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

Appendix F: Greater-Than-Class C Low-Level Radioactive Waste Light Water Reactor Projections

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

August 1991

GREATER-THAN-CLASS C LOW-LEVEL WASTE CHARACTERIZATION

Appendix F: Greater-Than-Class C Low-Level Radioactive Waste Light Water Reactor Projections

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P. Tuite
K. Tuite
A. Levin
M. O'Kelley

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Idaho National Engineering Laboratory
EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

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ABSTRACT

This study characterizes potential greater-than-Class C low-level radioactive waste streams, estimates the amounts of waste generated, and estimates their radionuclide content and distribution. Several types of low-level radioactive wastes produced by light water reactors were identified in an earlier study as being potential greater-than-Class C low-level waste, including specific activated metal components and certain process wastes in the form of cartridge filters and decontamination resins. Light water reactor operating parameters and current management practices at operating plants were reviewed and used to estimate the amounts of potential greater-than-Class C low-level waste generated per fuel cycle. The amounts of routinely generated activated metal components and process waste were estimated as a function of fuel cycle. Component-specific radionuclide content and distribution was calculated for activated metals components. Empirical data from actual low-level radioactive waste streams were used to estimate radionuclide content and distribution for process wastes. The greater-than-Class C low-level waste volumes that could be generated through plant closure were also estimated, along with volumes and activities for potential greater-than-Class C activated metals generated at decommissioning.

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APPENDIX F: GREATER-THAN-CLASS C LOW-LEVEL RADIOACTIVE WASTE LIGHT WATER REACTOR PROJECTIONS

1. INTRODUCTION

The objective of this study is to characterize potential greater-than-Class C (GTCC) low-level radioactive waste streams identified in an earlier WMG, Inc., study,¹ estimate the amounts of waste generated, and estimate their radionuclide content and distribution. The resultant estimates will be used by EG&G Idaho, Inc., as input to the Department of Energy (DOE) computer model for GTCC waste projections.

The earlier WMG, Inc., study analyzed low-level radioactive waste (LLW) streams to identify those waste streams that could be GTCC LLW as a function of concentration averaging scenario. Several types of LLW wastes produced by light water reactors (LWR) were identified as potential GTCC LLW under the most restrictive concentration averaging scenario considered. These wastes included specific activated metal components and certain process wastes in the form of cartridge filters and decontamination resins. The waste streams, their physical characteristics, and that portion of them which could be GTCC LLW was described in the previous study.

The amounts of GTCC LLW generated by LWRs are plant type and fuel cycle dependent. To define LWR operating parameters, a master list of LWRs was initially prepared. This list included start up dates, plant type, reactor model, and the most recent information on plant-specific fuel cycles and refueling outage times. Current LLW management practices at operating plants were then reviewed. These practice reviews were used to estimate the amounts of potential GTCC LLW generated per fuel cycle and routine generation rates. The amounts of routinely generated activated metal components and process waste were then estimated as a function of fuel cycle. Component-specific radionuclide content and distribution was calculated for activated metals components. Empirical data from actual LLW waste streams were used to estimate radionuclide content and distribution for process wastes. The GTCC LLW volumes that could be generated through plant closure were also estimated by LWR plant. These estimates were based on the GTCC LLW generation rates and the number of fuel cycles remaining for each LWR from January 1993 through plant closure. January 1993 is the date when the commercial LLW disposal sites are currently scheduled to close.

Volumes and activities for potential GTCC activated metals generated at decommissioning were also estimated. The methods used were similar to those described in the Nuclear Regulatory Commission (NRC) reports on decommissioning waste. Six typical LWR plants, four pressurized water reactors (PWRs) and two boiling water reactors (BWRs), were selected for detailed analysis. Plant-specific drawings and data were used to model the reactor vessel internals and conditions for each plant. Surveillance capsule reports, which are periodically prepared to determine integrated neutron exposure to the reactor vessel, were requested from the six plants selected for analysis. These reports were obtained from four of the six plants. The information in the surveillance capsule reports was used to normalize the neutron fluxes calculated by the methods customarily used for activation analysis of reactor internals. The normalized fluxes were then used to perform the activation calculations for the vessel internal components present in each plant-specific reactor model. Component-specific radionuclide content and distributions were calculated. Decommissioning GTCC LLW volumes were determined from drawings and data for the six plant-specific reactor models

analyzed. These values were then used to estimate the GTCC volumes that could be generated from the decommissioning of all LWRs based on 40-year operating cycles.

Section 2 presents the summary and conclusions for this study. The overall approach to the study and the LWR fuel cycle information used is presented in Section 3. Sections 4, 5, and 6 present the study results for routinely generated activated metals, process wastes, and decommissioning activated metal respectively. Section 7 presents the study references.

2. EVALUATION SUMMARY AND CONCLUSIONS

The generation of potential GTCC LLW from LWRs is both plant type and fuel cycle dependent. For this study, all LWRs, both operating and planned for startup, were considered. This included 113 LWRs composed of 36 BWRs and 77 PWRs.

Based on the results of a previous WMG study (Reference 1) three categories of potential GTCC LLW were considered:

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

This evaluation confirmed the results of the earlier WMG study, which indicated that GTCC LLW volumes would be much lower than previously estimated. Routinely generated activated metals and process wastes, in the form of cartridge filters, were found to represent a relatively small portion of the LLW routinely generated at LWRs. The estimated generation rates at BWRs ranged from a low of 3.5 ft³ to a high of 6.5 ft³ per plant per fuel cycle. The estimated generation rates at PWRs, which varied with reactor vendor, ranged from a low of 2.4 ft³ to a high of 3.5 ft³ per plant per fuel cycle. A summary of these estimated generation rates organized by waste type and reactor vendor is presented in Table F-1, along with the estimated total volumes of GTCC LLW that could be generated through plant closure. These cumulative volume estimates are based on the generation rates shown and the number of fuel cycles remaining through plant closure assuming 40-year plant life cycles. As shown, under the "no concentration averaging" or highest GTCC LLW volume scenario, the cumulative volumes are relatively small, amounting to about 6000 ft³ or 170 m³.

All indications are that full reactor coolant system (RCS) decontamination projects with fuel assemblies remaining in the vessel will result in GTCC LLW in the form of deep bed resins. Full RCS decontamination projects have never been performed and will not be until the mid-1990s. Initially, these operations will be performed with fuel out at a frequency of about once every 10 years per plant. After these decontamination processes have been fully demonstrated with fuel assemblies removed, plants will consider full RCS decontaminations with fuel assemblies in the vessel. To project RCS decontamination resin volumes, it was assumed that plants with less than 20 years remaining through closure would not perform full RCS decontaminations with fuel assemblies in the vessel. This assumption eliminated 43 plants--17 BWRs, and 26 PWRs--as potential generators of GTCC resin waste. The remaining 70 plants--19 BWRs, and 51 PWRs--were assumed to perform one full RCS decontamination project with fuel assemblies in the vessel before shutdown. The estimated volume of GTCC decontamination resin waste for a typical BWR was 100 ft³, and that for a PWR was 150 ft³. These estimates lead to cumulative volumes of about 9500 ft³ or about 270 m³ of decontamination resin for all LWRs through shutdown.

Table F-1. Summary of routine GTCC LLW

SUMMARY OF ROUTINE GTCC WASTES								
	Number Of Plants	Unit Volume (ft ³)	Units Per Cycle	Percent GTCC	GTCC Units Cycle	GTCC (ft ³) Cycle	Cycles To Shutdown	Total GTCC (ft ³)
ROUTINE ACTIVATED METALS								
1.BWR Operations	36							
a. Control Rod Blades								
- Small Normal	10	0.6	14	10	1	0.6	127	76
- Small Controlled Cell	7	0.6	13/11	25/10	4	2.4	85	204
- Large Normal	12	0.6	19	10	2	1.2	220	264
- Large Controlled Cell	7	0.6	19/15	25/10	6	3.6	131	473
b. LPRM Strings	36	0.1	12	75	9	0.9	564	507
c. IRM/SRM Dry Tubes	36	0.08	2	100	2	0.16	564	90
							Subtotal	1,614
'XX/YY' - Controlled Cell/Normal								
2. West. PWR Operations	55							
a. Thimble Plug Assemblies		0.024	16	10	1.6	0.0384	978	38
b. Thimble Tubes		0.06	2	100	2	0.12	978	117
c. Primary Source Rods		0.007	0.4	100	0.4	0.0028	978	3
							Subtotal	158
3. CE PWR Operations	15							
a. Instrument Strings		0.07	24	50	12	0.84	249	209
b. Primary Sources		0.02	0.4	100	0.4	0.008	249	2
							Subtotal	211
4. B&W PWR Operations	7							
a. In-Core Detectors		0.1	16	50	8	0.8	100	80
b. Primary Sources		0.002	0.4	100	0.4	0.0008	100	<1
								80
							Subtotal	2,063
ROUTINE PROCESS WASTES								
1.BWR Cartridge Filters	36							
a. Control Rod Drive		0.07	30	50	15	1.05	564	592
b. Fuel Pool/Vacuum		0.22	12	25	3	0.66	564	372
							Subtotal	964
2.PWR Cartridge Filters	77							
a. Reactor Coolant System		0.45	8	25	2	0.9	1,327	1,194
b. Seal Water Injection		0.1	16	25	4	0.4	1,327	530
c. Cavity Drain		0.1	2	25	0.5	0.05	1,327	66
d. Spent Fuel Pool		0.85	4	25	1	0.85	1,327	1,128
							Subtotal	2,918
3.B&W Crud Tank Filters	7	0.95	0.5	100	0.5	0.475	100	48
							Subtotal	3,930
							Grand Total	5,993

Decommissioning activated metals represent the largest source of GTCC LLW in terms of volume and activity. For this study, a detailed analysis was performed for six actual plants typical of LWRs. Two BWRs and four PWRs were analyzed in depth to conservatively estimate volumes and to realistically determine activities of potential GTCC components. Table F-2 summarizes the results of this analysis extended to all LWRs in terms of volumes. As shown, the cumulative volume of decommissioning activated metals for all plants considered is about 21,400 ft³ or 610 m³. These values are conservative, since only larger plants of each LWR type were evaluated in detail.

The study results for decommissioning activated metals were somewhat different than those previously reported in the standard NRC reference reports. For BWR components, the results of this study indicate that when calculated fluxes are normalized to surveillance capsule data, the calculated integrated fluxes are about 35% lower than those previously reported, resulting in lower activities. Thus, while the core shroud exceeds Class C limits, it does so by factors of less than 2. These results indicate that further investigations to better define BWR flux levels are warranted.

For PWR components, the results of this evaluation confirm that the core baffle or shroud is well above Class C limits for all PWRs considered and that the thermal shield is not GTCC LLW. Mixed results were found for the core barrel, a major PWR component. For the two Westinghouse PWRs, the core barrel was found to barely exceed Class C limits. Since these plants were considered to be representative of 55 PWRs, further evaluations could eliminate the core baffle as GTCC LLW in this group of PWRs.

For larger LWRs, this study showed that the use of surveillance capsule data to normalize activation fluxes eliminates some of the conservatism in earlier studies. Some of the older, smaller LWRs, with relatively long operating histories, were not analyzed. These older units should be analyzed to determine if the use of surveillance capsule data for plants with the higher integrated fluxes associated with longer operation further reduce activation flux estimates.

The methodology employed for the analysis of decommissioning activated metals relied on NUREG 3474² for materials compositions. These data have not been applicable to other activated metals components for some time. Activated materials heat data, or composites thereof, are typically used for activation analysis performed on routinely generated activated metal components. Similar current materials composition data obtained from actual component heats should be compiled for decommissioning components. These updated materials composition data could be used to eliminate some of the uncertainties in decommissioning component activation analysis and better define potential GTCC LLW volumes.

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

Table F-2. LWR decommissioning GTCC LLW summary

LWR DECOMMISSIONING GTCC WASTE SUMMARY					
	Number of Units	Core Shroud (ft³)	Core Baffle (ft³)	Core Barrel (ft³)	Total Volume (ft³)
BWR Plants					
1. Models 2/3	9	102	NA	NA	918
2. Model 4	13	102	NA	NA	1,326
3. Model 4/5	2	176	NA	NA	352
4. Model 6	12	176	NA	NA	2,112
Totals	36				4,708
PWR Plants					
1. B&W Reactors	7	NA	52	90	994
2. CE Reactors	15	NA	88	290	5,670
3. West. Moderate	28	NA	45	125	4,760
4. Westinghouse Large	27	NA	71	125	5,292
Totals	77				16,716
Grand Totals	113				21,424

3. TECHNICAL APPROACH

The study objective is to provide quantitative volumes and radioactivity amounts for the GTCC LLW that could be generated by LWRs over the period 1995 through 2025.

To achieve this objective, parameters that describe LWR plant performance and the impact of this performance on GTCC LLW generation were considered. These parameters are discussed below.

3.1 LWR Operations Parameters

At present, there are 36 General Electric BWR plants ranging in size from the 504 MWe Vermont Yankee plant to the 1205 MWe Perry plant. Table F-3 summarizes the overall plant parameters relevant to this study. Table F-4 illustrates similar information for PWR stations broken down by reactor vendor. As shown, there are 7 Babcock & Wilcox plants, 15 Combustion Engineering plants, and 55 Westinghouse plants. Three of the Westinghouse plants have not yet initiated operations. Since there is no indication that they will not be started up on the dates shown, they were included in the study.

The parameters shown on these tables include:

- Reactor Type--For BWRs: 2, 3, 4, or 6; for Westinghouse PWRs: 2, 3, or 4.
- Plant Name, Unit, and MWe.
- Startup Date--Actual (or expected) date of commercial operation.
- Closure Date--Assumed to be 40 years after commercial operation.
- Cycle Months--The latest published information on the length of the plants' fuel cycle.
- Outage Start and Finish Dates--The latest published information on outages in 1990, and those planned for 1991.
- Days Shut Down--The elapsed time between start and finish dates of the 1990-1991 outages.
- Capacity Factors--The latest published information on the plants, values for capacity factors for the periods 1984-1986 and 1987-1989. These factors are the effective full power days (EFPD) generated by the plant divided by the EFPD that would be generated if the plant had run at 100% power for 100% of the time.
- Best Capacity Factor--The highest capacity factor reported for the plant averaged over either the 1984-1986 or the 1987-1989 time period.

These operations parameters provided the overall framework for defining LWR fuel cycles and the basis for GTCC LLW generation rates.

Table F-3. BWR stations

BWR STATIONS													
#	Type	Plant Name	Unit	MWE Net	Startup Date	Closure Date	Cycle Mos	Outage Start	Outage Finish	Days SD	CF 84-86	CF 87-89	Best Factor
1	2	Nine Mile Point	1	610	Dec-69	Nov-2009		03/03/91	04/13/91	41	71.8	28.3	71.8
2	2	Oyster Creek	1	620	Dec-69	Nov-2009	21	02/15/91	08/01/91	41	31.2	52.9	52.9
3	3	Dresden	2	772	Jun-70	May-2010	15	09/01/90	11/15/90	75	58.4	59.3	59.3
4	3	Millstone	1	654	Mar-71	Feb-2011	21	03/30/91	05/10/91	41	81.6	83.8	83.8
5	3	Monticello		536	Jun-71	May-2011	19	04/01/91	05/27/91	56	55.2	75.0	75.0
6	3	Dresden	3	773	Nov-71	Oct-2011	17	03/30/91	06/08/91	70	43.1	65.5	65.5
7	4	Vermont Yankee		504	Nov-72	Oct-2012	16	09/01/90	10/13/90	42	62.0	83.3	83.3
8	3	Pilgrim		670	Dec-72	Nov-2012	20	05/04/91	07/13/91	70	34.7	9.9	34.7
9	3	Quad Cities	1	769	Feb-73	Jan-2013	14	10/27/90	01/06/91	71	66.7	69.4	69.4
10	3	Quad Cities	2	769	Mar-73	Feb-2013	16	09/14/91	11/23/91	70	68.7	71.1	71.1
11	4	Cooper		764	Jul-74	Jun-2014	15	10/06/91	11/30/91	55	42.0	70.9	70.9
12	4	Peach Bottom	2	1051	Jul-74	Jun-2014	18	01/12/91	03/20/91	67	41.6	19.3	41.6
13	4	Peach Bottom	3	1035	Dec-74	Nov-2014	18	08/31/91	11/08/91	69	55.6	5.9	55.6
14	4	Duane Arnold		538	Feb-75	Jan-2015	18	06/28/90	09/04/90	68	54.2	62.4	62.4
15	4	J.A. FitzPatrick		757	Jul-75	Jun-2015	24	04/01/91	04/10/91	9	70.0	68.5	70.0
16	4	Brunswick	2	790	Nov-75	Oct-2015	16	04/15/91	05/07/91	22	43.2	63.9	63.9
17	4	Edwin I. Hatch	1	753	Dec-75	Nov-2015	18	09/18/91	11/24/91	67	58.7	76.7	76.7
18	4	Brunswick	1	790	Mar-77	Feb-2017	14	09/27/90	03/14/91	168	59.8	58.7	59.8
19	4	Edwin I. Hatch	2	766	Sep-79	Aug-2019	18	03/20/91	05/24/91	65	52.7	68.6	68.6
20	4/5	Susquehanna	1	1032	Jun-83	May-2023	18	09/08/90	11/10/90	63	61.3	75.5	75.5
21	6	LaSalle County	1	1036	Jan-84	Dec-2023	18	02/16/91	04/27/91	70	42.4	55.3	55.3
22	6	LaSalle County	2	1036	Oct-84	Sep-2024							
23	6	WNP-2	2	1095	Dec-84	Nov-2024	10	05/22/91	07/10/91	49			
24	4/5	Susquehanna	2	1038	Feb-85	Jan-2025	18	03/09/91	05/10/91	62			
25	6	Grand Gulf		1142	Jul-85	Jun-2025	18	10/01/90	11/14/90	44			
26	6	Limerick	1	1055	Feb-86	Jan-2026		09/07/91	09/28/91	21			
27	6	River Bend		936	Jun-86	May-2026	16	09/15/90	11/14/90	60			
28	6	Hope Creek		1031	Dec-86	Nov-2026	17	12/26/90	02/13/91	49			
29	6	Clinton	1	930	Apr-87	Mar-2027	13	10/14/90	02/05/91	114			
30	6	Perry	1	1205	Nov-87	Oct-2027	18	09/07/90	11/24/90	78			
31	4	Fermi	2	1075	Jan-88	Dec-2027	13	03/16/91	05/29/91	74			
32	6	Nine Mile Point	2	1072	Apr-88	Mar-2028	15	09/05/90	01/15/91	132			
33	6	Limerick	2	1055	Jan-90	Dec-2029	24	03/23/91	06/08/91	77			
Averages				868			17.21			64.37	54.9952	58.2952	65.1

Table F-4. PWR stations

PWR STATIONS													
#	Type	Plant Name	Unit	MWE Net	Startup Date	Closure Date	Cycle Mos	Outage Start	Outage Finish	Days SD	CF 84-86	CF 87-89	Best Factor
B&W Plants													
1		Oconee	1	846	Jul-73	Jun-2013	13.9	07/25/91	09/18/91	55	77.2	77.8	77.8
2		Oconee	2	846	Sep-74	Aug-2014	14.2	12/12/91	02/05/92	55	77.8	76.2	77.8
3		Three Mile Island	1	808	Sep-74	Aug-2014	24.0	10/04/91	12/13/91	70	26.1	82.2	82.2
4		Arkansas Nuclear One	1	836	Dec-74	Nov-2014	14.0	10/01/90	12/07/90	67	59.8	54.1	59.8
5		Oconee	3	846	Dec-74	Nov-2014	14.6	02/13/91	03/23/91	38	69.8	74.5	74.5
6		Crystal River	3	821	Mar-77	Feb-2017		10/10/91	11/14/91	35	55.2	56.8	56.8
7		Davis-Besse		874	Jul-78	Jun-2018	15.9	09/01/91	11/09/91	69	26.2	56.9	56.9
Averages							16.1			55.5			69.4
CE Plants													
1		Palisades		768	Dec-71	Nov-2011	11.8	09/15/90	02/11/91	149	32.8	45.8	45.8
2		Maine Yankee		830	Dec-72	Nov-2012	18.0	10/05/91	11/30/91	56	77.1	73.7	77.1
3		Fort Calhoun		478	Sep-73	Aug-2013	18.0	09/28/91	01/08/92	102	71.6	71.5	71.6
4		Calvert Cliffs	1	825	May-75	Apr-2015		03/10/91	04/15/91	36	73.8	53.0	73.8
5		Millstone	2	863	Dec-75	Nov-2015	16.0	09/15/90	11/10/90	56	66.7	76.5	76.5
6		St. Lucie	1	839	Dec-76	Nov-2016	18.0	10/18/91	12/20/91	63	78.5	86.6	86.6
7		Calvert Cliffs	2	825	Apr-77	Mar-2017		10/04/91	11/17/91	44	80.8	58.0	80.8
8		Arkansas Nuclear One	2	858	Mar-80	Feb-2020	18.0	02/15/91	04/07/91	51	67.5	71.0	71.0
9		San Onofre	2	1070	Aug-83	Jul-2023	18.8	07/06/91	09/14/91	70	59.6	72.7	72.7
10		St. Lucie	2	839	Aug-83	Jul-2023	18.0	09/01/90	10/23/90	52	81.6	86.1	86.1
11		San Onofre	3	1080	Apr-84	Mar-2024							
12		Waterford	3	1075	Sep-85	Aug-2025	18.0	03/24/91	05/17/91	54			
13		Palo Verde	1	1221	Jan-86	Dec-2025		01/12/91	02/26/91	45			
14		Palo Verde	2	1221	Sep-86	Aug-2026	18.0	10/16/91	12/25/91	70			
15		Palo Verde	3	1221	Jan-88	Dec-2027	18.0	03/17/91	05/26/91	70			
Averages							17.3			65.5			74.2

Table F-4. (continued)

PWR STATIONS													
#	Type	Plant Name	Unit	MWE Net	Startup Date	Closure Date	Cycle Mos	Outage Start	Outage Finish	Days SD	CF 84-86	CF 87-89	Best Factor
West. Plants													
1		Yankee		167	Jul-61	Jun-2001	18.0	06/23/90	08/19/90	57	78.2	77.3	78.2
2	2	Haddam Neck		565	Jan-68	Dec-2007	14.0	10/05/91	11/15/91	41	66.2	57.4	66.2
3	2	San Onofre	1	436	Jan-68	Dec-2007	16.7	06/30/90	11/30/90	153	31.4	45.8	45.8
4	2	R. E. Ginna		470	Jul-70	Jun-2010	10.0	03/15/91	04/19/91	35	84.0	84.2	84.2
5	2	Point Beach	1	485	Dec-70	Nov-2010	10.9	04/05/91	05/17/91	42	78.2	84.2	84.2
6	2	Robinson	2	665	Mar-71	Feb-2011	15.0	09/09/90	01/10/91	123	55.5	55.3	55.5
7	2	Point Beach	2	485	Oct-72	Sep-2012	10.7	09/27/91	11/08/91	42	80.6	82.7	82.7
8	3	Surry	1	781	Dec-72	Nov-2012	14.0	09/06/91	09/20/91	14	64.8	50.6	64.8
9	2	Turkey Point	3	666	Dec-72	Nov-2012		12/01/90	09/21/91	294	69.7	43.3	69.7
10	2	Surry	2	781	May-73	Apr-2013	18.5	04/05/91	06/18/91	74	66.5	44.7	66.5
11	2	Turkey Point	4	666	Sep-73	Aug-2013	18.0	11/15/90	10/26/91	345	54.7	43.8	54.7
12	2	Prairie Island	1	503	Dec-73	Nov-2013	17.0	06/05/91	07/02/91	27	83.6	84.7	84.7
13	4	Zion	1	1040	Dec-73	Nov-2013	11.0	08/31/91	11/09/91	70	56.3	63.6	63.6
14	2	Kewaunee		503	Jun-74	May-2014	10.2	03/08/91	05/05/91	58	80.7	82.9	82.9
15	4	Indian Point	2	970	Aug-74	Jul-2014	22.0	01/12/91	05/27/91	135	58.2	68.3	68.3
16	3	Zion	2	1040	Sep-74	Aug-2014					67.4	71.1	71.1
17	2	Prairie Island	2	500	Dec-74	Nov-2014	17.0	09/12/90	10/10/90	28	81.6	87.6	87.6
18	4	Donald C. Cook	1	1020	Aug-75	Jul-2015	15.5	10/20/90	01/13/91	85	60.2	66.2	66.2
19	4	Trojan		1095	May-76	Apr-2016	9.8	03/27/91	06/05/91	70	63.0	54.6	63.0
20	3	Indian Point	3	965	Aug-76	Jul-2016		05/11/91	05/25/91	14	64.2	65.1	65.1
21	3	Beaver Valley	1	810	Oct-76	Sep-2016	16.1	04/12/91	06/21/91	70	70.2	65.6	70.2
22	4	Salem	1	1106	Jun-77	May-2017	12.0	02/09/91	03/30/91	49	63.5	67.3	67.3
23	3	Joseph M. Farley	1	824	Dec-77	Nov-2017	18.0	03/08/91	05/07/91	60	78.1	84.2	84.2
24	3	North Anna	1	915	Jun-78	May-2018	18.0	01/04/91	03/19/91	74	66.6	61.9	66.6
25	4	Donald C. Cook	2	1060	Jul-78	Jun-2018	15.5	06/30/90	09/13/90	75	53.2	48.4	53.2
26	3	North Anna	2	915	Dec-80	Nov-2020	18.0	09/07/90	11/21/90	75	73.6	81.5	81.5
27	3	Joseph M. Farley	2	828	Jul-81	Jun-2021	16.0	10/06/90	11/24/90	49	82.8	81.1	82.8
28	4	Sequoyah	1	1148	Jul-81	Jun-2021	18.0	10/15/91	12/15/91	61	33.5	31.8	33.5

Table F-4. (continued)

PWR STATIONS													
#	Type	Plant Name	Unit	MWE Net	Startup Date	Closure Date	Cycle Mos	Outage Start	Outage Finish	Days SD	CF 84-86	CF 87-89	Best Factor
West. Plants													
29	4	Salem	2	1106	Oct-81	Sep-2021	16.0	11/30/91	02/07/92	69	46.1	68.1	68.1
30	4	McGuire	1	1129	Dec-81	Nov-2021	13.0	09/06/91	11/23/91	78	59.6	72.6	72.6
31	4	Sequoyah	2	1148	Jun-82	May-2022	18.0	09/10/90	11/21/90	72	39.5	32.9	39.5
32	3	Virgil C. Summer		885	Jan-84	Dec-2023	16.3	09/13/91	11/22/91	70	70.0	66.0	70.0
33	4	McGuire	2	1129	Mar-84	Feb-2024	13.0	09/01/90	11/19/90	79			
34	4	Callaway		1125	Apr-85	Mar-2025	18.0	09/21/90	11/16/90	56			
35	4	Diablo Canyon	1	1073	May-85	Apr-2025	18.0	02/10/91	04/15/91	64			
36	4	Catawba	1	1129	Jun-85	May-2025	12.3	03/12/91	06/12/91	92			
37	4	Byron	1	1105	Sep-85	Aug-2025	15.0	09/07/91	11/16/91	70			
38	4	Wolf Creek		1135	Sep-85	Aug-2025	15.0	09/12/91	11/21/91	70			
39	4	Diablo Canyon	2	1087	Mar-86	Feb-2026	18.0	09/15/91	11/17/91	63			
40	4	Millstone	3	1142	Apr-86	Mar-2026	19.0	02/02/91	03/30/91	56			
41	4	Catawba	2	1129	Aug-86	Jul-2026	13.6	10/24/91	12/28/91	65			
42	3	Shearon Harris		860	May-87	Apr-2027	18.0	03/01/91	05/10/91	70			
43	4	Alvin W. Vogtle	1	1079	Jun-87	May-2027	18.0	09/13/91	10/28/91	45			
44	4	Byron	2	1105	Aug-87	Jul-2027	15.0	09/01/90	11/10/90	70			
45	3	Beaver Valley	2	833	Nov-87	Oct-2027	18.0	09/01/90	11/09/90	69			
46	4	Braidwood	1	1120	Jul-88	Jun-2028	16.0	03/16/91	05/25/91	70			
47	4	South Texas Project	1	1250	Aug-88	Jul-2028	18.0	04/01/91					
48	4	Braidwood	2	1120	Oct-88	Sep-2028	15.5	09/14/91	11/23/91	70			
49	4	Alvin W. Vogtle	2	1100	May-89	Apr-2029	16.1	09/14/90	11/03/90	50			
50	4	South Texas Project	2	1250	Jun-89	May-2029	12.0	11/01/90					
51	4	Seabrook		1150	Jul-90	Jun-2030	12.0	07/27/91	10/02/91	67			
52	4	Comanche Peak	1	1150	Aug-90	Jul-2030		04/13/91	05/24/91	41			
53	4	Watts Bar	1	1177	Jan-93	Dec-2032							
54	4	Comanche Peak	2	1150	Jan-93	Dec-2032							
55	4	Watts Bar	2	1177	Jan-95	Dec-2034							
Averages							14.9			72.1			64.5

3.2 Study Parameters

For BWRs, the waste types that could be GTCC under the no concentration averaging case include:

- Activated metals
 - Control rod blades
 - Local power range monitors
 - IRM/SRM dry tubes
- Process wastes
 - Cartridge filters
 - Full RCS decontamination resins
- Decommissioning activated metals
 - Core shroud
 - Upper fuel guide.

For PWRs, these waste types included:

- Activated metals
 - Thimble plug assemblies
 - Incore detectors and instrument strings
 - Primary sources
- Process wastes
 - Cartridge filters
 - Full RCS decontamination resins
- Decommissioning activated metals
 - Core baffle (shroud)
 - Core barrel.

Two of these wastes types, activated metals and cartridge filters, are routinely generated. Decontamination resins are generated intermittently. Decommissioning metals are generated at plant closure. The integration of LWR operations parameters and these GTCC LLW generation mechanisms are discussed below.

3.2.1 Routinely Generated Potential GTCC LLW

Routine GTCC LLW type generation is largely dependent on individual plant operation. The information shown on Tables F-3 and F-4 was expanded to define the following parameters relevant to potential GTCC LLW generation.

- Fuel Cycles--To determine this parameter for BWRs, the latest information in Table F-3 for 33 of the 36 plants was compiled and reviewed. The three Browns Ferry plants were excluded since they have not operated for six years.

This review indicated that the average BWR fuel cycle was 17.4 months, with 64 days of downtime between cycles for refueling. A 20-month cycle, which includes three months for refueling, was assumed.

As shown in Table F-4, fuel cycle lengths varied among PWR reactor vendors. For PWRs, two different cycles were used: a 19-month cycle for the Babcock & Wilcox and Westinghouse plants and a 20-month cycle for the Combustion Engineering plants. Both of these cycles include three months for refueling.

- **Exposure Per Cycle**--For activated metals, the EFPD per cycle determines a components exposure and its radioactivity. The average of the best capacity factors reported was used in conjunction with a three-month downtime between fuel cycles to define an average activation factor. This factor was used with the average power level for all plants to define the average EFPD of activated metal exposure per fuel cycle.

For BWRs, the average cycle exposure is 390 EFPD, which corresponds to a 17-month operating cycle at 76% capacity. Two different cycle exposures were used for PWRs: 400 EFPD for Babcock & Wilcox and Westinghouse plants, which corresponds to a 16-month operating cycle at 83% capacity; and 430 EFPD for Combustion Engineering plants, which corresponds to a 17-month operating cycle at 83% capacity.

- **Cycles to Closure**--Most plants do not have GTCC LLW under current criteria. After January 1, 1993, compact criteria can lead to GTCC LLW. The volumes that could be generated between this date and closure will be operating cycle dependent. The number of plant-specific fuel cycles to closure was determined for each plant by dividing the time between January 1, 1993, and the closure date by the average fuel cycle lengths. 20 months was used for BWRs. Two different cycle lengths were used for PWRs: 19 months for Westinghouse and Babcock & Wilcox plants and 20 months for Combustion Engineering plants.

These parameters were used to estimate routinely generated GTCC volumes and activities as described in Sections 4 and 5.

3.2.2 Decommissioning Potential GTCC LLW

Decommissioning GTCC LLW consists exclusively of activated metals, and generation is dependent on the integrated flux to which vessel internals are exposed during the plant's operating lifetime. The variable parameters that define integrated exposure are the plant's power level and capacity factor. The fixed parameter used for this study is a 40-year operating lifetime.

For BWRs, the average for the best capacity factors, 65%, was used in conjunction with the 40-year lifetime to estimate the radionuclide activities at the time of BWR decommissioning. For PWRs, the average for the best capacity factors varied among vendors. As shown in Table F-4, capacity was about 70% for Babcock & Wilcox, 74% for Combustion Engineering, and 68% for Westinghouse. For this study, a weighted overall average factor of 70% was used in conjunction with the 40-year lifetime to estimate the radionuclide activities at the time of PWR decommissioning. These parameters were used to estimate decommissioning GTCC volumes and activities as described in Section 6.

4. ACTIVATED METALS GENERATION AND ACTIVITY

Activated metals consist of the nonfuel-bearing components that are activated through exposure to neutron flux within light water, research, and test reactors. Two sources were considered: those components that are periodically replaced during normal operations, and those structural components in the reactor vessel that are removed at decommissioning.

The generation of activated metals depends on the plant's type, size, fuel cycle length, and average exposure during each fuel cycle. After removal from the core, these components are usually allowed to accumulate in the spent fuel pool until sufficient quantities are available for shipment. While the quantities that could be GTCC are a fraction of the volume discharged, it was assumed that current practices would continue in the post-1993 time frame. Thus, activities of components were estimated after two cycles of decay in the spent fuel pool. This decay does not affect 10 CFR Part 61³ classification status. It merely allows for the decay of the short-lived activation product radionuclides Cr-51, Mn-54, Fe-55, and Co-60. Since Fe-55 and Co-60 are the dominant radionuclides in activated metals, their decay time significantly reduces total activity.

This section discusses routinely generated components. It presents the estimated GTCC LLW volumes and curie content for activated metal components from BWRs and PWRs based on the operating cycles discussed in Section 3. Section 6 discusses decommissioning components.

4.1 BWR Activated Metals

Current data indicate that the typical BWR operates on a 20-month cycle, with three months allowed for refueling. During the 17 months of power operation, the average plant operates at 76% of full power.

These average conditions were compared to the actual operating cycle histories of ten representative BWRs, which operated for between 4 and 12 fuel cycles. Average power operating cycle days ranged from 380 to 560 days. Approximately 480 days (16 months) was the average for all plants. Since this average included unplanned outages, the 17-month typical fuel cycle assumption is reasonable. The average integrated exposure during these cycles was 360 EFPD, which represents an activation factor of 75% (360 EFPD/480 days). This compares very favorably to the overall average of 76% for all BWRs determined in Section 3. For conservative purposes, it was assumed that components received 390 EFPD exposure per fuel cycle, which is based on a 17-month cycle at 76% of rated capacity. It was also assumed that these components would not be available for shipment until 40 months after discharge.

Component activities were determined using the methods employed by WMG to characterize actual BWR components for disposal. The materials compositions used are based on composition data from manufacturer heat data of material used to manufacture the components considered.

BWR activated metal components that can exceed Class C limits are discussed below.

4.1.1 Control Rod Blades

Each core contains from 137 to 185 control rod blades (CRBs), which are distributed radially throughout the core. The CRB volume was assumed to be 0.6 ft^3 .

Control rod operating patterns vary by station. Most plants operate in a conventional control rod pattern, where alternating banks of deep and shallow CRBs are slowly withdrawn from the core. Some plants also run in what is referred to as the controlled cell mode. In this mode, CRB movement is restricted to a fixed group of 25 in a 5×5 array (137 CRB cores) to 37 CRBs in a 5×5 array surrounded by a 3×3 array (185 CRB cores). CRBs in the controlled cell mode are fully inserted while the rest of the CRBs are completely withdrawn.

Routine CRB discharges are exposure dependent. For BWRs not running in the controlled cell mode, historical data indicate that CRBs are not discharged until after the sixth or seventh fuel cycle. For BWRs running in the controlled cell mode, half of the 25 to 37 CRBs that operated in this mode are discharged each fuel cycle.

CRB generation rates depend on the core size (137 vs 185 CRBs) and the mode of CRB operation. Consistent with the earlier study findings, 10% of normal CRBs and 25% of controlled cell CRBs were assumed to exceed Class C limits.

The 36 BWRs considered were divided into four categories, based on plant size and mode of CRB operation. Table F-5 summarizes the CRB generation rates for each category and presents the estimated GTCC CRB waste volumes per plant per fuel cycle.

For plants run in the conventional mode of operation, curie estimates were based on blades with six fuel cycles of operation, or 2340 EFPD, of exposure. These CRBs contained 2570 Ci and were 1.4 times greater than Class C limits.

For plants run in the controlled cell mode, estimates were based on two cycles, or 780 EFPD, of exposure. These CRBs contained 3420 Ci and were 1.3 times greater than Class C limits.

The above curie estimates exclude the four upper stellite bearings (rollers) on a CRB that become highly activated. Stellite bearing volumes range from 0.002 to 0.003 ft^3 per CRB, dependent on the removal method. The activity of each set of four bearings ranges from 300 to 800 Ci. Alone, stellite bearings are 3 to 5 times greater than Class C limits. Most of the plants that have stellite bearings on their CRBs are gradually replacing them with nonstellite bearing CRBs. This replacement should be completed over the next five years, so stellites should not be a concern for the GTCC LLW project.

The above generation rates and curie contents assume that CRBs which are GTCC LLW will remain intact. This is not consistent with current practice. The CRB velocity limiter sections are almost always removed and, in many cases, the cruciform sections are sectioned prior to shipment. Where plants encounter GTCC CRBs, customary practices would be employed to reduce GTCC LLW volumes. This would, as a minimum, entail removal of the velocity limiter, reducing CRB metal volume from 0.6 to 0.53 ft^3 . If CRBs had to be processed for shipment, those axial sections that met Class C limits would also be segregated from the GTCC LLW portion of the CRB. For those CRBs that run in the normal mode of operation, the 0.53 ft^3 of the cruciform could be reduced by half, to about 0.26 ft^3 . For those CRBs that run in the controlled cell mode, the cruciform volume could be reduced by 25%, to about 0.40 ft^3 .

Table F-5. Control rod blade generation

CONTROL ROD BLADE GENERATION				
Plant Size	Moderate	Moderate	Large	Large
Mode of Operation	Normal CRB	Controlled Cell CRB	Normal CRB	Controlled Cell CRB
1. Number of Plants	10	7	12	7
2. Number of CRB's	137	137	185	185
3. Number of CC CRB's	0	25	0	37
4. Normal CRB / Cycle	14	11	19	15
5. Percent Normal GTCC	10	10	10	10
6. CC CRB / Cycle	NA	13	NA	19
7. Percent CC GTCC	NA	25	NA	25
8. Total Cubic Feet GTCC/Cycle	0.6	2.4	1.2	3.6

4.1.2 Local Power Range Monitor Strings

Each core contains 31 local power range monitor (LPRM) strings. The metal volume of an intact LPRM string is 0.1 ft^3 and the weight is about 50 lb. When sectioned, the 13.5 ft hot end has a waste volume of 0.017 ft^3 and weighs 8.5 lb. The cold end has a waste volume of 0.083 ft^3 and weighs 41 lb.

LPRM strings are routinely discharged based on fission chamber depletion or when strings malfunction. Historic data indicate that 12 LPRM strings are discharged each fuel cycle. 75% of the LPRMs discharged were assumed to be GTCC LLW, for a rate of 0.9 ft^3 per plant per fuel cycle.

To estimate typical activities, it was assumed that the LPRMs received three cycles, or 1,170 EFPD, of exposure. These LPRMs contained 930 Ci and were 1.6 times greater than Class C limits.

Over 99% of the activity in the LPRM strings is contained in the hot end. Removal and shipment of those cold end sections that readily meet Class C limits would substantially reduce generation rates. For LPRM strings, the generation rate would be reduced from 0.9 ft^3 to about 0.15 ft^3 per plant per cycle. The hot end section would still contain about 930 Ci. Having essentially the same activity in the reduced volume would result in the hot end exceeding Class C limits by a factor of 9.4.

4.1.3 IRM/SRM Dry Tubes

Each core contains four source range monitor (SRM) dry tubes and eight intermediate range monitor (IRM) dry tubes. The metal volume of an intact dry tube is 0.08 ft^3 , and the weight is about 39 lb. When sectioned, the 13.5 ft hot end has a waste volume of 0.001 ft^3 and weighs 5 lb. The cold end has a waste volume of 0.007 ft^3 and weighs 34 lb.

Dry tubes are not routinely discharged. They are replaced only when damaged or embrittled. Historic data indicate that all 12 dry tubes are replaced at intervals of five to six fuel cycles. An average rate of 2 units or 0.16 ft^3 per plant per fuel cycle was assumed.

To estimate typical activities, it was assumed that the dry tubes received six cycles, or 2340 EFPD, of exposure. These dry tubes contained 1690 Ci and were 4.6 times greater than Class C limits.

Over 99% of the activity in the dry tubes is contained in the hot end. The removal and shipment of those cold end sections that readily meet Class C limits would substantially reduce generation rates. For dry tubes, the generation rate would be reduced from 0.16 ft^3 to about 0.02 ft^3 per plant per cycle. The hot end section would still contain about 1,630 Ci. Having essentially the same activity in the reduced volume would result in the hot end exceeding Class C limits by a factor of 37.

4.1.4 Radionuclide Content and Distribution

Table F-6 presents the estimated activities of the major radionuclides for these components at the time of shipment, and their status relative to Class C limits. The activities are based on 40 months of decay after discharge. For LPRMs and IRM/SRM dry tubes, the classification status is shown for the entire component as well as for the hot end by itself.

Table F-6. BWR hardware components

BWR HARDWARE COMPONENTS				
Activity Estimates				
1. Component Type	Control Rod Blade	Control Rod Blade (CC)	LPRM String	IRM/SRM Dry Tube
2. Volume (ft³)	0.6	0.6	0.1	0.08
3. Weight (lbs)	205	205	50	39
4. Number of Cycles	6	2	3	6
5. Exposure – EFPD (390 EFPD/Cycle)	2,340	780	1,170	2,340
6. Part 61 Status				
– Table 1 Fraction	1.36	1.28	1.57	4.61
– Table 2 Fraction	0.71	0.68	1.37	4.01
– Hot End Table 1	–	–	9.24	36.88
– Hot End Table 2	–	–	8.06	32.08
7. Activity (Curies)				
H-3	2.20E+01	2.30E+01	NA	NA
C-14	1.20E-01	1.10E-01	3.40E-02	8.10E-02
Mn-54	2.00E+01	4.50E+01	8.80E+01	1.20E+01
Fe-55	1.21E+03	1.78E+03	5.30E+02	9.05E+02
Co-60	1.23E+03	1.49E+03	3.60E+02	7.15E+02
Ni-59	5.30E-01	5.50E-01	1.80E-01	3.90E-01
Ni-63	8.50E+01	8.20E+01	2.70E+01	6.40E+01
Nb-94	3.75E-03	3.50E-03	6.24E-04	1.50E-03
TRU	NA	NA	3.00E-05	NA
Total Curies	2.57E+03	3.42E+03	9.30E+02	1.69E+03

4.2 PWR Activated Metals

As discussed in Section 3, current data indicate that typical PWRs operate on 19- to 20-month cycles, with three months allowed for refueling. The 20-month cycle is representative of the Combustion Engineering plants. The average plant operates at 83% of full power during the 16 to 17 months of operation.

These average conditions were compared to actual operating cycle histories for several PWRs from each manufacturer and were found to be reasonably consistent.

For PWRs, the activities of activated components vary with the manufacturer, and these are discussed separately below. It was assumed that Combustion Engineering plant components received 430 EFPD exposure per fuel cycle, and other PWRs received 400 EFPD exposure per fuel cycle. Like BWRs, activities of components were estimated after two cycles of decay in the spent fuel pool. It was assumed that Westinghouse and Babcock & Wilcox plants had 38 months decay time, and that Combustion Engineering plants had 40 months decay time.

Component activities were determined using the methods employed by WMG to characterize actual PWR components for disposal. The materials compositions used are based on actual data furnished by clients for disposal projects.

The basis for projecting volumes for potential GTCC LLW from routine PWR operations is presented below.

4.2.1 Thimble Plug Assemblies (Westinghouse)

For Westinghouse plants that have discharged thimble plug assemblies in the last five years, the average discharge rate is 16 units per plant per fuel cycle. Since only a small portion of these are GTCC, the generation rate is 1.6 units, or 0.04 ft³ per plant per fuel cycle.

To estimate activities, it was assumed that the units received five fuel cycles of exposure (2000 EFPD). These units contained 150 Ci and were 1.3 times greater than Class C limits.

4.2.2 Incore Detectors (Babcock & Wilcox)

Babcock & Wilcox plants use rhodium emitters with inconel lead wires and stainless steel sheaths. The entire detector has a metal volume of about 0.1 ft³. The 12 to 14 ft hot end section has a metal volume of about 0.013 ft³.

For those plants that have disposed of incore detectors, the average rate has been 16 detectors per plant per fuel cycle. Since 50% are GTCC, the generation rate is 8 detectors, or 0.8 ft³ per plant per fuel cycle.

To estimate activities, it was assumed that the detectors received three fuel cycles, or 1200 EFPD, of exposure. These detectors contained 360 Ci and were 2 times greater than Class C limits.

Over 99% of the activity is contained in the hot end section. The removal and shipment of those cold end sections that readily meet Class C limits would substantially reduce generation rates. For these incore detectors, the generation rate would be reduced from 0.8 ft³ to about 0.1 ft³ per

plant per cycle. The hot end section would still contain about 360 Ci. Having essentially the same activity in the reduced volume would result in the hot end exceeding Class C limits by a factor of 16.

4.2.3 Instrument Strings (Combustion Engineering)

Combustion Engineering plants use between 28 and 61 incore instrument assemblies. These instrument strings use rhodium emitters attached to inconel lead wires. The sheathing material is either inconel or stainless steel; the inconel-sheathed instrument strings are the worst case due to high nickel content. Both types have a total length of 35 to 116 feet and weigh between 7 and 34 lb. The typical volume is 0.07 ft^3 . The 12 to 14 ft hot end section has a metal volume of about 0.008 ft^3 .

For the past five years, these instrument strings have been changed out at an average rate of 24 units per plant per fuel cycle. Since 50% of these are GTCC, the generation rate is 12 instrument strings, or 0.84 ft^3 per plant per fuel cycle.

To estimate activities, it was assumed that the instrument strings received three fuel cycles, or 1290 EFPD, of exposure. These instrument strings contained 385 Ci and were 4 times greater than Class C limits.

Over 99% of the activity is contained in the hot end. The removal and shipment of those cold end sections that readily meet Class C limits would substantially reduce generation rates. For these instrument strings, the generation rate would be reduced from 0.84 ft^3 to about 0.1 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 exceeding Class C limits by a factor of 35.

4.2.4 Incore Detectors and Thimble Tubes (Westinghouse)

Each Westinghouse plant uses fixed and movable incore detectors, which contain fission chambers housed in a stainless steel sheath. These are inserted into about 50 thimble tubes located throughout the core. The thimble tube has a volume of 0.06 ft^3 , and the hot end section has a volume of about 0.007 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 all of these are GTCC, the generation rate is 2 units, or 0.12 ft^3 per plant per fuel cycle.

To estimate activities, it was assumed that the thimble tubes received seven fuel cycles, or 2800 EFPD, of exposure. These thimble tubes contained 1,060 Ci and were 4.3 times greater than Class C limits.

Most of the activity in the thimble tubes is contained in the hot end section. The removal and shipment of those cold end sections that readily meet Class C limits would substantially reduce generation rates. For thimble tubes, the generation rate would be reduced from 0.12 ft^3 to about 0.015 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 exceeding Class C limits by a factor of 37.

4.2.5 Primary Sources

Primary sources greatly exceed Class C limits, and current practices are to accumulate them in the spent fuel pool. When DOE provides for disposal of GTCC LLW, the sources that have accumulated will be shipped with other GTCC LLW.

Babcock & Wilcox plants use two primary neutron sources that are stainless steel-clad americium beryllium. The waste volume of each unit is 0.02 ft³.

Combustion Engineering plants also use two primary neutron sources. The source assemblies are 99 to 117 in. long and weigh between 4.5 and 10.9 lb. The source assemblies are 316 stainless steel with inconel springs. The waste volume is 0.02 ft³ per unit. Each source assembly contains one antimony-beryllium and two plutonium-beryllium sources.

Westinghouse plants use two primary source rods in assemblies that also contain burnable poison rods. The primary source assemblies have a total weight of 19 to 52 lb. The primary source rods themselves only weigh 2.7 to 3.6 lb., and are 110 to 152 in. long. The transuranic source materials are either californium, or plutonium-beryllium. It was assumed that the source rods are separated from the assembly and that the two rods have a waste volume of 0.007 ft³.

Sources are typically expended over five fuel cycles. Since all units are GTCC, the generation rate is 0.4 units per plant per fuel cycle. GTCC volumes by plant type per fuel cycle are as follows:

<u>Reactor Vendor</u>	<u>GTCC Ft³ Per Cycle</u>
Babcock & Wilcox	0.0008
Combustion Engineering	0.008
Westinghouse	0.0028

Source strengths vary by vendor. Typical activities are 50 (Pu-Be) Ci for Westinghouse plants, 340 (Pu-Be) Ci for Combustion plants, and 25 (Am-Be) Ci for Babcock & Wilcox plants.

4.2.6 Radionuclide Content and Distribution

Table F-7 presents the estimated activities of the major radionuclides for these components at the time of shipment, and their status relative to Class C limits. The activities shown include two fuel cycles of decay after discharge. For incore detectors, instrument strings and thimble tubes, the classification status is shown for the entire component, as well as for the hot end by itself.

4.3 Activated Metals Summary

Table F-8 summarizes the generation rates for BWR and PWR activated metals components. Two sets of values are presented: a maximum set that addresses intact components, and a minimum set that assumes that plants will separate individual components to isolate those portions that are GTCC LLW from those that could be routinely disposed of with other metals.

This table also shows the projected cumulative volumes of GTCC LLW generated from January 1993 through plant closure. These cumulative values are the generation rates per cycle by reactor vendor times the number of fuel cycles remaining before closure for each plant considered. As shown, the impact of component separation practices can be significant. It reduces total activated metal volumes from about 2,060 ft³ or 58 m³ to about 740 ft³ or 21 m³.

Table F-7. PWR hardware components

PWR HARDWARE COMPONENTS							
Activity Estimates							
PWR Vendor	West.	West.	West.	CE	CE	B&W	B&W
1. Component Type	Thimble Plugs	Thimble Tubes	Primary Source Rods	Instrument Strings	Primary Sources	In-Core Detectors	Primary Sources
2. Volume (ft ³)	0.024	0.06	0.007	0.07	0.02	0.1	0.002
3. Weight (lbs)	12	30	3.5	34	10	50	1
4. Number of Cycles	5	7	5	3	5	3	5
5. Exposure - EFPD	2,000	2,800	2,000	1,290	2,150	1,200	2,000
6. Part 61 Status							
- Table 1 Fraction	1.04	3.33	300,000	3.25	750,000	0.822	550,000
- Table 2 Fraction	1.28	4.29	NEG	4.05	NEG	2.059	NEG
7. Activity (Curies)							
C-14	7.80E-03	5.70E-02	NA	1.83E-02	NA	1.56E-02	NA
Mn-54	1.65E+00	8.40E+00	NA	2.08E-01	NA	1.89E-01	NA
Fe-55	8.85E+01	4.90E+02	NA	9.52E+00	NA	8.41E+00	NA
Co-60	5.45E+01	5.10E+02	NA	3.35E+02	NA	3.11E+02	NA
Ni-59	4.40E-02	4.20E-01	NA	3.26E-01	NA	3.33E-01	NA
Ni-63	5.30E+00	5.10E+01	NA	3.99E+01	NA	4.08E+01	NA
Nb-94	6.30E-05	5.80E-05	NA	1.34E-04	NA	1.24E-04	NA
TRU	NA	NA	50(Pu-Ba)	NA	340(Pu-Ba)	NA	25(Am-Ba)
Total Curies	1.50E+02	1.06E+03	5.00E+01	3.85E+02	3.40E+02	3.61E+02	2.50E+01

Table F-8. Summary of activated metals--GTCC LLW

SUMMARY OF ACTIVATED METALS GTCC WASTES								
	Number Of Plants	Unit Volume (ft ³)	Units Per Cycle	Percent GTCC	GTCC Units Cycle	GTCC (ft ³) Cycle	Cycles To Shutdown	Total GTCC (ft ³)
MAXIMUM ESTIMATES								
1. BWR Operations	36							
a. Control Rod Blades								
- Small Normal	10	0.6	14	10	1	0.6	127	76
- Small Controlled Cell	7	0.6	13/11	25/10	4	2.4	85	204
- Large Normal	12	0.6	19	10	2	1.2	220	264
	7	0.6	19/15	25/10	6	3.6	131	472
b. LPRM Strings	36	0.1	12	75	9	0.9	564	508
c. IRM/SRM Dry Tubes	36	0.08	2	100	2	0.16	564	90
							Subtotal	1,614
'XX/YY' - Controlled Cell/Normal								
2. West. PWR Operations	55							
a. Thimble Plug Assemblies		0.024	16	10	1.6	0.0384	978	38
b. Thimble Tubes		0.06	2	100	2	0.12	978	117
c. Primary Source Rods		0.007	0.4	100	0.4	0.0028	978	3
							Subtotal	158
3. CE PWR Operations	15							
a. Instrument Strings		0.07	24	50	12	0.84	249	209
b. Primary Sources		0.02	0.4	100	0.4	0.008	249	2
							Subtotal	211
4. B&W PWR Operations	7							
a. In-Core Detectors		0.1	16	50	8	0.8	100	80
b. Primary Sources		0.002	0.4	100	0.4	0.0008	100	<1
							Subtotal	80
							Maximum Totals	2,063
MINIMUM ESTIMATES								
1. BWR Operations	36							
a. Control Rod Blades								
- Small Normal	10	0.26	14	10	1	0.26	127	33
- Small Controlled Cell	7	0.37	13/11	25/10	4	1.48	85	126
- Large Normal	12	0.26	19	10	2	0.52	220	114
- Large Controlled Cell	7	0.37	19/15	25/10	6	2.22	131	290
b. LPRM Strings	36	0.017	12	75	9	0.153	564	86
c. IRM/SRM Dry Tubes	36	0.012	2	100	2	0.024	564	14
							Subtotal	663
2. Westinghouse PWR Operations	55							
a. Thimble Plug Assemblies		0.024	16	10	1.6	0.0384	978	38
b. Thimble Tubes		0.007	2	100	2	0.014	978	14
c. Primary Source Rods		0.007	0.4	100	0.4	0.0028	978	3
							Subtotal	55
3. CE PWR Operations	15							
a. Instrument Strings		0.008	24	50	12	0.096	138	13
b. Primary Sources		0.02	0.4	100	0.4	0.008	138	1
							Subtotal	14
4. B&W PWR Operations	7							
a. In-Core Detectors		0.013	16	50	8	0.104	100	10
b. Primary Sources		0.002	0.4	100	0.4	0.0008	100	<1
							Subtotal	10
							Minimum Totals	742

5. PROCESS WASTE GENERATION AND ACTIVITY

Process wastes consist of the wet wastes that are generated from the cleanup of liquids containing soluble and insoluble radioactive constituents. Potential GTCC LLW forms consist of cartridge filters and the cation bead resin produced during full reactor coolant system (RCS) decontamination. Filters are generated routinely, while decontamination resins are generated infrequently. These wastes are usually packaged and shipped shortly after their generation.

The activities of these waste types can best be estimated by reviewing empirical data for the waste types themselves. These empirical data consist of typical radionuclide content and distribution for the waste type and radiation levels. Decontamination of a complete RCS has not yet been performed, and actual data for these decontamination resins were not available. Data for smaller decontamination projects at both BWRs and PWRs were used in conjunction with several studies related to full RCS decontaminations.

5.1 Cartridge Filters

These process wastes are generated during cleanup of contaminated liquids. They represent relatively small unit waste volumes, which typically range from less than 0.1 ft³ to 1 ft³. In certain applications, when cartridge filters are changed out on pressure differential, filter radiation levels can range from 50 to 200 rem/hr. At these radiation levels, some filters are potential GTCC LLW. Changes in plant practices to remove filters from service at preset radiation levels would essentially eliminate all filters, except BWR control rod driveline (CRD) strainers, as GTCC LLW. Table F-9 illustrates the characteristics of filter cartridges that are potential GTCC LLW. These filters are discussed below.

5.1.1 BWR Operations

BWRs routinely generate control rod drive strainers and cartridge filters that are potential GTCC LLW.

- Control Rod Drive Strainers--Each control rod drive contains three metallic strainers. The two strainers at the spud end of the drive can exceed Class C limits. These strainers have volumes of 0.06 ft³ and 0.006 ft³ respectively. About 50% of the strainers generated from this source exceed Class C limits with no concentration averaging.

Control rod drive strainers are replaced when control rod drives are refurbished or replaced. About 15% of the control rod drives in a plant are refurbished during each refueling outage. Drives are replaced infrequently. For projection purposes, it was assumed all control rod drives would be replaced over 6 fuel cycles (ten years). This corresponds to a rate of about 1 ft³ per plant per cycle.

Strainer activities were estimated from typical direct sample data on strainer crud and strainer radiation levels from a representative plant. Typically, these filters can 150 rem/hr on contact, contain several curies, and are 1.4 times greater than Class C limits.

- Fuel Pool/Vacuum Filters--The typical filter is 6 in. in diameter and 30 to 36 in. long. This corresponds to a waste volume of about 0.22 ft³.

Table F-9. Cartridge filter summary

CARTRIDGE FILTER SUMMARY				
	Diameter (in.)	Length (in.)	Waste Volume (ft ³)	Weight (lbs)
BWR FILTERS				
a. Control Rod Drive (Inner)	2.1	3	0.006	0.5
b. Control Rod Drive (Outer)	3.75	9	0.06	2.5
c. Spent Fuel Pool / Vacuum	6	30	0.22	6
PWR FILTERS				
a. Reactor Coolant System	6.25	7.5	0.45	9
b. Seal Water Injection	2.75	20	0.1	2.5
c. Reactor Cavity Drain	2	30	0.1	4
d. Spent Fuel Pool	10	16	0.85	20
e. Crud Tank (B&W only)	10	17	0.95	40

These filters are usually generated during refueling outages and arise from vacuuming the spent fuel pool and the reactor cavity. During a typical outage, a dozen filters can be generated. About 25% of these, or three units (0.7 ft^3) per plant per fuel cycle, can be GTCC without concentration averaging.

Data were reviewed for two different plants where several particularly hot filters were seen. The worst case sample data set in terms of 10 CFR Part 61 classification was used to estimate activities for this filter type. Typically, these filters register over 100 rem/hr on contact, contain about 4 Ci, and are about 1.5 times greater than Class C limits.

5.1.2 PWR Operations

PWRs generate a wide range of filter types, some of which can be GTCC. Generation rates are dependent on filter type and plant practices. The rates considered for this study are typical.

- Reactor Coolant Filters--These are the highest activity filters generated by a PWR. The typical filter is 6.25 in. in diameter and 17.5 in. long. This corresponds to a volume of about 0.45 ft^3 . These filters are usually generated at an average rate of about 8 units per fuel cycle. About 25% of these, or two units per fuel cycle, can be GTCC without concentration averaging. This corresponds to a generation rate of 0.9 ft^3 per plant per fuel cycle.

Data from a single plant with filter radiation levels in the 200 rem/hr range were used to estimate activity. At these radiation levels, a typical filter contains about 4 Ci and is about 4.3 times greater than Class C limits.

- Seal Water Injection Filters--These are the second highest activity filters generated by a PWR. The typical filter is 20 in. in diameter and 2.75 in. long. This corresponds to a volume of about 0.1 ft^3 . These filters are usually generated at a rate of about 16 units per fuel cycle. About 25% of these, or two units per fuel cycle, can be GTCC without concentration averaging. This corresponds to a generation rate of 0.4 ft^3 per plant per fuel cycle.

Data from a single plant with filter radiation levels in the 50 rem/hr range were used to estimate activity. At these radiation levels, a typical filter contains about 1 Ci and is about 2.9 times greater than Class C limits.

- Cavity Drain Filters--These are the usually relatively low activity filters, except in those infrequent cases when the cavity is decontaminated. The typical filter is 2 in. in diameter and 30 in. long. This corresponds to a volume of about 0.1 ft^3 . These filters are generated at a rate of about 2 units per fuel cycle. About 25% of these, or 0.5 units per fuel cycle, can be GTCC without concentration averaging. This corresponds to a generation rate of 0.05 ft^3 per plant per fuel cycle.

Data from a single plant with filter radiation levels in the 50 rem/hr range were used to estimate activity. At these radiation levels, a typical filter contains about 1 Ci and is about 2.3 times greater than Class C limits.

- Fuel Pool Filters--The typical filter is 10 in. in diameter and 16 in. long. This corresponds to a volume of about 0.85 ft^3 . These filters are usually generated during

refueling outages, and arise from vacuuming the spent fuel pool and reactor cavity. During a typical outage, four such filters can be generated. About 25% of these, or 1 unit per fuel cycle, can be GTCC without concentration averaging. This corresponds to a generation rate of 0.85 ft^3 per plant per fuel cycle.

Data from several plants were reviewed with filter radiation levels in the 100 rem/hr range. At these radiation levels, a typical filter contains about 3 Ci and is about 1.6 times greater than Class C limits.

- **Crud Tank Filter**--These are the highest activity filters in all PWRs and are generated only at Babcock & Wilcox plants. They collect the media backflushed from etched disk filters. The typical collection filter is 10 in. in diameter and 17 in. long. This corresponds to a volume of about 0.95 ft^3 . These filters are usually generated at a rate of about 1 unit every two fuel cycles. Since the filters always exceed Class C limits without concentration averaging, the generation rate is 0.48 ft^3 per plant per fuel cycle.

Actual data for these filters were not readily available.

5.1.3 Radionuclide Content and Distribution

Cartridge filters represent very low volume sources of potential GTCC LLW and exceed Class C limits because the activity is distributed over a relatively small volume. Table F-10 presents a summary of potential GTCC cartridge filters. As shown, radiation levels are relatively high and, in all cases, Class C limits are exceeded due to transuranic concentrations.

5.2 Full RCS Decontamination Resins

Two decontamination processes have been employed to date at BWRs, which are referred to as the LOMI and CANDECON processes. Although the primary systems differ, the same two decontamination processes have also been employed at PWRs. A third process, referred to as CANDEREM, is also being evaluated for the full RCS system decontamination at PWRs. All these processes deposit the crud and scale removed from the reactor internal surfaces on deep bed resins.

Each of these decontamination processes involves several steps and uses different types of resin including cation, mixed, and anion beds. The chelation step generates the cation resins, which can contain up to 75% of the total activity removed during the decontamination. Thus, the cation resins have the highest curie loading and could exceed Class C limits.

Previous studies (References 1 and 4) indicate that some of the resins generated during full RCS decontamination could be GTCC LLW. One study⁴ that specifically addressed GTCC LLW concluded that these resins could be GTCC only when the RCS was decontaminated with fuel assemblies remaining within the core. This conclusion was based on a review of actual sample data for less than full RCS decontamination projects at both PWRs and BWRs. These projects involved relatively low curie loadings on resin beds, and there was some uncertainty as to whether the data were for the cation beds employed or for mixed beds. Since most of the activity is deposited on the cation resin beds, data for these beds are the best indicator of classification status. This study also considered full RCS decontaminations of BWRs, indicating that potential GTCC LLW could be generated with fuel assemblies remaining in the vessel. The estimated resin volumes, which are

Table F-10. Cartridge filters--activity estimates

CARTRIDGE FILTERS						
Activity Estimates						
Reactor Type	BWR	BWR	PWR	PWR	PWR	PWR
1. Filter Type	Control Rod Drive	Fuel Pool Vacuum	Reactor Coolant	Seal Water Injection	Cavity Drain	Fuel Pool
2. Volume(ft ³)	0.07	0.22	0.45	0.1	0.1	0.85
3. Weight(lbs)	3	6	9	2.5	4.0	20
4. Part 61 Status						
-Table 1 Fraction	1.35	1.52	4.27	2.90	2.25	1.63
-Table 2 Fraction	0.02	0.03	0.03	0.03	0.03	0.01
5. R/hr on Contact	150	100	200	50	50	100
6. Activity(Curies)						
H-3	1.34E-04	2.55E-03	1.44E-03	3.41E-04	8.55E-04	3.24E-03
C-14	3.43E-07	1.10E-03	1.06E-02	2.51E-03	5.13E-03	1.94E-02
Mn-54	3.10E-02	1.35E-01	7.56E-02	1.80E-02	8.55E-03	3.24E-02
Fe-55	3.03E+00	2.29E+00	1.51E+00	3.58E-01	2.37E-01	8.96E-01
Co-58	-	-	9.80E-01	2.17E-01	2.85E-01	1.08E+00
Co-60	5.08E-01	1.14E+00	9.12E-01	2.33E-01	2.74E-01	1.04E+00
Ni-63	2.82E-02	1.16E-01	2.33E-01	5.53E-02	5.79E-02	2.19E-01
Zn-65	-	2.85E-01	-	-	2.85E-03	1.08E-02
Sr-90	1.30E-05	1.22E-03	8.06E-04	1.91E-04	9.50E-05	3.60E-04
Cs-137	5.26E-04	1.80E-02	2.00E-03	4.75E-04	1.90E-04	7.20E-04
Pu-241	1.70E-03	2.05E-03	2.13E-02	5.06E-03	4.42E-03	1.67E-02
TRU	1.35E-04	3.48E-04	1.08E-03	5.61E-04	5.31E-04	9.06E-04
Total Curies	3.60E+00	3.99E+00	3.75E+00	8.90E-01	8.77E-01	3.32E+00

assumed to be cation resins, ranged from 207 ft³ for a large BWR LOMI to 275 ft³ for a large BWR CANDECON. Volumes for full RCS decontamination cation resins at PWRs were not estimated in the study.

Actual data available from two similar projects on cation resin beds, one BWR and one PWR, were reviewed for this project. These data confirmed that the scaling factors for the transuranics, which determine classification status, were within the bounds of the previous GTCC study (Reference 4). The resin curie loadings, which directly affect concentration and hence classification, were also relatively low for these data sets.

The other study,⁵ which specifically addressed full RCS decontamination projects with fuel assemblies out at PWRs, indicated relatively large curie loadings on the cation resin beds. These estimated curie loadings on the cation resin beds for three and four loop PWRs and the cation resin volumes are shown on Table F-11.

This study also estimated activity on the cation resins, but the transuranic scaling factors were at least one order of magnitude lower than those discussed in the decontamination resin GTCC study (Reference 4). When the cation bed resin loadings estimated for this study are adjusted to reflect the transuranic scaling factors found in actual data, the cation resins could be GTCC LLW with the fuel assemblies out.

The above studies bound the problem; however, full RCS decontamination projects have never been performed and will not be until the mid-1990s. For this study, it was assumed that the cation resins generated during full RCS decontamination with fuel assemblies out would not exceed Class C limits. On the other hand, decontamination operations performed with fuel assemblies in the vessel could lead to GTCC LLW at both BWRs and PWRs. While full RCS decontaminations are being considered for implementation in the mid-1990s, these operations will be performed with fuel out. Full RCS decontaminations with fuel assemblies in the vessel will be performed only after full RCS decontamination projects with fuel assemblies removed become standard practice in the industry.

To project full RCS decontamination resin volumes, it was assumed that 50% of the cation resins generated from a typical project would be GTCC LLW and that one such decontamination project would be performed at each plant. The estimated volume of GTCC LLW for a typical BWR was 100 ft³; volume for a PWR was 150 ft³. It was also assumed that plants with less than 20 years left in their 40-year life cycles would not perform full RCS decontaminations with fuel assemblies in. This eliminated 17 BWRs and 26 PWRs from consideration in the volume projections. Cumulative projected RCS decontamination resins volumes are 9,500 ft³, or about 270 m³.

Activities were also estimated. For PWRs, the estimates in the PWR full RCS decontamination study (Reference 5) were revised to increase transuranic scaling factors by a factor of 10. For BWRs, the plant-specific cation resin bed data evaluated for this project were used in conjunction with a higher curie loading on the cation resin bed. These estimates are shown in Table F-12.

This table also shows the transuranic scaling factors for the estimated radionuclide distributions. These scaling factors compare favorably to those reported from actual data on smaller decontamination projects in the GTCC decontamination study (Reference 4).

Table F-11. Full RCS decontamination resin volume estimates--PWRs

FULL RCS DECON RESIN VOLUME ESTIMATES – PWR'S					
	Total Curies	3 Loop Resin (ft^3)	4 Loop Resin (ft^3)	3 Loop (uCi/cc)	4 Loop (uCi/cc)
CAN-DEREM Process					
DEREM 0/1 Cation	5,032	300	300	592	592
DEREM 6/1 Anion	335	546	735	22	16
AP 0/1 Cation	470	990	990	17	17
AP 3/11 Cation	872	692	1,254	45	25
Total	6,709	2,528	3,279	676	650
LOMI Process					
LOMI 0/1 Cation	5,032	752	916	236	194
LOMI 1/0 Anion	335	616	832	19	14
AP 0/1 Cation	470	990	990	17	17
AP 3/11 Cation	872	692	1,254	45	25
Total	6,709	3,050	3,992	317	250

Table F-12. Fuel in/full RCS decontamination resin activities--PWRs

FUEL IN FULL RCS DECONTAMINATION RESIN ACTIVITIES – PWR'S			
Plant Type	PWR RCS Fuel-In	PWR RCS Fuel-In	BWR RCS Fuel-In
1. Decon Method	CAN DEREM	LOMI	LOMI
2. Cation Resin (ft^3)	150	150	100
3. Resin Weight (lbs)	7,500	7,500	5,000
4. Part 61 Status			
– Table 1 Fraction	4.68	1.56	5.84
– Table 2 Fraction	NEG	NEG	NEG
5. Activity (Curies)			
C-14	3.28E+00	1.09E+00	1.00E-02
Mn-54	5.91E+01	1.97E+01	4.27E+00
Fe-55	5.88E+02	1.95E+02	4.51E+01
Co-58	4.10E+02	1.37E+02	1.51E+00
Fe-59	2.63E+01	8.74E+00	1.20E-01
Co-60	1.17E+03	3.90E+02	2.10E+02
Ni-63	7.22E+01	2.40E+01	5.53E+00
Zn-65	1.64E+01	5.46E+00	2.23E+00
Sr-90	6.57E+00	2.18E+00	2.00E-02
Nb-95	4.93E+01	1.64E+01	2.40E-01
Cs-134	6.57E+00	2.18E+00	6.00E-02
Cs-137	9.85E+00	3.28E+00	7.00E-02
Ce-141	2.96E+01	9.83E+00	4.00E-02
Ce-144	6.57E+01	2.18E+01	6.00E-01
Pu-241	3.30E+00	1.10E+00	2.92E+01
Cm-242	1.60E+00	5.00E-01	2.00E-02
TRU	1.50E+00	5.00E-01	4.90E-01
Total Curies	2.52E+03	8.39E+02	3.00E+02
Activity uCi/cc	594	197	106
Transuranic Scaling Factors			
TRU/Ce-144	2.28E-02	2.29E-02	8.17E-01
Pu-241/Ce-144	5.02E-02	5.04E-02	4.87E+01
CM-242/Ce-144	2.44E-02	2.29E-02	3.33E-02
TRU/Co-60	1.28E-03	1.28E-03	2.33E-03
Pu-241/Co-60	2.81E-03	2.82E-03	1.39E-01
CM-242/Co-60	1.36E-03	1.28E-03	9.50E-05
Ni-63/Co-60	6.16E-02	6.16E-02	2.63E-02
C-14/Co-60	2.80E-03	2.80E-03	4.75E-05

5.3 Process Waste Summary

Table F-13 summarizes the generation rates for both BWR and PWR process waste. Two sets of values are presented: a maximum set that addresses all cartridge filters; and a minimum set that assumes that PWR plants will manage all filters to eliminate them as GTCC LLW, and that BWR plants will similarly manage fuel pool filters. RCS decontamination resin estimates are the same for both cases.

This table also shows the projected cumulative volumes of GTCC LLW generated from the plants considered from January 1993 through plant closure. These cumulative values are the generation rates per cycle by reactor vendor times the number of fuel cycles remaining before closure for each plant considered. The impact of filter cartridge practices can be significant. Improved filter practices can reduce total process waste volumes by about 25%. The maximum case volume of about 13,500 ft³, or 380 m³, is reduced to about 10,100 ft³, or 285 m³.

Table F-13. Summary of process GTCC LLW

SUMMARY OF PROCESS GTCC WASTES								
	Number Of Plants	Unit Volume (ft ³)	Units Per Cycle	Percent GTCC	GTCC Units Cycle	GTCC (ft ³) Cycle	Cycles To Shutdown	Total GTCC (ft ³)
MAXIMUM ESTIMATES								
1.BWR Cartridge Filters	36							
a. Control Rod Drives		0.07	30	50	15	1.05	564	592
b. Fuel Pool/Vacuums		0.22	12	25	3	0.66	564	372
							Subtotal	964
2.PWR Cartridge Filters	77							
a. Reactor Coolant Systems		0.45	8	25	2	0.9	1,327	1,194
b. Seal Water Injections		0.1	16	25	4	0.4	1,327	530
c. Cavity Drains		0.1	2	25	0.5	0.05	1,327	66
d. Spent Fuel Pools		0.85	4	25	1	0.85	1,327	1,128
							Subtotal	2,918
3.B&W Crud Tank Filters	7	0.95	0.5	100	0.5	0.475	100	48
4.BWR RCS Decon Resins	19	200	NA	50	100	NA	1	1,900
5.PWR RCS Decon Resins	51	300	NA	50	150	NA	1	7,650
							Maximum Totals	13,480
MINIMUM ESTIMATES								
1.BWR CRD Filters	36	0.07	30	50	15	1.05	564	592
2.BWR RCS Decon Resins	19	200	NA	50	100	NA	1	1,900
3.PWR RCS Decon Resins	51	300	NA	50	150	NA	1	7,650
							Minimum Totals	10,142

6. DECOMMISSIONING ACTIVATED METALS PROJECTIONS AND ACTIVITIES

The largest sources of potential GTCC LLW in terms of volume and activity are the activated metals within the reactor vessel. This section describes these components and estimates their volumes and activities at plant shutdown. It also describes the methods used to estimate component activities at shutdown.

6.1 Potential GTCC LLW Sources

These activated metal components consist of the structural components that support the fuel assemblies in the reactor vessel. Their physical characteristics vary with LWR plant type and the component application. 10 CFR Part 61 activation product radionuclide content is directly proportional to the integrated neutron flux received by the component during a plants operating lifetime. A previous WMG study (Reference 1) identified the components described below as potential GTCC LLW.

6.1.1 BWR Decommissioning Internals

The potential GTCC LLW in BWRs consists of the shroud and, in some cases, the upper fuel guide. These are described below.

- Core Shroud--The shroud is a large, cylindrical, type 304 stainless steel structure of about 90 to 180 ft³. It surrounds the core and provides a barrier to separate the upward flow through the core from the downward flow in the annulus. A flange at the top of the shroud mates with a flange on the steam separator assembly to form the core discharge plenum.
- Top Fuel Guide--The top fuel guide is a stainless steel structure that provides lateral support for the top section of the fuel assemblies. It is composed of about 10 to 12 ft³ of stainless steel beams that are joined at right angles, forming square openings for the fuel assemblies. The beams are fastened to a peripheral rim that rests on the core shroud.

As shown previously in Table F-3, 36 BWRs were considered. Nine of these plants were General Electric reactor models 2 or 3; 13 were reactor model 4; two were referred to as model 4/5; and 12 were reactor model 6, General Electric's largest plants. Generally, the size of the potential GTCC components increases with the reactor model number. This evaluation considered two of the larger reactor models, a BWR-4 and a BWR-6.

6.1.2 PWR Decommissioning Internals

The potential GTCC LLW in PWRs consist of the core baffle (shroud) and the lower core barrel. The thermal shield was also considered. These are described below.

- Core Baffle (Shroud)--The baffle is a type 304 stainless steel structure that surrounds the core. It provides for a limited water flow between the baffle and the core barrel for additional cooling of the barrel. The core shroud has a metal volume of about 45 to 90 ft³.

- Lower Core Barrel--The lower core barrel is a type 304 stainless steel cylinder that provides lateral support for the core and directs the flow of coolant water. The lower core barrel typically has a metal volume of about 75 to 125 ft³. For Combustion Engineering plants, the core barrel is relatively large with volumes in the 250 to 300 ft³ range. These units do not have thermal shields.
- Thermal Shield--The thermal shield is attached to the lower core barrel in the high flux region. It protects the reactor vessel by attenuating gamma radiation and fast neutrons that escape the core. The heat energy deposited in the thermal shield is removed by the reactor coolant. The thermal shield is type 304 stainless steel and has a metal volume of about 50 to 110 ft³.
- Flow Skirt--Combustion Engineering plants have an inconel flow skirt as part of the lower core barrel assembly. This component, which has a metal volume of about 10 ft³, will be GTCC LLW. This component was not identified in the previous WMG study (Reference 1).

As shown previously in Table F-4, 77 PWRs were considered. Seven of these plants were Babcock & Wilcox units, 15 were Combustion Engineering plants, and 55 were Westinghouse plants. This evaluation considered the larger plants from each reactor vendor.

6.2 Methodology Used to Determine Component Activities and Classification Status

The ANISN⁶ and ORIGEN⁷ codes were employed to estimate end-of-life activities for potential GTCC components. These are the same codes used previously with typical PWR and BWR plant results listed in NRC reports^{8,9} on decommissioning. For these calculations, the reactor models differed slightly from those reported in the NRC documents. The ANISN code was used to develop radial neutron transport models. Unlike the NUREG approach, the ANISN results were then normalized to actual surveillance capsule flux data obtained from plant-specific surveillance reports. This normalization benchmarked the ANISN results to actual data and provided a more realistic determination of the flux levels and energy spectra at the radial locations of interest for the decommissioning components considered. The surveillance capsule normalized flux data were then used in conjunction with the operating histories discussed in Section 3 as input to ORIGEN2 to determine component activities. The ORIGEN2 cross section libraries used were more recent versions than those used in the NRC documents.

The stainless steel materials compositions used for the activation calculations were obtained directly from NUREG 3474 (Reference 2).

Details of how this methodology was employed, and one of the cases considered are presented below.

6.2.1 Reference Plants

Models could not be developed for all the LWRs considered. Table F-14 summarizes the characteristics of the six reactors chosen for detailed evaluation. For BWRs, the two plants chosen are the larger BWR-4 and BWR-6 models. For projection purposes, the BWR-4 plants will

Table F-14. Reference reactors summary

REFERENCE REACTORS SUMMARY						
Reference	A	B	C	D	E	F
Reactor Vendor	GE	GE	B&W	CE	West.	West.
Reactor Type	BWR-4	BWR-6	177a		Mid-Size	SNUPS
Number of Fuel Assemblies	560	748	177	217	173	193
Heat Output (MWt.)	2,440	3,580	2,770	3,390	3,020	3,410
Initial Enrichment	3.50 %	3.50 %	4.50 %	4.50 %	4.50 %	4.50 %
Potentially GTCC Components and Volumes (ft ³)	Core Shroud (102)	Core Shroud (176)	Core Baffle (52)	Core Baffle (88)	Core Baffle (45)	Core Baffle (71)
			Core Barrel (90)	Core Barrel (290)	Core Barrel (125)	Core Barrel (125)
			Thermal Shield (96)	Thermal Shield None	Thermal Shield (110)	Thermal Shield (45)
Total Volumes	(102)	(176)	(238)	(378)	(280)	(241)

conservatively estimate values for the smaller BWR-2 and BWR-3 models. Both the Babcock & Wilcox and Combustion Engineering plants selected are also among the largest from these vendors. Similarly, two of the larger reactor models from Westinghouse were selected. The mid-size plant evaluated will conservatively estimate values for the smaller Westinghouse plants.

6.2.2 One-Dimensional Transport Calculations

The ANISN computer code was used to perform the one-dimensional neutron transport calculations required by this evaluation. The ANISN cylindrical geometry source was used in the radial model calculations.

The CASK-81 cross section library was the source of neutron cross sections used. The data in the library were collapsed from a fine-group structure using a weighting function representative of a water-uranium mixture.

The GIP computer code¹⁰ was used to read the nuclide-organized CASK cross section library, and create a group-organized input library to be read by ANISN. GIP was utilized to homogenize the compositions of components or materials in given regions of the ANISN model and to create a working library of homogenized cross sections suitable for input into ANISN.

6.2.3 Point Neutron Activation Calculations

The ORIGEN2 computer code was used to calculate the activation and depletion of radionuclides of components exposed to neutrons. One-group neutron cross sections for 22 reactor models are available as input to the code, three of which were used specifically for this evaluation. The first of these one-group cross section sets is representative of a neutron spectrum collapsed across the reactor core of a PWR that has operated to a fuel burnup of 50,000 MW·d/MTU (PWR-UE, Reference 7). The second cross section data set is representative of a BWR which has a fuel burnup of 27,500 MW·d/MTU (BWR-USO, Reference 7). The third is a thermal cross section set, containing only thermal neutron cross sections.

For each location or component of interest and for each material of interest, the volume-averaged total neutron flux (neutrons/cm²·sec), as calculated by ANISN, was input into two ORIGEN2 calculations. One calculation used the reactor core cross section set as input (PWR-UE or BWR-USO); the other used the thermal cross section set. Thermal curie per gram values were adjusted for the local area temperatures to reflect the decrease of activation cross sections at elevated temperatures. Based on the ANISN spectral results of thermal-to-fast-neutron flux ratios, the two ORIGEN2 calculations were weighted to obtain location-specific neutron activation.

6.2.4 Operating Histories

For the two BWR cases, it was assumed that the plants operated for 40 years after commercial operation at a 65% capacity factor. For the four PWR cases, a 70% capacity factor over the same 40-year time period was used.

6.2.5 Radial ANISN Models

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

The PWR ANISN radial models consisted of the reactor core, the core baffle, the core barrel, the thermal shield, the reactor vessel cladding and wall, and the primary shield liner and primary shield wall in cylindrical geometry.

The BWR ANISN models consisted of the reactor core, the core shroud, the reactor vessel cladding and wall, and the sacrificial shield liner and wall in cylindrical geometry. Radial distances from the core centerline at the core midplane elevation for these components have been used. Small components that may be found as part of the reactor internals were neglected for this evaluation. The models included all air and water gaps.

The reactor core region of the reference reactor radial ANISN models were assumed to be comprised of three regions utilizing a fuel loading strategy whereby fuel with the most exposure resides in the core center, and fuel with the least exposure resides around the core periphery.

The core compositions were determined by running ORIGEN2 with plant-specific parameters for fresh, once exposed, and twice exposed assemblies over a fuel cycle. The resulting U-235, U-238, Pu-239, and Pu-240 concentrations for each core region were both volume averaged and cycle averaged, and they were used as ANISN input.

6.2.6 Radial Model ANISN Calculations

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

6.3 Description of a Typical Reference Plant Case

The characteristics of the large Westinghouse unit (Case F) are summarized in Table F-15. The parameters used for the radial ANISN model are presented in Table F-16. The model neglected the reactor vessel thermal insulation, but did consider the primary shield wall. The biological shield was assumed to be six feet of standard concrete.

Usually, two ANISN cases are required per reactor model. The configuration of the thermal shield (four separate pads) in the large Westinghouse PWR required four ANISN cases. Two cases were required for the core baffle and barrel, and two other cases were required for the thermal shield pads. The first two cases assumed an axially averaged, flat radial core power distribution. The neutron source was set equal to that calculated by ORIGEN2 for a three-region core. In the second two cases, the ANISN-calculated neutron flux above 1 MeV was normalized to the results obtained by Westinghouse for the plant-specific surveillance capsule. The configuration of the core internals and the location of the surveillance capsule are illustrated in Figure F-1.

Based on the surveillance capsule analysis, the neutron flux above 1 MeV for the maximum axial and the maximum azimuthal position on the reactor vessel clad interface was calculated to be $3.14\text{E}+10$ n/cm²·sec. This value represents an average neutron flux above 1 MeV of $2.38\text{E}+10$ n/cm²·sec. The ANISN radial model was normalized to this value. A plot for the surveillance capsule normalized thermal flux as a function of radial location is shown on Figure F-2.

Table F-15. Large Westinghouse PWR--reactor summary

LARGE WESTINGHOUSE PWR Reference Case F Reactor Summary	
Rated Thermal Power (MWt.)	3,411
Number of Fuel Assemblies	193
Average Moderator Temperature	591.8
Average Boron Concentration in Moderator	800
Average Core Void Fraction	0
Reactor Assembly Pitch (in.)	8.466
Fuel Assembly Type	West 17x17 OFA
Number of Rod Locations	289
Number of Fueled Rods	264
Fuel Pellet O.D. (in.)	0.3088
Fuel Rod O.D. (in.)	0.36
Fuel Clad Thickness (in.)	0.0225
Fuel Clad Material	Zircaloy-4
Active Fuel Length (in.)	144
Fuel Pin Pitch (in.)	0.496
Metric Tonnes Uranium Metal (MT)	0.426
Initial Enrichment (w/o U-235) %	4.50
Core Volume Calculations	
Total Core Volume (cm.**3)	3.264E+07
Fuel Volume (cm.**3)	9.005E+06
Clad Volume	2.868E+06
Zircaloy-4 Volume (cm.**3)	4.870E+05
Inconel-718 Volume (cm.**3)	2.180E+04
Moderator Volume (cm.**3)	1.924E+07
Void Volume (cm.**3)	1.015E+06

Table F-16. Large Westinghouse PWR--core model parameters

LARGE WESTINGHOUSE PWR Reference Case F Core Model Parameters	
Core Fractions	
Fuel	0.2759
Moderator	0.5896
Zircaloy-4	0.1028
Inconel-718	0.0007
Void	0.0311
Total Core Fractions	1.0000
Moderator Density at Power	0.6884
Model Geometry (cm.)	
Core Equivalent Radius	168.54
Water Region	168.62
Core Shroud/Baffle Equivalent Outer Radius	170.84
Water Region	187.96
Core Barrel Outer Radius	193.68
Thermal Pad Thickness	6.98
Thermal Shield Outer Radius	200.66
Water Region	219.39
Vessel Cladding Outer Radius	219.71
Vessel Wall Outer Radius	41.62
Air Gap Outer Radius	307.34
Primary Shield Liner Outer Radius	307.98
Primary Shield Outer Radius	490.86

Figure F-1. Arrangement of surveillance capsules in the reactor vessel.

Westinghouse Large PWR Thermal Neutron Flux Surveillance Capsule Normalized

Figure F-2. PWR thermal neutron flux.

The thermal and total neutron fluxes were then determined using the normalized ANISN flux results for the core baffle, core barrel, and thermal shield. The total neutron flux was then input to four ORIGEN2 cases for each component, a total of twelve ORIGEN2 cases. For the simpler geometries of the other reference plants, only six ORIGEN2 cases were required. Normally, the fraction of the neutron flux that fell into the thermal energy range was calculated from the ANISN output data. The fraction of neutron flux above thermal energies was also calculated. The resultant spectral fractions were then compared to the ANISN spectral fractions in the core region. Finally, ORIGEN2 neutron activation calculations were performed to obtain radionuclide activities in curies per gram for the composite core spectrum. The resultant ORIGEN2 calculated activities in curies per m³, and the corresponding curie content for the three components considered, are presented in Table F-17. This table also shows the 10 CFR Part 61 classification status for the three components. As expected, the core baffle is well above Class C limits, and the thermal shield is below Class C limits. The core barrel is only slightly above Class C limits. This indicates that the core barrel may not be GTCC LLW in all cases. Since this component represents a significant GTCC LLW volume, investigations beyond those performed for this evaluation are warranted.

Since no axial ANISN model cases were run, the curie results are slightly conservative. They assume that the activation at the periphery of the core also applies to component masses above and below the core. Typically, 10-20% of the mass of the three components of interest lies above and below the core.

The results for this case were compared to those reported in the NUREG report (Reference 8) for a large PWR, the Trojan plant. These comparisons of concentrations for the radionuclides Fe-55, Co-60, Ni-63, and Nb-94 are shown in Table F-18. In some cases, the NRC report reported high or maximum radionuclide values. This evaluation determined average values for these radionuclides. Thus, the NUREG values shown in the Table F-18 have been reduced by a factor of 1.22 to account for peaking. As shown, the values obtained from this evaluation for the core baffle are comparable to the NUREG reported values. This is not the case for the core barrel, where results of this evaluation indicate activities well below the NUREG results.

6.4 Results of Reference Plant Cases

The results of the reference plant analysis for the two BWRs are summarized in Table F-19. As shown, the core shrouds for both the BWR-4 and BWR-6 are above Class C limits. This table also presents the ANISN-calculated and surveillance capsule normalized flux for the BWR-6 case. It is significant that surveillance capsule normalization for this case reduces the incident flux on the shroud by about 35% with corresponding reductions in activity. It is likely that a similar reduction in activity would result for the BWR-4 case if the surveillance capsule data were made available and analyzed.

The BWR cases evaluated were for the larger plants with relatively short operating histories. Some of the older, smaller BWRs have operating histories of up to 12 fuel cycles. Evaluations of these plants could reduce some of the uncertainties in the 10 CFR Part 61 classification status of BWR core shrouds, since the plants are farther along in their life cycles.

Table F-17. Large Westinghouse PWR--summary of results

LARGE WESTINGHOUSE PWR Reference Case F Summary of Results			
	Core Baffle	Core Barrel	Thermal Shield
1. Component Volume (ft ³)	71	125	45
2. Flux (n/cm ² -sec)	1.53E+14	7.90E+12	4.72E+12
3. Normalized Flux (n/cm ² -sec)	7.32E+13	3.98E+12	2.26E+12
4. ORIGEN2 Concentrations (Curies/M ³)			
C-14	1.75E+02	1.17E+01	3.50E+00
Mn-54	9.95E+04	5.00E+03	3.20E+03
Fe-55	9.61E+05	6.57E+04	1.90E+04
Co-60	5.67E+05	4.40E+04	1.71E+04
Ni-59	6.98E+02	6.74E+01	2.09E+01
Ni-63	1.12E+05	8.60E+03	2.52E+03
Nb-94	3.18E+00	1.99E-01	1.10E-01
Totals	1.74E+06	1.23E+05	4.18E+04
5. Part 61 Status			
- Table 1 Fraction	21.26	1.45	0.69
- Table 2 Fraction	16.01	1.23	0.36
6. Radionuclide Content (Curies)			
C-14	3.52E+02	4.14E+01	4.46E+00
Mn-54	2.00E+05	1.77E+04	4.08E+03
Fe-55	1.93E+06	2.33E+05	2.42E+04
Co-60	1.14E+06	1.56E+05	2.18E+04
Ni-59	1.40E+03	2.39E+02	2.66E+01
Ni-63	2.25E+05	3.04E+04	3.21E+03
Nb-94	6.39E+00	7.04E-01	1.40E-01
Totals	3.50E+06	4.37E+05	5.33E+04

Table F-18. Large Westinghouse PWR--radionuclide concentration comparison

LARGE WESTINGHOUSE PWR Reference Case F Radionuclide Concentration Comparison			
Core Baffle			
Radionuclides	Study Result (Curies/M ³)	NUREG Result (Curies/M ³)	Ratio NUREG/ Study (Factor)
Fe-55	9.61E+05	1.07E+06	1.11E+00
Co-60	5.67E+05	7.87E+05	1.39E+00
Ni-63	1.12E+05	9.84E+04	8.78E-01
Nb-94	3.18E+00	4.43E+00	1.39E+00
Core Barrel			
Radionuclides	Study Result (Curies/M ³)	NUREG Result (Curies/M ³)	Ratio NUREG/ Study (Factor)
Fe-55	6.57E+04	1.23E+05	1.87E+00
Co-60	4.40E+04	7.60E+04	1.73E+00
Ni-63	8.60E+03	1.23E+04	1.43E+00
Nb-94	1.99E-01	2.10E-01	1.06E+00

Table F-19. BWR decommissioning (activity estimates)

BWR DECOMMISSIONING		
Activity Estimates		
Reference Reactor	A	B
	GE	GE
Vendor	BWR 4	BWR 6
1. Component Type	Shroud	Shroud
2. Volume (ft ³)	102	176
3. Weight (lbs)	50,400	87,000
4. Flux (n/cm ² -sec) (ANISN)	5.85E+12	5.57E+12
5. Normalized Flux (Surveillance Capsule)	Not Available	3.55E+12
6. Part 61 Status		
– Table 1 Fraction	2.76	1.58
– Table 2 Fraction	3.37	1.81
7. Activity – (Curies)		
C-14	9.23E+01	8.39E+01
Mn-54	1.28E+04	1.36E+04
Fe-55	5.45E+05	4.89E+05
Co-60	3.60E+05	3.57E+05
Ni-59	4.81E+02	4.69E+02
Ni-63	6.81E+04	6.32E+04
Nb-94	9.22E-01	9.33E-01
Total Curies	9.87E+05	9.23E+05

The results of the reference plant analysis for the four PWRs are summarized in Table F-20. Note that the Combustion Engineering plant does not have a thermal shield. It does, however, have an inconel flow skirt with a volume of about 10 ft³. This flow skirt will be well above Class C limits, and its volume is included in the core baffle volume of 88 ft³.

Surveillance capsule data were available for three of the four PWR cases analyzed. In two of the three cases, the normalized flux was consistently lower than the ANISN-calculated flux for each component resulting in reduced activities. For the moderate-size PWR (Case E), the normalized flux was higher than the ANISN-calculated flux. This apparent anomaly was discussed with the operator of this reference plant. During the early years of plant operation, measures to reduce flux at the vessel wall were not implemented, and this mode of operation is reflected in the surveillance capsule data. The current mode of operation at this reference plant will reduce fluxes at the core periphery. Thus, surveillance capsule normalization for PWRs also leads to more realistic estimates of flux for use in activation calculations.

As shown in Table F-20, PWR core baffles are well above Class C limits, and, as expected, the thermal shields are below Class C limits. There is still some uncertainty for some of the core barrels that represent a significant GTCC volume. The analyses indicate that these components in Westinghouse plants are just above Class C limits. Thus, additional investigation of core barrels appears warranted.

The PWR cases evaluated were for the larger plants with relatively short operating histories. Similar analyses of some of the smaller PWRs with longer operating histories would reduce some of the uncertainties in the 10 CFR Part 61 status of the PWR core barrels.

Table F-20. PWR decommissioning

PWR DECOMMISSIONING						
Activity Estimates						
Reference Reactor	E	E	E	F	F	F
Vendor	West. 1	West. 1	West. 1	West. 2	West. 2	West. 2
1. Component Type	Core Baffle	Core Barrel	Thermal Shield	Core Baffle	Core Barrel	Thermal Shield
2. Volume(ft ³)	45	125	110	71	125	45
3. Weight(lbs)	22300	61700	54500	35200	46000	22200
4. Flux(n/cm ² -sec) (ANISN)	1.38E+14	1.94E+12	3.49E+11	1.53E+14	7.90E+12	4.72E+12
5. Normalized flux (Surveillance Capsule)	2.06E+14	2.91E+12	5.22E+11	7.32E+13	3.98E+13	2.26E+12
6. Part 61 Status						
-Table 1 Fraction	45.41	1.12	0.20	21.26	1.45	0.69
-Table 2 Fraction	37.36	0.94	0.16	16.00	1.23	0.36
7. Activity(Curies)						
C-14	5.95E+02	3.14E+01	4.62E+00	3.52E+02	4.14E+01	4.46E+00
Mn-54	3.09E+05	1.21E+04	1.94E+03	2.00E+05	1.77E+04	4.08E+03
Fe-55	3.09E+06	1.69E+05	2.49E+04	1.93E+06	2.33E+05	2.42E+04
Co-60	1.46E+06	1.18E+05	1.80E+04	1.14E+06	1.56E+05	2.18E+04
Ni-59	1.47E+03	1.84E+02	2.81E+01	1.40E+03	2.39E+02	2.66E+01
Ni-63	3.33E+05	2.33E+04	3.49E+03	2.25E+05	3.04E+04	3.21E+03
Nb-94	8.74E+00	5.42E-01	8.53E-02	6.39E+00	7.04E-01	1.40E-01
Total Curies	5.19E+06	3.23E+05	4.84E+04	3.50E+06	4.37E+05	5.33E+04

Table F-20. (continued)

PWR DECOMMISSIONING					
Activity Estimates					
Reference Reactor	C	C	C	D	D
Vendor	B&W	B&W	B&W	CE	CE
1. Component Type	Core Baffle	Core Barrel	Thermal Shield	Core Baffle	Core Barrel
2. Volume(ft ³)	52	90	96	78	290
3. Weight(lbs)	44400	25600	47600	43900	143200
4. Flux(n/cm ² -sec) (ANISN)	1.26E+13	1.20E+14	4.45E+12	1.31E+14	1.23E+13
5. Normalized flux (Surveillance Capsule)	7.34E+12	7.02E+13	2.60E+12	NA	NA
6. Part 61 Status					
-Table 1 Fraction	21.50	2.90	0.97	33.01	4.59
-Table 2 Fraction	16.83	2.60	0.77	26.43	3.91
7. Activity(Curies)					
C-14	2.99E+02	6.41E+01	1.98E+01	5.50E+02	3.19E+02
Mn-54	1.55E+05	2.77E+04	1.09E+04	4.09E+05	1.41E+05
Fe-55	1.70E+06	3.75E+05	1.15E+05	3.84E+06	1.81E+06
Co-60	8.56E+05	2.32E+05	7.94E+04	1.96E+06	1.13E+06
Ni-59	1.03E+03	3.44E+02	1.17E+02	2.16E+03	1.62E+03
Ni-63	1.73E+05	4.65E+04	1.47E+04	4.09E+05	2.25E+05
Nb-94	4.65E+00	1.01E+00	3.71E-01	1.12E+01	5.27E+00
Total Curies	2.89E+06	6.82E+05	2.20E+05	6.63E+06	3.30E+06

7. REFERENCES

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