Both core shrouds are also above Class C limits, but pieces of them could be below Class C limits. Due to the distance between the core support plate and the active fuel region and the positioning of the control rod blades, the core support plates are below Class C limits. The fuel support castings protrude through the core support plate with about 3 in. above the plate and 5 in. below the plate. Since the axially averaged flux levels for the fuel support castings were within 8% of those for the core support plate, these components are also below Class C limits.

Table 17. General Electric BWR-4 reference Case A—results of activation analysis of decommissioning components, after 40 years operation, immediately after shutdown.

Component name	Top fuel guide	Core shroud	Fuel support castings	Core support plate
Volume m ³	0.28	2.85	0.50	0.44
(ft ³)	10.0	100.8	17.6	15.6
Weight (lb) ^a	5000	50400	8800	7800
Mass (kg) ^a	2273	22909	4000	3545
Flux (n/cm ² -sec) (ORIGEN2/ANISN) ^b	5.82E+13	5.89E+12	1.30E+12	1.20E+12
Normalized flux (Surv. capsule)	N/A	N/A	N/A	N/A
10 CFR 61 status -Table 1 Fraction	23.57	3.04	0.43	0.40
-Table 2 Fraction	24.94	3.53	0.22	0.20
Activity (curies)				
C-14	1.01E+02	9.50E+01	1.04E+00	8.55E-01
Mn-54	1.27E+04	1.39E+04	7.61E+02	6.24E+02
Fe-55	5.80E+05	5.65E+05	5.84E+03	4.79E+03
Co-60	1.85E+05	3.02E+05	5.06E+03	4.15E+03
Ni-59	2.34E+02	5.16E+02	6.35E+00	5.21E+00
Ni-63	4.94E+04	7.05E+04	7.53E+02	6.18E+02
Nb-94	7.98E-01	9.41E-01	3.02E-02	2.48E-02
Total curies	8.27E+05	9.52E+05	1.24E+04	1.02E+04

a. From Table 13.

b. Flux calculated by ANISN based on input core flux calculated by ORIGEN2.

Table 18. General Electric BWR-6 reference Case B—results of activation analysis of decommissioning components, after 40 years operation, immediately after shutdown.

Component name	Top fuel guide plate	Core shroud	Fuel support castings	Core support plate
Volume m ³ (ft ³)	0.46 16.40	4.92 174.0	0.64 22.70	0.53 18.80
Weight (lb) ^a	8200	87000	11350	9400
Mass (kg) ^a	3727	39545	5159	4273
Flux (n/cm ² -sec) (ORIGEN2/ANISN) ^b	4.77E+13	5.57E+12	1.60E+12	1.48E+12
Normalized flux (Surv. capsule)	3.04E+13	3.55E+12	1.02E+12	9.43E+11
10 CFR 61 status -Table 1 Fraction	11.61	1.76	0.21	0.21
-Table 2 Fraction	11.23	1.92	0.11	0.11
Activity (curies)				
C-14	6.95E+01	8.72E+01	6.50E-01	5.38E-01
Mn-54	1.23E+04	1.55E+04	4.75E+02	3.93E+02
Fe-55	4.01E+05	5.17E+05	3.64E+03	3.01E+03
Co-60	1.47E+05	2.96E+05	3.16E+03	2.61E+03
Ni-59	1.90E+02	5.07E+02	3.96E+00	3.28E+00
Ni-63	3.65E+04	6.60E+04	4.70E+02	3.89E+02
Nb-94	6.68E-01	9.60E-01	1.88E-02	1.56E-02
Total curies	5.97E+05	8.95E+05	7.74E+03	6.41E+03

a. From Table 13.

b. Flux calculated by ANISN based on input core flux calculated by ORIGEN2.

5.4.2 Westinghouse PWR Case Results

Table 19 presents the reference plant results for the intermediate reference plant, and Table 20 presents the results for the large reference plant, both after 40 years of operation. This operating time period corresponded to 26.4 operating cycles, and the results reflect conditions after discharge. These tables also show the ORIGEN2/ANISN calculated and normalized fluxes for the two Westinghouse cases.

As shown in the tables, the core baffles are well above Class C limits after 40 years of operation. This corresponds to all earlier study results and the experience at Yankee Rowe.ⁱ As a general rule, core baffles will always exceed Class C limits.

The lower core support barrels are above Class C limits; this result is consistent with the experience at Yankee Rowe.

The upper and lower core support plates also exceed Class C limits with the lower core support plate as the higher specific activity component. Experience at Yankee Rowe showed a significant axial flux variation in both the upper and lower core support plates.

5.4.3 Other PWR Case Results

Table 21 presents the reference plant results for the Combustion Engineering plant case, and Table 22 presents the results for the Babcock & Wilcox reference plant case after 40 years of operation. This operating time period corresponded to 26.4 operating cycles, and the results reflect conditions after discharge. These tables also show the ORIGEN2/ANISN calculated and normalized fluxes for both cases.

For the Combustion Engineering reference plant in Table 21, the core baffle, core barrel, and grid plate are well above Class C limits. Unlike the results for the Westinghouse reference plants, the fuel assembly alignment plate (upper core support plate) is below Class C limits.

For the Babcock & Wilcox reference plant in Table 22, the core baffle is well above Class C limits and is clearly GTCC LLW. The core support assembly barrel results are much lower, and the lower grid just exceeds Class C limits. Like the Combustion Engineering reference plant results, the upper plenum assembly (upper core support plate) is below Class C limits.

5.4.4 Time Dependence of Case Results

ORIGEN2 cases were run for each component considered for plant operating study time frames of 20, 30, 40 and 60 years. Sometimes the end of a study time frame happened to coincide with the end of an operating cycle, so that a period of no reactor operation occurred at the end of the time frame. When this happened, decay of the short-lived radionuclides caused the total activity to decrease significantly.

i. WMG Inc., *Reactor Internals Preliminary Characterization and Packaging Plan for Yankee Rowe*, WMG-9305, June 1993.

Table 19. Westinghouse intermediate reference Case C—results of activation analysis of decommissioning components, after 40 years operation, immediately after shutdown.

Component name	Upper core support plate	Core baffle	Lower core support barrel	Lower core support plate
Volume m ³	0.35	1.26	3.49	0.38
(ft ³)	12.24	44.6	123.4	13.4
Weight (lb) ^a	6120	22300	61700	6700
Mass (kg) ^a	2782	10136	28045	3045
Flux (n/cm ² -sec) (ORIGEN2/ANISN) ^b	3.27E+12	1.38E+14	1.94E+12	2.41E+13
Normalized flux (Surv. capsule)	4.89E+12	2.06E+14	2.91E+12	3.61E+13
10 CFR 61 status -Table 1 Fraction	2.09	47.58	1.13	12.99
-Table 2 Fraction	2.06	37.33	0.94	11.68
Activity (curies)				
C-14	6.76E+00	5.88E+02	3.09E+01	5.23E+01
Mn-54	2.06E+03	3.46E+05	1.35E+04	1.67E+04
Fe-55	3.85E+04	3.18E+06	1.74E+05	2.93E+05
Co-60	2.35E+04	1.48E+06	1.19E+05	1.38E+05
Ni-59	3.75E+01	1.46E+03	1.81E+02	1.80E+02
Ni-63	5.00E+03	3.29E+05	2.30E+04	3.10E+04
Nb-94	9.14E-02	8.65E+00	5.34E-01	6.73E-01
Total curies	6.91E+04	5.33E+06	3.29E+05	4.79E+05

a. From Table 13.

b. Flux calculated by ANISN based on input core flux calculated by ORIGEN2.

Table 20. Westinghouse large reference Case D—results of activation analysis of decommissioning components, after 40 years operation, immediately after shutdown.

Component name	Upper core support plate	Core Baffle	Lower core support barrel	Lower core support plate
Volume m ³	0.52	1.99	3.49	0.33
(ft ³)	18.4	70.4	123.4	11.8
Weight (lb) ^a	9200	35200	61700	5900
Mass (kg) ^a	4182	16000	28045	2682
Flux (n/cm ² -sec)	3.63E+12	1.53E+14	7.90E+12	2.26E+13
(ORIGEN2/ANISN) ^b				
Normalized flux (Surv. capsule)	3.52E+12	1.48E+14	7.65E+12	2.19E+13
10 CFR 61 status	1.38	43.93	2.99	9.29
-Table 1 Fraction				
-Table 2 Fraction	1.21	32.43	2.50	8.28
Activity (curies)				
C-14	5.99E+00	7.06E+02	8.27E+01	3.25E+01
Mn-54	2.35E+03	4.01E+05	3.55E+04	1.07E+04
Fe-55	3.38E+04	3.88E+06	4.64E+05	1.82E+05
Co-60	2.20E+04	2.29E+06	3.13E+05	8.70E+05
Ni-59	3.37E+01	2.82E+03	4.76E+02	1.14E+02
Ni-63	4.41E+03	4.52E+05	6.10E+04	1.93E+04
Nb-94	9.57E-02	1.28E+01	1.41E+00	4.25E-01
Total curies	6.03E+04	6.62E+06	8.39E+05	2.88E+05

a. From Table 13.

b. Flux calculated by ANISN based on input core flux calculated by ORIGEN2.

Table 21. Combustion Engineering reference Case E—results of activation analysis of decommissioning components, after 40 years operation, immediately after shutdown.

Component name	Fuel assembly alignment plate	Core baffle	Core barrel	Core support plate
Volume m ³ (ft ³)	0.57 20.2	2.48 87.8	8.11 286.4	0.33 11.6
Weight (lb) ^a	10100	43900	143200	5800
Mass (kg) ^a	4591	19955	65091	2636
Flux (n/cm ² -sec) (ORIGEN2/ANISN) ^b	3.03E+11	1.31E+14	1.23E+13	1.85E+13
Normalized flux (Surv. capsule)	N/A	N/A	N/A	N/A
10 CFR 61 status -Table 1 Fraction	0.11	34.29	4.65	6.55
-Table 2 Fraction	0.08	26.34	3.89	5.77
Activity (curies)				
C-14	4.48E-01	7.80E+02	3.12E+02	2.21E+01
Mn-54	2.34E+02	4.37E+05	1.32E+05	7.49E+03
Fe-55	2.50E+03	4.25E+06	1.75E+06	1.24E+05
Co-60	1.78E+03	2.20E+06	1.10E+06	6.01E+04
Ni-59	2.58E+00	2.43E+03	1.58E+03	7.85E+01
Ni-63	3.28E+02	4.57E+05	2.20E+05	1.32E+04
Nb-94	8.75E+03	1.26E+01	5.16E+00	2.96E-01
Total curies	4.85E+03	7.34E+06	3.21E+06	2.05E+05

a. From Table 13.

b. Flux calculated by ANISN based on input core flux calculated by ORIGEN2.

Table 22. Babcock & Wilcox reference Case F—results of activation analysis of decommissioning components, after 40 years operation, immediately after shutdown.

Component name	Upper grid assembly	Core baffle	Core support assembly barrel	Lower grid
Volume m ³ (ft ³)	0.42 15	1.45 51.2	2.51 88.8	0.41 14.4
Weight (lb) ^a	7500	25600	44400	7200
Mass (kg) ^a	3409	11636	20182	3273
Flux (n/cm ² -sec) (ORIGEN2/ANISN) ^b	6.18E+11	1.20E+14	1.26E+13	4.89E+12
Normalized flux (Surv. capsule)	3.61E+11	7.02E+13	7.34E+12	2.86E+12
10 CFR 61 Status -Table 1 Fraction	0.17	21.70	2.93	1.11
-Table 2 Fraction	0.19	16.66	2.57	1.12
Activity (curies)				
C-14	7.55E-01	2.90E+02	6.23E+01	5.71E+00
Mn-54	1.72E+02	1.35E+05	2.42E+04	1.29E+03
Fe-55	4.33E+03	1.59E+06	3.52E+05	3.22E+04
Co-60	2.49E+03	8.24E+05	2.24E+05	1.30E+04
Ni-59	4.14E+00	1.00E+03	3.37E+02	1.69E+01
Ni-63	5.59E+02	1.70E+05	4.52E+04	2.87E+03
Nb-94	8.64E+03	4.52E+00	9.80E-01	5.91E-02
Total curies	7.56E+03	2.72E+06	6.45E+05	4.94E+04

a. From Table 13.

b. Flux calculated by ANISN based on input core flux calculated by ORIGEN2.

Table 23 presents the results for the BWR-4 reference plant core shroud as a function of operating time. This table presents values for the relatively short-lived radionuclides: Mn-54, Fe-55, and Co-60; and the longer lived radionuclides, C-14, Ni-59, Ni-63, and Nb-94. The short-lived radionuclides represent over 90% of the activity. As shown, total activity decreases with time after 20 years due to saturation of the shorter-lived radionuclides. Conversely, the longer-lived radionuclide activities, which drive 10 CFR 61 classification, increase with operating time. The 10 CFR 61 Table 1 and Table 2 sums of fractions increase linearly from 20 to 60 years of operation.

Table 24 presents the results for the PWR large Westinghouse reference plant core barrel as a function of operating time. This table also presents values for the relatively short-lived radionuclides, Mn-54, Fe-55, and Co-60, and the longer-lived radionuclides, C-14, Ni-59, Ni-63, and Nb-94. The short-lived radionuclides represent over 90% of the activity. As shown, total activity increases with time up to about 30 years and then levels off due to saturation of the shorter-lived radionuclides. Note that for this reactor, the end of the 40-year study time frame coincided with a period of no reactor operation, during which decay of the short-lived radionuclides caused the total activity to decrease. As with all components, the longer-lived radionuclide activities, which drive 10 CFR 61 classification, consistently increase with operating time. The 10 CFR 61 Table 1 and Table 2 sums of fractions increase linearly from 20 to 60 years of operation.

Table 25 summarizes the activation status for all components considered in each reference plant as a function of operating time from 20 to 60 years. A "Yes" indicates that either the 10 CFR 61 Table 1 or Table 2 sum of fractions exceeded unity immediately following final reactor shutdown after the period of operation indicated. In some cases, like the BWR top fuel guides and PWR core baffles, the sum of the fractions was well above unity. In other cases, like the PWR core barrels and upper core support plates, the sum of the fractions was only several times unity.

Table 23. BWR-4 core shroud activation as a function of time.

Operating time (years) No. of operating cycles	20 (10)	30 (15)	40 (20)	60 (30)
10 CFR 61 status				
-Table 1				
Fraction	1.52	2.35	3.04	4.39
-Table 2	1.98	2.83	3.53	4.74
Fraction				
Activity (curies)				
C-14	4.85E+01	7.24E+01	9.49E+01	1.41E+02
Mn-54	1.29E+04	1.53E+04	1.39E+04	1.12E+04
Fe-55	2.16E+06	5.93E+05	5.64E+05	5.19E+05
Co-60	2.99E+05	3.14E+05	3.02E+05	2.78E+05
Ni-59	2.85E+02	4.08E+02	5.16E+02	7.07E+02
Ni-63	3.96E+04	5.64E+04	7.04E+04	9.46E+04
Nb-94	4.85E-01	7.21E-01	9.41E-01	1.38E+00
Total curies	2.51E+06	9.78E+05	9.51E+05	9.03E+05

Table 24. Large Westinghouse lower core support barrel activation as a function of time.

Operating time (years)	20	30	40	60
No. of operating cycles	(13.5)	(19)	(26.4)	(39)
<hr/>				
10 CFR 61 status				
-Table 1				
Fraction	1.54	2.27	2.99	4.47
-Table 2	1.38	1.97	2.50	3.47
Fraction				
Activity (curies)				
C-14	4.21E+01	6.23E+01	8.27E+01	1.24E+02
Mn-54	3.58E+04	4.08E+04	3.55E+04	4.07E+04
Fe-55	4.64E+05	4.91E+05	4.64E+05	4.90E+05
Co-60	2.99E+05	3.21E+05	3.13E+05	3.18E+05
Ni-59	2.51E+02	3.65E+02	4.76E+02	6.95E+02
Ni-63	3.38E+04	4.81E+04	6.10E+04	8.48E+04
Nb-94	7.21E-01	1.06E+00	1.41E+00	2.11E+00
Total curies	7.97E+05	8.60E+05	8.38E+05	8.94E+05

Table 25. GTCC classification status for decommissioning components considered in each reference LWR as a function of plant operating time.^a

Operating times	20 years GTCC	30 years GTCC	40 years GTCC	60 years GTCC
A. BWR-4				
1. Top fuel guide	Yes	Yes	Yes	Yes
2. Core shroud	Yes	Yes	Yes	Yes
3. Fuel support castings	No	No	No	No
4. Core support plate	No	No	No	No
B. BWR-6				
1. Top fuel guide (plate)	Yes	Yes	Yes	Yes
2. Core shroud	Yes	Yes	Yes	Yes
3. Fuel support castings	No	No	No	No
4. Core support plate	No	No	No	No
C. Westinghouse intermediate				
1. Upper core support plate	Yes	Yes	Yes	Yes
2. Core baffle	Yes	Yes	Yes	Yes
3. Thermal shield	No	No	No	No
4. Lower core support barrel	No	No	Yes	Yes
5. Lower core support plate	Yes	Yes	Yes	Yes
D. Westinghouse large				
1. Upper core support plate	No	Yes	Yes	Yes
2. Core baffle	Yes	Yes	Yes	Yes
3. Thermal shield	No	No	No	No
4. Lower core support barrel	Yes	Yes	Yes	Yes
5. Lower core support plate	Yes	Yes	Yes	Yes
E. Combustion Engineering				
1. Fuel assembly alignment plate	No	No	No	No
2. Core baffle	Yes	Yes	Yes	Yes
3. Core barrel	Yes	Yes	Yes	Yes
4. Lower forging (top plate)	Yes	Yes	Yes	Yes
F. Babcock & Wilcox				
1. Upper grid assembly	No	No	No	No
2. Core baffle	Yes	Yes	Yes	Yes
3. Core support assembly barrel	Yes	Yes	Yes	Yes
4. Thermal shield	No	No	No	No
5. Lower grid	No	No	Yes	Yes

a. A "Yes" indicates that either of the 10 CFR 61 Table 1 or Table 2 sums of fractions exceeded unity immediately following final reactor shutdown after the period of operation indicated.

6. SUMMARY AND CONCLUSIONS

How much GTCC LLW an LWR generates depends on both plant type and fuel cycle. For this study, all LWRs listed in Tables 1 and 2 were considered, including 110 LWRs comprising 37 BWRs and 73 PWRs. The percentage of LLW components that individually exceed Class C limits was estimated for each waste type of interest. These percentages are listed in Table 26.

It is important to note that these estimates may not represent actual amounts of LWR waste that will be classified as GTCC LLW. These estimates are used only as input to a data base that factors in concentration averaging assumptions to calculate amounts of GTCC LLW from light water reactors.

Three types of LLW were considered:

- **Operations Activated Metals**—The components that are activated through exposure to neutron flux and periodically replaced during routine operations
- **Process Wastes**—The wastes, in the form of cartridge filters and decontamination project resins, that are generated from cleanup of liquids containing soluble and insoluble radioactive constituents
- **Decommissioning Activated Metals**—The structural components within the reactor vessel that are activated by neutron exposure and removed at decommissioning.

This study found that LLW exceeding Class C limits, in the form of operations activated metals and cartridge filters, represents a relatively small portion of the LLW routinely generated at LWRs. The estimated generation rates of BWR operations waste that exceeds Class C limits ranged from a low of 0.13 m^3 (4.5 ft^3) to a high of 0.21 m^3 (7.4 ft^3) per plant per fuel cycle. The PWR estimated generation rates, which varied with reactor vendor, ranged from a low of 0.050 m^3 (1.7 ft^3) to a high of 0.090 m^3 (3.2 ft^3) per plant per fuel cycle. Table 27 summarizes these estimated generation rates, organized by waste type and reactor vendor.

Full RCS decontamination projects with fuel assemblies remaining in the vessel will result in some deep bed resins that exceed Class C limits. Full RCS decontamination is an untried process; it has 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, the study assumed that plants with less than 20 years remaining through closure (i.e., closure in year 2013 or earlier) would not perform full RCS decontaminations with fuel assemblies in the vessel. The remaining plants were assumed to perform one full RCS decontamination project with fuel assemblies in the vessel sometime during their operating life; the resulting resin wastes that exceed Class C limits are reported as decommissioning waste quantities in DOE/LLW-114 Revision 1. For a typical BWR, the estimated volume of decontamination resin waste that exceeds Class C limits was 2.83 m^3 (100 ft^3) and for a PWR was 4.25 m^3 (150 ft^3). Table 27 summarizes the results of this analysis.

Table 26. Percentages of LLW components that individually exceed Class C limits.^a

	<u>Percent above Class C limits</u>
Operations Activated Metals	
1. BWR operations	
a. Control rod blades (CRBs)	
- Midsize plant conventional CRBs	10
- Midsize plant controlled cell CRBs	25/10 ^b
- Large plant conventional CRBs	10
- Large plant controlled cell CRBs	25/10 ^b
b. LPRM strings	75
c. IRM/SRM dry tubes	100
2. Westinghouse PWR operations	
a. Thimble plug assemblies	25
b. Thimble tubes	75
c. Primary source rods	100
3. CE PWR operations	
a. Instrument strings	75
b. Primary sources	100
4. B&W PWR operations	
a. In-core detectors	75
b. Primary sources	100
Process Wastes	
1. BWR cartridge filters	
a. Control rod drive	50
b. Fuel pool/vacuum	25
2. PWR cartridge filters	
a. Misc. cartridge filters	
- Reactor coolant system	25
- Seal water injection	25
- Cavity drain	25
- Spent fuel pool	25
b. B&W crud tank filters	100
3. BWR RCS decon resins	50
4. PWR RCS decon resins	50
Decommissioning Activated Metals	
1. BWR decommissioning	
a. Top fuel guide	100
b. Core shroud	100
2. PWR decommissioning	
a. Upper core support plate	50
b. Core baffle	100
c. Lower core barrel	100
d. Lower core support plate	100

a. For example, 75% of all BWR LPRM strings replaced from routine plant operations will have radionuclide concentrations above the Class C limit.

b. These two numbers represent the controlled cell CRBs and the conventional mode CRBs that operate together within the same plant.

Table 27. Operations activated metals and process wastes that exceed Class C limits.

	Number of plants	Volume per component m ³ (ft ³)	Total LLW units per cycle	Percent above Class C ^a	Units above Class C, per cycle ^b	Volume above Class C, per cycle ^c m ³ (ft ³)
Operations Activated Metals						
1. BWR operations	37					
a. Control rod blades (CRBs)						
- Midsize plant conventional CRBs	14	0.017 (0.6)	14	10	1.4	0.024 (0.84)
- Midsize plant controlled cell CRBs	15	0.017 (0.6)	14/11 ^d	25/10 ^d	4.4	0.074 (2.6)
- Large plant conventional CRBs	5	0.017 (0.6)	19	10	1.9	0.032 (1.1)
- Large plant controlled cell CRBs	3	0.017 (0.6)	19/15 ^d	25/10 ^d	6.2	0.106 (3.8)
b. LPRM strings	37	0.0028 (0.1)	12	75	9	0.026 (0.9)
c. IRM/SRM dry tubes	37	0.0023 (0.08)	2	100	2	0.005 (0.16)
2. Westinghouse PWR operations	51					
a. Thimble plug assemblies		0.0007 (0.024)	16	25	4	0.0028 (0.1)
b. Thimble tubes		0.0017 (0.06)	2	75	1.5	0.0025 (0.09)
c. Primary source rods ^e		0.0002 (0.007)	NA	100	NA	0.0002 (0.007)
3. CE PWR operations	15					
a. Instrument strings		0.002 (0.07)	24	75	18	0.036 (1.26)
b. Primary sources ^e		0.0006 (0.02)	NA	100	NA	0.0006 (0.02)
4. B&W PWR operations	7					
a. In-core detectors		0.0028 (0.1)	16	75	12	0.034 (1.2)
b. Primary sources ^e		0.001 (0.02)	NA	100	NA	0.001 (0.02)
Process Wastes						
1. BWR cartridge filters	37					
a. Control rod drive		0.002 (0.07)	30	50	15	0.03 (1.05)
b. Fuel pool/vacuum		0.014 (0.05)	12	25	3	0.042 (1.5)
2. PWR cartridge filters	73					
a. Misc. cartridge filters						
- Reactor coolant system		0.009 (0.3)	8	25	2	0.018 (0.6)
- Seal water injection		0.0028 (0.1)	16	25	4	0.011 (0.4)
- Cavity drain		0.0028 (0.1)	2	25	0.5	0.001 (0.05)
- Spent fuel pool		0.014 (0.5)	4	25	1	0.014 (0.5)
b. B&W crud tank filters	7	0.022 (0.8)	0.5	100	0.5	0.011 (0.4)
3. BWR RCS decon resins ^f		5.66 (200)	NA	50	NA	2.83 (100)
4. PWR RCS decon resins ^f		8.50 (300)	NA	50	NA	4.25 (150)

a. Source: Table 26.

b. Units per cycle × percent above Class C limits.

c. Volume per component × units above Class C, per cycle.

d. These two numbers represent the controlled cell CRBs and the conventional mode CRBs that operate together within the same plant.

e. Although primary source data are included under the column headings for unit volume and units per cycle, that terminology is not applicable to primary sources. Since primary sources are used only once at the startup of a reactor, the reported volumes are per plant, not per cycle. However, the volumes reported are per unit.

f. Although resin data are included under the column headings for unit volume and units per cycle, that terminology is not applicable to resin waste. First, resins are not quantified in separate units, as are the filters; the amounts listed reflect the total amount of resin produced during reactor decontamination operations. Second, this report assumes that a reactor undergoes full RCS fuel-in decontamination only once during its operating life; the volumes listed are total for the life of the plant, not per cycle. For this reason, the main report (DOE/LLW-114 Revision 1) includes RCS decontamination resins with decommissioning waste.

Decommissioning activated metals represent the largest potential source of nuclear utility 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 components that individually exceed Class C limits. Table 28 summarizes the results of this analysis extended to all LWRs. 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 (References 6 and 7). For BWR components, the results of this study indicated that, when calculated fluxes are normalized to surveillance capsule data, the calculated integrated fluxes were about 35% lower than those reported by the NRC, resulting in lower activities. Thus, while the BWR core shroud exceeds Class C limits, it does so by factors of 3 or less. These results indicate that further investigations are warranted to better define BWR flux levels.

For the small and large BWR plants, the fuel support castings and the core support plates are well below Class C limits even in scenarios where a reactor operates for 60 years. The top fuel guide is always well above Class C limits.

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 below Class C limits even in scenarios where a reactor operates for 60 years (i.e., under a 20-year license extension).

For the Combustion Engineering plants, the fuel assembly alignment plate is well below Class C limits; for the Babcock and Wilcox plants, the upper grid assembly is well below Class C limits even in scenarios where a reactor operates for 60 years.

Mixed results were found for the lower core barrel, a major PWR component. For the two Westinghouse PWRs, the lower core barrel was found to barely exceed Class C limits. Since these plants were considered to be representative of 51 PWRs, further evaluations could eliminate the lower core barrel from estimates of potential GTCC LLW in this group of PWRs. In the case of the Babcock and Wilcox and the Combustion Engineering plants, the core barrels exceed Class C limits even at 20 years of operation.

Table 28. Decommissioning activated metals that exceed Class C limits.

		Core shroud or baffle above Class C, volume	Lower core barrel above Class C, volume	Top fuel guide or Upper support plates above Class C, volume per plant	Orificed fuel supports or Lower support plates above Class C, volume per plant	Thermal shield above Class C, volume per plant
Number of plants	per plant m ³ (ft ³)	m ³ (ft ³)	m ³ (ft ³)	m ³ (ft ³)	m ³ (ft ³)	m ³ (ft ³)
Decommissioning activated metals						
BWR plants						
1. Midsize BWR plants	29	2.85 (101)	NA	NA	0.28 (10)	NA
2. Large BWR plants	8	4.92 (174)	NA	NA	0.46 (16)	NA
Total number of BWR plants	37					
PWR plants						
1. B&W reactors	7	1.45 (51)	2.51 (89)	NA	NA	0.41 (14)
2. CE reactors	15	2.48 (88)	8.11 (286)	NA	NA	0.33 (12)
3. Midsize Westinghouse	20	1.26 (45)	3.49 (123)	0.35 (12)	0.38 (13)	3.08 (109)
4. Large Westinghouse	31	1.99 (70)	3.49 (123)	0.52 (18)	0.33 (12)	1.26 (44)
Total number of PWR plants	73					
Grand total number of reactor plants	110					
NA = Not applicable						

7. REFERENCES

1. P. Tuite, K. Tuite, A. Levin, and M. O'Kelley, *Appendix F: Basis for Greater-Than-Class C Low-Level Radioactive Waste Light Water Reactor Projections*, DOE/LLW-114F, August 1991.
2. "World List of Nuclear Power Plants," *Nuclear News*, Vol. 36 No. 3, March 1993, pp. 56-58.
3. "List of Scheduled Outages at U.S. Nuclear Power Plants," *Nuclear News*, Vol. 36 No. 1, January 1993, p. 31.
4. "List of Scheduled Outages at U.S. Nuclear Power Plants," *Nuclear News*, Vol. 37 No. 1, January 1994, p. 40.
5. "U.S. Capacity Factors: Improvement Even at the Bottom," *Nuclear News*, Vol. 37 No. 7, May 1994, p. 38.
6. U.S. Nuclear Regulatory Commission, *Technology, Safety, and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station*, NUREG/CR-0130, Vols. 1 and 2 plus Addenda 1-3, June 1978.
7. U.S. Nuclear Regulatory Commission, *Technology, Safety, and Costs of Decommissioning a Reference Boiling Water Reactor Power Station*, NUREG/CR-0672, Vols. 1 and 2 plus Addenda 1-2, June 1980.
8. P. Tuite and K. Tuite, *Greater-Than-Class C Low-Level Radioactive Waste Characterization, Appendix E-5: Impact of the 1993 NRC Draft Branch Technical Position on Concentration Averaging of Greater-Than-Class C Low-Level Radioactive Waste*, DOE/LLW-114E-5, August 1994.
9. Code of Federal Regulations, 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," Office of the Federal Register, December 27, 1982.
10. James Sejvar et al., *Full RCS Decontamination Program: Waste Management Methodology and Waste Characterization*, NATD/PA/1207/89, Revision 1, December 1989.
11. Oak Ridge National Laboratory, *ANISN-ORNL One-Dimensional Discrete Ordinates Transport Code System with Anisotropic Scattering*, CCC-254H, Oak Ridge National Laboratory Radiation Shielding Information Center, April 1991.
12. Oak Ridge National Laboratory, *ORIGEN2 - Oak Ridge Isotopic Generation and Depletion Code Version 2.1*, CCC-371, Oak Ridge National Laboratory Radiation Shielding Information Center, August 1991.
13. J. C. Evans, *Long-Lived Activation Products in Reactor Materials*, NUREG/CR-3474, Pacific Northwest Laboratory, August 1984.
14. Oak Ridge National Laboratory, *CASK-81 - 22 Neutron, 18 Gamma-Ray Group, P₃ Cross*

Sections for Shipping Cask Analysis, DLC-023G, Oak Ridge National Laboratory Radiation Shielding Information Center, June 1987.

15. Oak Ridge National Laboratory, *GIP - Code System for Preparation of Group-Organized Cross-Section Libraries*, PSR-229 MICRO, Oak Ridge National Laboratory Radiation Shielding Information Center, April 1986.
16. Oak Ridge National Laboratory, *SAILOR - Coupled Self-Shielded 47 Neutron, 20 Gamma-Ray Group Cross Section Library for Light Water Reactors*, DLC-76B, Oak Ridge National Laboratory Radiation Shielding Information Center, March 1981.
17. S. B. Ludwig, *Standard and Extended Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code*, ORNL/TM-11018, Oak Ridge National Laboratory Radiation Shielding Information Center, December 1989.