o General Electric Company (GE)

As of December 31, 1984, General Electric Company has fabricated over 55,000 fuel assemblies containing approximately 3.2 million fuel rods. No information on the overall performance of these fuel rods is available directly from GE. However, GE reported that data for 1983 indicated only 0.007% failure for 8 x 8 array fuel and no failure for Zr-barrier 8 x 8 array fuel (Baily 1985).

Burnups as high as 58,000 MWD/T have been achieved by several test BWR fuel assemblies with no noticeable life-limiting effects on the fuel rod. However, at higher burnups increased production of helium is observed (due to ternary fissions and alpha decay of transuranic products) and also a decrease in fuel rod diameter (Baily 1985, Marlowe 1985, Cheng 1985).

o Westinghouse Electric Corporation

The Westinghouse Electric Corporation does not release information on defective SNF on grounds this is utility proprietary data (Miller 1987). There are no available statistics on Westinghouse fuel defect rates, except those that can be inferred from the NRC and DOE/EIA data. Westinghouse uses the indirect method of monitoring the fission product levels in the reactor primary system (Skaritka 1983, 1985).

2.5.6.3 Data from Nuclear Fuel Service Companies

The Brown Boveri Company (BBC) has provided data on the failure rates of BBC-inspected fuel and on the accuracy of the ultrasonic scanning method as compared to other methods. BBC markets the Failed Fuel Rod Detection System (FFRDS), which uses the ultrasonic method to inspect all PWR fuel rods within an intact fuel assembly. Because the method is based on the presence of water inside the fuel cladding, the FFRDS identifies leaked fuel pins.

As of December 1986, BBC has inspected 1,117 PWR assemblies involving 265,923 fuel rods in the United States and 2,022 PWR assemblies involving 406,471 fuel rods overseas. The U.S. data are presented in Table 2.5.2 (Snyder 1987).

2.5.6.4 Data from the Electric Power Research Institute (EPRI)

EPRI has a specialized fuel performance data base which is not, however, available for external use. The data base includes reactor-specific radioactivity of the reactor coolant and off-gases, fuel cycle history,

coolant chemistry data, and end-of-cycle examination data for failed assemblies (Franklin 1983, 1985; Bailey 1986b; Rumble 1980; Michaels 1985; Lawson 1986b). Communication with EPRI indicates that this data base is of little value to the present study (Franklin 1987).

An EPRI document, EPRI NP-4561, documented and interpreted LERs related to fuel handling and pool storage as of 1986 (Bailey 1986b). It was found that there was no evidence of further deterioration of SNF in storage in wet pools at the reactor sites.

2.5.6.5 Data from the Nuclear Regulatory Commission (NRC)

The data on defective fuel from the NRC comes from the nuclear utilities in the form of licensee event reports (LERs) and from the nuclear fuel vendors. These reports serve their own specific purposes, are selective, and are therefore difficult to use. Since 1978 the NRC has sponsored the Pacific Northwest Laboratory in compiling an annual report on fuel performance with the purpose of providing integrated information for licensing decisionmaking (Houston 1979, Tokar 1981, Bailey 1981b, 1982, 1984, 1985, 1986a).

Tables 2.5.3 and 2.5.4 show data on BWR and PWR fuel rod failures that have been abstracted from the available reports in the above series. Some of the data were calculated. Only failed fuel rod data are presented; other fuel defects are reported in the NRC report series but are more in the nature of isolated observations and are not included here. Since these data were not designed to fit into a unified data base structure, there may be omissions or inconsistencies; one must therefore be cautious in drawing conclusions or making comparisons.

Data from the LER data base that is maintained for the NRC by the Nuclear Operations Analysis Center (NOAC) of Oak Ridge National Laboratory (ORNL) were also reviewed. The nature and causes of off-normal fuel-related events that resulted in an LER were extracted. These data are presented in Table 2.5.5 (Cletcher 1987).

"Fuel failure" as defined in the LERs appears to be subjective. It may include events that did not result in fuel rod failure. As there were 43 reported assembly failures and 95 reported rod failures, the average number of rod failures per assembly failure is 2.2 (1.7 for BWRs and 2.3 for PWRs). This ratio compares favorably with direct inspection observations such as

those made by the Brown Boveri Fuel Service Company (Snyder 1987) using the ultrasonic method.

2.5.6.6 Data From DOE/EIA

The following data were extracted from the RW-859 data file, Section 5, dated Nov. 14, 1986 (Andress 1986). This includes all data extant as of Dec. 31, 1985, and possibly some data "to date" (terminology used in RW-859). It was indicated that the number of 1986-discharged SNF assemblies that were included in the data file for 1985 is very small (Andress 1987).

o Total Discharged and Total Defective Fuel Assembly Populations

As of Dec. 31, 1985, there were 28 BWRs and 52 PWRs that had discharged spent fuel into their spent fuel storage pools. An additional 3 BWRs and 8 PWRs were in operation but had not discharged any spent fuel. Furthermore, there were 10 BWRs and 14 PWRs that were in the process of fuel loading or construction. The total discharged SNF population and the defective SNF population are as follow:

	Total Number	Defective Number	Defective %
BWR fuel assemblies	27,446	3,374	12.3
Long-cooled (before 1981)	18,340	2,953	16.1
Short-cooled (1981-1985)	9,106	421	4.6
PWR fuel assemblies	18,123	1,317	7.3
Long-cooled (before 1981)	10,588	861	8.1
Short-cooled (1981-1985)	7,535	456	6.1

o Distribution of Total and Defective Discharged Fuel Assemblies by Reactor

The number of reactors reporting data is summarized as follows:

	BWRs	PWRs
Total number of reporting reactors	28	53
No. that also reported defective fuel	22(79%)	24(45%)

A higher fraction of BWR sites reported defective fuel than did the PWR sites. This difference may be explained by functional differences, namely:

- BWRs practice reconstitution to a greater degree;
- the channels on BWRs facilitate in-core sipping.

o Distribution of Total and Defective Fuel Assemblies by Rod Array

Many assembly designs have been developed for BWR and PWR fuel over the years. The designs were originally based on core physics, core configuration, manufacturing capability, and economics. They were subsequently

improved with new knowledge on in-service material behavior, new core cooling requirements (such as the emergency core cooling criteria), and higher burnup possibilities. More recently, zircaloy has almost completely replaced stainless steel and inconel as cladding material and grid spacers; the fuel pin diameter has decreased; fuel assemblies contain more fuel rods; fuel pellets are manufactured with higher density and with dished ends; and a zirconium metal barrier is added between the fuel and the zircaloy cladding. Quality assurance and improved manufacturing technologies have allowed burnup to increase from the original design values to higher than 36,000 MWD/T for BWRs and to higher than 40,000 MWD/T for PWRs. Along with these changes, in-service fuel failure has also drastically decreased.

Table 2.5.6 shows the statistics on defective rates of various assembly arrays. Since defective fuel data were not reported for 4254 discharged BWR and 8450 discharged PWR assemblies, the actual numbers with defective fuel may be larger than the values shown.

The average burnup of defective BWR assemblies is generally lower than the average burnup of intact assemblies of the same type. On the other hand, the average burnup of defective PWR assemblies is about the same as that of intact assemblies of the same type. This may be a reflection of the fact that leaky BWR fuel assemblies were easier to detect during refueling by the sipping method and were reconstituted or discharged early, whereas leaky PWR fuel assemblies tended to stay in the core for the entire burnup duty unless the leak was such that technical specifications or occupational radiation burdens were of concern.

o Distribution of Total and Defective Fuel by Defect Categories

Table 2.5.7 shows the statistics for defective fuel assemblies by RW-859 defect codes and by the three 10CFR961 defect categories. The reconciliation method presented in Section 2.5.5.4 has been used.

o Distribution of Total and Defective Discharged Assemblies by Burnup and by Year

The number of annual discharged spent fuel assemblies steadily increased between 1970 and 1985. This of course is a reflection of the increase in the number of reactors and the amount of nuclear electricity being generated by these reactors during the period. The number of defective discharged spent fuel assemblies, on the other hand, is showing a

down trend. This, in spite of the larger number of total discharged fuel assemblies, clearly indicates that fuel failure rates have been significantly reduced.

The average burnup also shows a clear upward trend, from 5000-6000 MWD/T to 25,000-30,000 MWD/T for BWR fuel and from 18,000-20,000 MWD/T to 30,000-35,000 MWD/T for PWR fuel. This trend continues as more reliable fuel and better reactor operation are achieved.

o Discharged Fuel Assembly Defect Rates

Because of incomplete reporting on the number of defective SNF assemblies in RW-859, data from this source have some uncertainties. Two estimates for the defect rates have been made, using two different assumptions.

Table 2.5.8 shows the "low" estimate. It is considered low because it includes the total discharged SNF population of those reactors that reported defective SNF at least once (but may have omitted some defective fuel). The defect rates are 9.9% and 10.5% for BWR and PWR discharged spent nuclear fuel, respectively. Table 2.5.9 shows the "high" estimate. It is considered high because only batches of discharged SNF that contained known defective fuel are included. The defective rates are 14.4% and 23.8% for BWR and PWR discharged spent nuclear fuel, respectively.

2.5.6.7 Reconciliation of Fuel Rod Failure Rates and Fuel Assembly Defect Rates

A reconciliation of fuel rod failure rates and fuel assembly defect rates has been approximated by using (a) the data on assembly defect category to convert defect rate to leaker rate (90% for BWRs and 54% for PWR), (b) the LER and Brown Boveri data for the approximate number of failed fuel rods in a failed assembly (1.6 for BWRs and 2.1 for PWRs), and (c) assuming 49 fuel rods in a BWR fuel assembly and 205 fuel rods each in a PWR fuel assembly. The results are:

	BWRs	PWRs
Fuel rod failure rates as determined from LERs, NRC reports, and vendors' reports (a)		
Prior to 1981	0.04-0.76%	0.01-0.05%
1981-1985	0.01-0.02%	0.01-0.05%
Fuel assembly defect rates as determined from EIA-RW-859 data file (b)		
Prior to 1981	2-41%	3-51%
1981-1985	1-20%	4-30%
Derived fuel rod failure rates from fuel assembly defect rates		
Prior to 1981	0.06-1.13%	0.02-0.28%
1981-1985	0.03-0.55%	0.02-0.17%

(a) From tables 2.5.3 and 2.5.4
(b) From tables 2.5.8 and 2.5.9. The value of about 94% for BWR fuel defect rate for 1973 was considered an outlier and therefore excluded.

The above reconciliation indicates that the number of leaky fuel rods in storage may be more than twice more numerous than can be inferred from the NRC-associated fuel reliability data. The difference is not surprising since the LER data would not be expected to include all known leakers, for a number of reasons, such as : (a) not all leakers trigger an LER and (b) some leakers are not identified until after final discharge.

2.5.7 Conclusions

Examination and analysis of defective fuel data from the various industry and government sources leads to the following conclusions:

1. The majority of data supplied by the nuclear utilities, fuel vendors, and nuclear fuel service companies deals with leaked fuel while the reactor is in operation. Fuel rod failure rates are between 0.02% and 0.07% yon a cumulative basis. Instances of rod failure a factor of 10 higher have been reported for certain fuel lots. Newer fuel is claimed to have a reliability factor up to 10 lower. While these data give confidence in the ability to operate reactors with very low fuel leakage, they do not provide

all the information on the defect rates of discharged SNF that is required for characterization.

2. Other data from the nuclear fuel vendors or from industry organizations, such as the Electric Power Research Institute, deal mostly with fuel research, development, and marketing and are of limited use in terms of overall characterization.

3. The major source of data that is of significance to the OCRWM mission has been collected since 1984 and updated annually by the DOE/EIA. The data collected for calendar year 1985 have shown significant improvement over that for 1984 and further improvements are expected in the future.

4. The percentage of PWR respondents that reported defective discharged fuel assemblies is less than that of BWR respondents. This may be due to the fact that (a) PWR fuel assemblies were not routinely checked for fuel defects in the same manner as BWR fuel assemblies because such a check is more expensive and yet may not lead to assembly reconstitution or fuel warranty claim and (b) the RW-859 form does not require reporting on fuel defects if such information is not available.

5. Discharged fuel assemblies between 1970 and 1985 show a clear trend of increase in number and in burnup. Defective assemblies, however, show a clear trend of decrease for BWRs, but a not-so-clear trend for PWRs.

6. A reconciliation of the fuel rod failure rates and fuel assembly defect rates can roughly be made by modifying the latter using the fraction of leaks among defects, the number of leaked fuel rods in a defective assembly, and the number of fuel rods within a fuel assembly. This reconciliation leads to a number of leaked fuel rods now in storage more than twice the number that can be calculated from NRC-associated fuel reliability data.

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Table 2.5.1. DATA FROM NUCLEAR FUEL VENDORS

Fuel Vendor	Irradiated Fuel Rods	Average Rod Failure Rate
Advanced Nuclear Fuel (Formerly Exxon)	(cumulative as of 1/1987)	
BWR	365,938	0.027%
PWR	1,002,495	0.014%
Babcock & Wilcox		
1982	287,872	0.006%
1983	336,128	0.009%
1984	310,000	0.011%
Combustion Engineering		
1982	not available	0.01%
1983	not available	0.02%
1984	333,883	0.02%
General Electric		
1983	not available	0.007%
Westinghouse	not available	not available

Table 2.5.2. DATA FROM POOLSIDE ULTRASONIC INSPECTION
(Source: Snyder 1987)

	Fuel Assemblies			Fuel Rods		
Fuel Type	Total Number	No. with Leaked Rods	Percent Leaked	Total[c] Number	No. of Leaked Rods	Percent Leaked
B&W 15[a]	117	14	12.0	26,325	18	0.07
CE 14	7	4	57.1	1,372	4	0.29
CE 16	269	35	13.0	68,864	124	0.18
Exxon 16	36	2	5.6	9,216	2	0.02
W 14CE[b]	65	42	64.6	12,740	51	0.40
W 14	117	19	16.2	22,932	39	0.17
W 15	340	86	25.3	76,500	149	0.19
W 17	166	14	8.4	47,947	29	0.06
Total	1,117	216	19.3	265,923	416	0.16

(a) B&W = Babcock and Wilcox, CE = Combustion Engineering, Exxon = Exxon Nuclear Company (changed to Advanced Nuclear Fuel Corporation as of 1987), W = Westinghouse.

(b) W 14 CE is 14 x 14 fuel assemblies manufactured by W for a CE reactor.

(c) Brown Boveri has counted the rod positions available in an assembly as the number of fuel rods. This is a high estimate of fuel rods because there are also control rods in some of those positions.

Table 2.5.3. ABSTRACT OF BWR FUEL ROD FAILURE DATA REPORTED IN NRC ANNUAL REPORTS (a)

Fuel Vendors and Dates	Incore and Discharged Fuel Rods		Failure Rates		Calculated No. of Defective Rods	
	Annual	Cumulative	Annual	Cumulative	Annual	Cumulative
Exxon Nuclear Co.(b)						
12/81	--	116,891	--	0.012%	--	14
12/82	32,391	142,471	0.130%	--	42	56
12/83	45,801	170,710	0.012%	--	6	62
12/83	53,324	172,176	0.022%	0.043%	12	74
GE 7x7 Array(c)						
9/71	--	>440,000	--	0.200%	--	880
9/74	--	>810,000	--	0.760%	--	6156
12/76	--	1,040,000(e)	--	Proprietary	--	--
12/76	--	110,000I, R(d)	--	0.043%	--	47R
1/80	--	285,376I, R	--	0.066%R	--	188
1/80	504,161	--	1.010%	--	5092	--
GE 8x8 Array						
5/79	--	676,053	--	0.028%R	--	189(e)
	--	117,676I	--	0.	--	0
1/80	--	758,016	--	0.016%R	--	121(e)
	--	268,398I	--	0.002%R	--	5
12/80	--	1,239,000	--	0.020%R	--	248
12/81	--	1,489,000	--	<0.020%R	--	<298
12/82	--	1,821,338	--	0.020%R	--	364
12/83	1,300,000R	--	0.007%R	--	91	455
12/84	1,300,000R	>2,460,840R	0.019%R	0.024%R	130	585

(a) Source: Compiled from Houston 1979; Tokar 1981; Bailey 1981b, 1982, 1984, 1985, 1986a.

(b) Note that the numbers given here are slightly different from those reported by Exxon Nuclear Company (Sofer, 1985).

(c) Counting of 7x7 arrays no longer reported beyond 1980.

(d) I = improved (rearranged) array; R = reported (not calculated) data.

(e) The reported numbers are duly reported; however, they may be inconsistent with other reported or calculated numbers. Efforts to resolve the inconsistencies have not been successful. At any rate, the accuracy of the individual reported reliability should not be of much importance.

Table 2.5.4. ABSTRACT OF PWR FUEL ROD FAILURE DATA REPORTED IN NRC ANNUAL REPORTS (a)

Fuel Vendors and Dates	Incore and Discharged Fuel Rods(b,c) Annual	Incore and Discharged Fuel Rods(b,c) Cumulative	Failure Rates (c) Reported Annual	Failure Rates (c) Calculated Cumulative	Calculated (c,d) No. of Defective Rods Annual	Calculated (c,d) No. of Defective Rods Cumulative
Babcock & Wilcox						
12/80	--	586,034R	--	0.037%	--	216R
12/81	--	647,728R	--	0.037%	24R	240
12/82	--	686,608R	--	0.037%	30R	256
12/83	--	769,184R	--	0.037%	30R	286
12/84	--	797,472R	--	0.040%	33R	319
Combustion Engineering						
12/79 to 12/81	--	579,201R	--	0.047%	--	274R
12/82	266,000R	680,613R	0.010%R	0.044%	27	301
12/83	333,700R	780,872				
12/84	333,883R	839,980R	0.020%R	0.052%	67	435
Exxon Nuclear						
12/81	--	410,965R	--	0.013%R	--	52R
12/82	160,856R	547,725R	0.110%R(e)	0.042%	177	229
12/83	237,807R	623,572R	0.013%R	0.042%	31	260
12/84	339,013R	798,737R	0.002%R	0.033%R	7	267
Westinghouse	No data available except a plot of reactor coolant activity of Westinghouse reactors between 1972-1984					

(a) Source: Complied from Houston 1979; Tokar 1981; Bailey 1981b, 1982, 1984, 1985, 1986a.

(b) Note that annual numbers cannot be added to obtain the cumulative number because fuel rods stay in the core for more than one year.

(c) R = reported (not calculated)

(d) The calculated cumulative failure rate was obtained by dividing the calculated cumulative number of defective fuel rods by the reported number of cumulative fuel rods. The calculated number of defective fuel rods was obtained by multiplying the reported failure rate by the population of exposed fuel rods.

(e) This figure appears to be high by a factor of 10 in the source document.

Table 2.5.5 LICENSEE EVENT REPORT (LER) DATA [a]
(Cletcher 1987)

	Fuel Rods		Fuel Assemblies	
	BWR	PWR	BWR	PWR
Total number of failures	10	85	6	37
Nature of failure				
Leaks	9	73	4	20
Deformation/maladjustment	1	11	2	16
Others	0	1	0	1
Cause of failure				
Cladding degradation	80%		19%	
Vibration	6%		-	
Component failure	5%		56%	
Wear, age	2%		7%	
Drop, impact	-		2%	
Others	7%		16%	

[a] Data analysis by Cletcher 1987 for LER data taken from computerized data base maintained by the Nuclear Operations Analysis Center of the Oak Ridge National Laboratory.

Table 2.5.6 DEFECT RATES BY ARRAY
(Data corrected for those reactors that did not report defects as of 12/31/1985)
Source: DOE/EIA RW-859

Array	Total No. Discharged Assemblies	No. Discharged That Reported Defects	No. Of Defective Assemblies	Rate (%)
Boiling Water Reactors				
Unknown	12,709	10,623	1,362	12.8
6 x 6	888	888	168	18.9
7 x 7	5,308	4,200	260	6.2
7 x 7NP[a]	484	484	10	2.1
8 x 8	4,851	2,917[d]	32[d]	1.1
8 x 8NP[b]	920	920	81	8.8
8 x 8R [c]	1,859	1,531	128	8.4
9 x 9	78	78	24	30.8
10 x 10	235	235	102	43.4
11 x 11	114	114	5	4.4
TOTAL	27,446	21,990	2,172	9.9
Pressurized Water Reactors				
Unknown	4,573	3,568	687	19.3
4 x 4	1	0	-	-
14 x 14	3,482	363	6	1.7
14 x 14[e]	1	0	-	-
15 x 15	6,704	3,661	200	5.5
16 x 16	719	226[f]	5[f]	2.2
17 x 17	2,643	1,514	78	5.2
TOTAL	18,123	9,332	976	10.5

BWR notes: [a] At Hatch 1; [b] at Hatch 1 & 2; [c] at Browns Ferry 1, 2, & 3; Peach Bottom 2 & 3; Fitzpatrick; and Susquehanna 1; [d] excludes 1,202 at Vermont Yankee.

PWR notes: [e] A special assembly at Calvert Cliff; [f] excludes 341 at Yankee Rowe.

Table 2.5.7. DEFECTIVE FUEL ASSEMBLIES BY DEFECT CATEGORY AND/OR CODE

Defect Category or Code[a]	Boiling Water Reactors			Pressurized Water Reactors		
	Number	Weight(T)	Average Burnup (MWD/T)	Number	Weight(T)	Average Burnup (MWD/T)
			Data from RW-859			
F-1	135	24.752	22,670	398	181.249	31,144
F-2	1,860	344.729	12,894	206	91.468	24,501
F12	-	-	-	60	27.246	35,585
F-3	46	8.765	1,134	-	-	-
LAT(ER)[a]	(691)	(126.962)	23,350	(341)	(83.159)	27,605
1	7	1.271	19,182	14	5.885	25,193
1 7	151	27.897	24,123	-	-	-
15	48	5.760	14,168	28	12.724	28,995
137	-	-	-	2	0.734	26,406
17	8	1.459	13,986	63	25.676	25,705
157	-	-	-	13	5.950	9,633
3	11	2.020	13,609	5	1.795	20,744
5	2	0.304	17,854	-	-	-
7	406	71.296	15,697	184	81.664	25,193
71	7	1.282	22,339	3	1.388	19,064
715	1	0.183	23,753	-	-	-
751	1	0.183	23,416	-	-	-
TOTAL	3,374	616.863	16,162	1,276	435.78	28,105
			Reconciliation with 10CFR961[b]			
F-1	203 (7%)	36.107	20,525	445 (46%)	201.653	30,742
F-2	2,434 (91%)	445.576	14,123	531 (54%)	234.126	25,760
F-3	46 (2%)	8.765	1,134	-	-	-
LAT(ER)	691	126.962	23,350	341	83.159	27,605
TOTAL	3,374	615.410	16,162	1,317	518.938	28,105

[a] The definition of categories and codes has been given in the text.
LAT(ER) means that the defective fuel will be reported at a later date.

[b] Reconciliation is achieved by assigning all items having a defect codes F12 and 7 to category F-2; defect code 2 (there is none) to category F-3; and defect codes 1, 3, 4, 5, 6, (but not F12, 2 or 7) to category F-1.

Table 2.5.8. DEFECT RATES OF ASSEMBLIES: LOW ESTIMATE

(All Discharged SNF from Reactors that Reported Defective Fuel One Time or More Are Included [a])

	Boiling Water Reactors			Pressurized Water Reactors		
	No. of Assemblies	No. Ass. Defective	Defect Rate (%)	No. of Assemblies	No. Ass. Defective	Defect Rate (%)
Before 1971	125	32	25.6	160	1	0.6
1971	373	115	30.8	52	-	-
1972	574	65	11.3	136	25	18.4
1973	288	270	93.8	111	3	2.7
1974	846	334	39.5	231	31	13.4
1975	925	367	39.7	253	17	6.7
1976	1,459	388	26.6	646	78	12.1
1977	1,662	107	6.4	682	35	5.1
1978	1,769	110	6.2	655	131	20.0
1979	1,764	91	5.2	943	103	10.9
1980	2,768	43	1.6	836	57	6.8
1981	2,039	41	2.0	979	155	15.8
1982	1,849	56	3.0	732	70	9.6
1983	1,957	25	1.3	990	99	10.0
1984	1,935	31	1.6	867	97	11.2
1985	1,515	97	6.4	863	33	3.8
TEMP[b]	142	-	-	196	41	-
TOTAL	21,990	2,172	9.9	9,332	976	10.5

[a] Data from Vermont Yankee (BWR) and Yankee Rowe (PWR) were excluded.
[b] TEMP = Temporary discharge; may be reinserted into the reactor.

Table 2.5.9. DEFECT RATES OF ASSEMBLIES: HIGH ESTIMATE
(Only those SNF Lots that Reported Defects are Included)

	Boiling Water Reactors			Pressurized Water Reactors		
	Pop. W/Def. Assemblies	Defective Assemblies	Defect Rate (%)	Pop. W/Def. Assemblies	Defective Assemblies	Defect Rate (%)
Before 1971	94	32	34.0	60	1	1.7
1971	373	115	30.8	-	-	-
1972	574	65	11.3	83	25	30.1
1973	288	270	93.8	56	3	5.4
1974	809	334	41.3	150	31	20.5
1975	925	367	39.7	114	17	14.9
1976	1,455	388	26.7	341	78	22.9
1977	1,408	107	7.6	110	35	31.8
1978	1,501	110	7.3	315	131	41.6
1979	1,325	91	6.9	405	103	25.4
1980	2,746	43	1.6	112	57	50.9
1981	814	41	5.0	511	155	30.3
1982	593	56	9.4	495	70	14.1
1983	692	25	3.6	479	99	20.7
1984	843	31	3.7	461	97	21.0
1985	479	97	20.3	272	33	12.1
TEMP[a]	142	0	-	144	41	28.5
TOTAL	15,061	2,172	14.4	4,108	976	23.8

Note: Only those discharged SNF batches that also contained known defective assemblies were selected. The data for Vermont Yankee and Yankee Rowe were not used.

[a] TEMP = Temporary discharge; may be reinserted into the reactor.

2.6 SPECIAL LWR FUEL FORMS

Most, but not all, LWR fuel assemblies are currently being stored intact and have relatively standard dimensions. This section provides information on fuels that are different from most LWR fuel assemblies because they have been disassembled or highly degraded in some fashion, their design parameters are radically different in some way, or their fuel rods have been consolidated. Fuel rods and assemblies that have been cut apart and disassembled for testing, evaluation, and research are covered in Section 4.5, Miscellaneous Fuels.

2.6.1 Degraded Fuel from TMI-2

LWR fuel at TMI-2 is highly degraded and will require special handling. It can be assumed that the entire core will be handled as debris and placed in special canisters. The core loading at the time of the accident included 82,023 kg of uranium, of which 2,064.4 kg is 235U. This debris, which includes Zircaloy cladding and other assembly structural materials, is being shipped to DOE's Idaho facility for storage. Three styles of containers are being used - a fuel canister, a knockout canister, and a filter canister (Childress 1986). These are specially designed for different modes of loading, depending on the physical state of the degraded fuel. All three have the same external dimensions - 14-in. diameter, 150-in. overall length, with dished bottoms and flat tops. The internal designs differ as well as the manner in which neutron poisons (for criticality control) are placed in the canisters.

As of December 31, 1986, a total of 13.9 MT of debris had been shipped to Idaho (Ball 1987). This material had a volume of 13.2 m3 and contained about 700,000 Ci of radioactivity. This work will continue until the entire core has been removed.

2.6.2 Nonstandard Fuel Assemblies

Certain fuel assemblies may require special handling because of nonstandard dimensions, unique designs, extremely high burnups, or

differences in the fuel pellets. Westinghouse 17 x 17 assemblies for the South Texas plants are more than 3 feet longer than standard 17 x 17 fuel assemblies. Early fuel assemblies for both PWR's and BWR's were shorter than current fuel assemblies. The fuel assemblies at the Dresden-1 and Humboldt Bay were 6 x 6 arrays; at the Big Rock Point reactor, the original fuel assembly was a 12 x 12 array but has been replaced with both 9 x 9 and 11 x 11 arrays. The assemblies from the Indian Point 1 and Yankee-Rowe reactors are non-square arrays. Several assemblies at various reactors have been exposed to extremely high burnups; annular fuel pellets have been used in others; all manufacturers test a new change on a few test assemblies before implementing the change in all assemblies - the properties of these unique assemblies may need special characterization. Differences in many of these areas are covered in the LWR Assemblies Data Base, but it is important to make a special note of the difficulties they may present. Any of these factors may require specialized equipment for the safe storage, transportation, consolidation, and/or disposal of these assemblies.

2.6.3 Consolidated LWR Fuel

Consolidated LWR spent fuel must also be characterized in terms of physical descriptions (length, width, and weight of cans), quantitative information (how many cans of PWR fuel? BWR fuel? When?), and radiological properties (radioactivity, thermal output, neutron production, photon spectra). Consolidation studies and demonstrations are ongoing at several sites. An early demonstration of wet consolidation was done at West Valley. Northeast Utilities have recently done an underwater test on six assemblies using Combustion Engineering equipment; they achieved a 2-to-1 volume reduction at the end of the test. Northern States Power has just finished an underwater test with 40 assemblies, using Westinghouse equipment. Dry consolidation tests are planned at INEL. Data from these tests will be incorporated in the data base as this information becomes available.

2.6.4 Shippingport LWBR Fuel

The Shippingport PWR was subsequently converted to serve as a test unit for a Light Water Breeder Reactor (LWBR) and was fueled with a mixture of 233U and thorium oxide. This spent fuel has been shipped to DOE's Idaho facility for storage (Schreiber 1987). A total of 65 assemblies were involved with a uranium content of about 700 kg (mostly 233U) and a thorium content of 47 MT as a mixed (Th,U) dioxide.

2.6.5 References for Section 2.6

Ball 1987. Letter from L.J. Ball, EG&G Idaho, to S.N. Storch, Oak Ridge National Laboratory, dated April 9, 1987.

Childress 1986. P.C. Childress, et al., "TMI-2 Defueling Canisters", in High-Level Nuclear Watse Disposal, H.C. Burkholder, ed., 1986.

Schreiber 1987. Letter from J.J. Schreiber, Shippingport Station Decomissioning Project Office, to S.N. Storch, Oak Ridge National Laboratory, dated April 14, 1987.

2.7 SPENT FUEL DISASSEMBLY HARDWARE

2.7.1 Overview

Spent Fuel Disassembly (SFD) hardware is defined as the pieces of a fuel assembly left after the fuel rods have been removed. Generally, SFD hardware for PWR fuel assemblies includes guide tubes; instrument tubes; top and bottom nozzles; grid spacers; hold-down springs; and attachment components, such as nuts and locking caps. Guide tubes and instrument tubes are hollow metal cylinders into which control elements, neutron sources and poisons, and/or instrumentation are inserted. Recently, most tubes have been made of Zircaloy, although early tubes were made of stainless steel. The top and bottom nozzles, which are relatively large solid pieces of stainless steel, direct the flow of water around the fuel rods and provide structural support. Most vendors make nozzles from stainless steel 304, although other similar alloys (SS304L, SS348, CF3M) have been used. Grid spacers, which historically were made of a springy material like Inconel, have recently been made of Zircaloy because of the low neutron absorption cross section. They are attached in some manner to the instrument and/or guide tubes at various locations throughout the assembly to provide both positioning and support for the fuel rods. Together these items make up the skeleton of the fuel assembly. Different vendors use different methods to attach these various components - spot welding, bolting in place, and crimping are all used. The nuts, locking caps, and grid sleeves used in these different methods of attachment are also SFD hardware. The hold-down springs are typically made of a nickel-base alloy and are used to hold the fuel rods down against the bottom nozzle, opposed to the upward flow of water through the assembly.

For BWR fuel assemblies, SFD hardware includes the top and bottom tie plates, compression springs for individual fuel rods, grid spacers, and water rods. In a BWR assembly, structural support is provided by the grid spacers, the top and bottom tie plates, fueled tie rods, and water rods. The tie plates have typically been made of stainless steel

304 and the grid spacers of Zircaloy. The position of the grid spacers is usually determined by welded tabs on the water rod. These tabs are welded onto the water rod so as to not damage the integrity of the fuel-rod cladding. The water rod(s) is similar to guide tubes in that it is a hollow Zircaloy tube. Its purpose is to provide additional water for neutron moderation rather than a location for control elements, etc. BWR assemblies typically use a separate compression spring for each individual fuel rod. These springs, located in the gas plenum region, hold the fuel rod against the bottom tie plate.

Both PWR and BWR assemblies contain some unique pieces of SFD hardware. Structural support for the fuel assemblies for the Palisades and Yankee-Rowe plants is provided by solid bars of Zircaloy rather than guide tubes. The top end fittings of Combustion Engineering's 14 x 14 and 16 x 16 fuel assembly designs are not solid pieces of metal but rather are two flat plates separated by five large metal posts surrounded by Inconel hold-down springs. Westinghouse's new entry into the BWR reload market, the QUAD+, has a unique channel assembly that consists of a water cross welded to a relatively standard outside channel. It is mechanically attached to the bottom nozzle.

With the exception of the QUAD+ channel assembly, BWR fuel channels are not included with SFD hardware in this data base. This is simply a reflection of the fact that reuse of fuel channels is becoming more prevalent. The radioactive properties of these reused channels can be more accurately described by using the format of the LWR NFA Hardware Data Base, where items may be subject to irradiation for more than 1 assembly lifetime.

Nonfueled burnable poison rods in PWR assemblies are also not included as SFD hardware since these rods would probably not be separated from the fueled rods during consolidation but would be included in the consolidated canister with the fuel rods. The extra

handling operations involved with identifying, separating, and disposing of the nonfueled rods safely greatly offset the disadvantage of a slight increase in volume of consolidated fuel-rod canisters.

SFD hardware is not a major technical issue unless fuel assemblies are consolidated prior to emplacement in a repository. Without consolidation, SFD hardware would remain with the assembly where the radioactivity of the hardware is small compared to the radioactivity of the fuel and fission products. If fuel assemblies are consolidated, several concerns arise with regard to the remaining hardware. These include: How much of it is there going to be? and What are the radiological characteristics of it? The issues addressed by this study are the quantitative and radiological characterization of SFD hardware.

2.7.2 Quantitative Characterization

A general description of SFD hardware for the different assembly types is included in the descriptions of the versions in section 2.2 of this report. Specific pieces of SFD hardware are described in as much detail as possible in the LWR Assemblies Data Base. This description includes the name of the specific pieces, the number of pieces per assembly, the weights, and the construction materials. For each assembly, these parts are listed on page 2 of the Physical Description Report. Physical Description Reports for all assemblies for which data are available are given in Appendix 2A, Physical Descriptions of LWR Fuel Assemblies.

Summary quantitative information on how much SFD hardware is associated with PWR fuel assemblies is given in Table 2.7.1, and for BWR assemblies, in Table 2.7.2. The number of assemblies of each type is a question that is addressed by both the LWR Quantities Data Base (Appendix 2D) and, to some extent, the shipping records supplied by the various vendors (Appendix 2G).

2.7.3 Methodology for Radiological Characterization

The disposal of radioactive wastes is primarily regulated by two sections of the Code of Federal Regulations (CFR). The disposal of high-level wastes and spent nuclear fuel in a deep geologic repository is governed by 10 CFR 60, whereas the disposal of low-level wastes in near-surface burial is governed by 10 CFR 61. Although SFD hardware [and Nonfuel Assembly (NFA) hardware] is not specifically included in either set of these regulations, neither is it specifically excluded. 10 CFR 61 puts low-level wastes into four categories -- Class A, Class B, Class C, and Greater than Class C. Inclusion in any one of these categories is based on concentrations of radioactive isotopes in the material to be disposed of. Because SFD hardware does not contain uranium or other actinides, activation products are the sole source of radioactivity. In particular, 10 CFR 61 puts limits on the concentrations of 14C, 59Ni, 63Ni, and 94Nb that are permitted to be present.

The computer code ORIGEN2 has been used to estimate the concentrations of these isotopes that are present in SFD hardware. ORIGEN2 is a revised edition of the widely used code ORIGEN - the Oak Ridge Isotope Generation and Depletion Code (Croff 1980). It is a versatile code for use in the simulation of the conditions occurring in the nuclear fuel cycle and in the calculation of the nuclide compositions and characteristics of the materials contained therein. In particular, ORIGEN2 can model the effects of irradiation on a material.

ORIGEN2 calculates the concentration of all isotopes present in an activated material at a given time on the basis of the initial isotopic composition of the material, the intensity and duration of the neutron flux to which it has been exposed, the cross sections for neutron activation, and the half-lives and decay products of the radioisotopes involved.

Because of the severity of conditions to which materials are exposed in the core of a nuclear power reactor, relatively few materials have been used in the fabrication of fuel assemblies. These materials are alloys of zirconium (Zircaloy-2 and Zircaloy-4), alloys of nickel (Inconel-625, Inconel-718, and Inconel X-750), and stainless steels (stainless steel 304). These materials were chosen for the resistance to corrosion, retention of structural strength after intense irradiation, and low neutron absorption cross sections.

The elemental composition of these materials is determined by standards set by the American Society for Testing and Materials (ASTM). Because they are the precursors to the isotopes that determine low-level waste categories, the initial amounts of nitrogen, nickel, cobalt, and niobium are of particular interest. If these elements are included in the material specifications, it is generally as an upper limit for an impurity. Often they are not included at all, although they are present in trace quantities. For niobium in particular, trace quantities may be sufficient for the irradiated material to exceed the Class C limits. Niobium as an impurity does not affect the physical and chemical characteristics of the material. Thus, as long as the niobium concentrations are below some reasonable level, the actual amount is not of concern to the ingot manufacturer or the fuel assembly vendor. The input to ORIGEN2 of the elemental concentrations of these materials has been based on reasonable upper limits and ASTM specifications. The values used for nitrogen, cobalt, nickel, and niobium are given below.

Material	Nitrogen	Cobalt	Nickel	Niobium
Inconel-625	---	1.00%	57.9%	3.65%
Inconel-718	1300 ppm	4700 ppm	52.0%	5.55%
Inconel X-750	1300 ppm	6490 ppm	72.2%	0.9%
St. Steel 304	1300 ppm	800 ppm	8.92%	100 ppm
Zircaloy-2	80 ppm	10 ppm	500 ppm	120 ppm
Zircaloy-4	80 ppm	10 ppm	20 ppm	120 ppm

The neutron flux intensity and exposure used by ORIGEN2 are a function of the input variables of type of reactor, burnup, initial enrichment, and neutron exposure zone. ORIGEN2 has been used to model the conditions in the core of several different reactor types, including high-temperature gas-cooled reactors, PWRs, and BWRs. Activation of the materials used in SFD hardware has been calculated for a standard burnup and a high burnup. For PWR's, concentrations of isotopes in materials exposed to standard burnup (33,000 MWd/MTIHM) were calculated using the standard PWR model; concentrations in materials exposed to high burnup (60,000 MWd/MTIHM) were calculated using the PWR extended burnup model. An initial enrichment of 3.2% was used for the former and 4.15% for the latter. The concentrations of isotopes in materials exposed to standard burnup (27,500 MWd/MTIHM) in a BWR were calculated using the standard BWR model. A high burnup run for the BWR case was not made; the development of a BWR extended burnup model has just recently been completed. An initial enrichment of 2.75% was used for the BWR case.

Much of the hardware is outside the active core region, and a correction is required for this. The flux is a maximum in the core region and drops off rapidly outside the core. The neutron flux has been modeled in four exposure zones - the top end region, the gas plenum region, the core region, and the bottom end region. The neutronic flux in each region has been calculated (Luksic 1986a, Croff 1978) on the basis of the materials present in each region. These relative neutron flux factors, which were incorporated into the ORIGEN2 runs, are listed below.

		Top End	Gas Plenum	In Core	Bottom End
Luksic	PWR	0.011	0.083	1.000	0.063
	BWR	0.0063	0.065	1.000	0.071
Croff	PWR	0.011	0.042	1.000	0.011
	BWR	0.131	0.500	1.000	0.131

The later values, calculated by Luksic, were used for the present calculations.

Neutronic flux calculations were based on an average PWR and an average BWR assembly. Currently, efforts are under way to perform the same modeling for different array designs to determine if these relative flux factors and effective cross sections are assembly model dependent. Differences between the array designs of different manufacturers are expected to be minor in terms of effect on neutron flux.

To simplify the computational algorithm and conserve storage space, ORIGEN2 uses a single energy-averaged neutron absorption cross section for each isotope rather than a set of individual neutron energy dependent cross sections. New effective cross sections were calculated in the different zones for the precursors of the isotopes of particular interest because the neutron energy distribution changes from the in-core neutron zone to the gas plenum, top end and bottom end zones. These isotopes were 14C, 50Ni, 63Ni, and 94Nb. These isotopes are primarily the result of the following reactions:

14N(n,p)14C
59Co(n,β)60Co; 59Co(n,β)60Co*
58Ni(n,β)59Ni
62Ni(n,β)63Ni
93Nb(n,β)94Nb; 93Nb(n,β)94Nb*

The neutron spectra and the individual neutron energy dependent cross sections were energy averaged to obtain spectral flux factors for each neutron zone. These spectral flux factors are multiplied by the cross section supplied by the appropriate ORIGEN2 library to give the effective energy averaged cross section for these reactions in a particular zone. The neutron flux factors and the spectral flux factors obtained by Croff (Croff 1978) differ from the results obtained by Luksic (Luksic 1986a). Experimental studies on actual samples of SFD hardware are ongoing and are expected to reconcile these differences. The spectral flux factors obtained by Luksic (Luksic 1986a) and incorporated in the ORIGEN2 runs are given below.

Reactor Type	Region	Nitrogen	Cobalt	Nickel	Niobium
PWR	Top End	4.3	3.1	4.6	1.6
PWR	Gas Plenum	6.3	4.4	6.7	2.1
PWR	In Core	1.0	1.0	1.0	1.0
PWR	Bottom End	4.3	3.2	4.6	1.7
BWR	Top End	1.8	2.1	1.8	1.3
BWR	Gas Plenum	3.6	3.2	3.7	2.1
BWR	In Core	1.0	1.0	1.0	1.0
BWR	Bottom End	2.3	2.1	2.3	1.5

Data on radioactive half-lives, decay products, relative natural abundances of isotopes, and effective cross sections for other isotopes were taken directly from the appropriate ORIGEN2 libraries. A portion of these data is reproduced in Appendix 1B, ORIGEN2 Data Libraries.

2.7.4 Results of Radiological Characterization

Thirty eight ORIGEN2 runs have been made with different materials being irradiated in different exposure zones of different reactor types. Cases that have been run are listed in Table 2.7.3. The output of these ORIGEN2 runs consists of the concentrations of an isotope in one kilogram of material. These concentrations are expressed in grams, curies, and watts. The overall photon spectra from the irradiated material are also an output. These results have been downloaded from the ORIGEN2 output to data files in the LWR Assemblies Data Base. Isotopic concentrations have been subjected to a set of cutoffs to limit the space required to store these induced radioactivity data. These cutoffs indicate that the concentration of an isotope at a given time after discharge is significant if the radioactivity (in curies) or the thermal power (in watts) resulting from the isotope is greater than 1 part in 100,000 of the total radioactivity or thermal power of all isotopes in the material at that time. Even if the isotope does not meet these criteria at time A, it is included in the induced radioactivity data in

the data base if it meets these criteria at some later time B. Thus, although 94Nb is not generally a significant contributor to the total radioactivity after short decay times, it is included in the data base for all times because of its significance at longer decay times. If either the radioactivity or the thermal output of an isotope causes it to be significant at a particular time, the curies, watts, and grams of the isotope are saved in a data file. The photon spectra from the ORIGEN2 runs are downloaded to another file without any cutoffs being imposed.

Two Radiological Description Reports are available from the data base. The first is the Material Report. This report gives the radiological description of a particular material that has been exposed to a specified (Standard or High) burnup in a given zone. The user must also specify the time after discharge for which he/she wants this radiological characterization. An example of the Material Report is given in Tables 2.7.4, 2.7.5, and 2.7.6 for Zircaloy-2, stainless steel 304 and Inconel-718, respectively. These reports are for materials exposed to standard burnup in the core of a BWR, 15 years after discharge.

Downloaded ORIGEN2 output is combined with the physical information given on page 2 of the Physical Description Report to produce a second Radiological Description Report, the Assembly Report. This report gives the radiological characteristics of the SFD hardware associated with a particular assembly type. This report is available for a particular piece of SFD hardware, for all parts within a specified zone, or for all the SFD hardware associated with the particular assembly. An example of the Radiological Description Report for a Babcock & Wilcox 15 x 15 Mark BZ fuel assembly is given in Table 2.7.7.

These results generally show that stainless steel and Zircaloy may be acceptable for low-level waste disposal, depending on the actual niobium content of the initial material and the regulatory status of Greater than Class C low-level wastes. Since 94Nb is the primary isotope that routinely exceeds the Class C limits in these materials,

niobium concentrations in the materials that are far below the ASTM limits may allow for disposal of SFD hardware made from these materials as Class C low-level wastes. Even if the niobium content is near the maximum limits, some SFD hardware may be acceptable for near surface burial if the NRC should approve a specific proposal for disposal of these Greater than Class C wastes. A particular problem with the 94Nb concentration lies in the difficulty of accurately observing and quantifying its decay. The primary photon associated with its decay has an energy of 871 keV. The photopeak associated with this relatively small decay is completely obscured by Compton backscattering from the high energy 60Co gamma rays. If SFD hardware made of stainless steel and Zircaloy is disposed of as low-level wastes, the heat output of the 60Co may create stiff surcharges at low-level waste disposal areas. As taken from Tables 2.7.4 and 2.7.5, the concentration of 60Co in Zircaloy-2 and stainless steel 304 is 880 and 88,000 Ci/m^3, respectively. The Class A limit for 60Co is 700 Ci/m^3. No Class B or Class C limit is specified for 60Co. The initial concentrations of nickel and niobium in Inconel alloys preclude the possibility that SFD hardware made from Inconel alloys will be acceptable for near-surface burial.

Experimental verification of these results is being pursued by a series of in-depth experiments currently being conducted by Pacific Northwest Laboratories (PNL) on pieces of SFD hardware with known histories. These experiments will determine the initial concentration of nitrogen, cobalt, nickel, and niobium, as well as the concentrations of the radionuclides present after irradiation. These results will provide more accurate input to ORIGEN2 and will allow verification of the flux factors and effective cross sections used in the radiological characterization of SFD hardware.

2.7.5 References for Section 2.7

Cooper 1986. Letter from R.G. Cooper, Babcock & Wilcox, to A. R. Irvine, Oak Ridge National Laboratory, dated August 28, 1986.

Croff 1978. A. G. Croff et al., Revised U-Pu Cycle PWR and BWR Models for the ORIGEN Computer Code, ORNL/TM-6051, September 1978.

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Table 2.7.1 Spent Fuel Disassembly Hardware for Different PWR Assembly Types. Listing by Array Design.

Babcock & Wilcox 15 x 15 Array Design	
Babcock & Wilcox 15 x 15 Mark B	35.6 kg
Babcock & Wilcox 15 x 15 Mark BZ	35.6 kg
Babcock & Wilcox 17 X 17 Array Design	
Babcock & Wilcox 17 x 17 Mark C	42.3 kg
Combustion Engineering 14 x 14 Array Design	
Combustion Engineering 14 x 14 Standard	29.8 kg
Combustion Engineering 14 x 14 Ft. Calhoun	27.6 kg
Advanced Nuclear Fuels 14 x 14 CE	33.3 kg
Westinghouse Electric 14 x 14 Model C	34.1 kg
Westinghouse Electric 14 x 14 Ft. Calhoun	N/A
Combustion Engineering 15 x 15 Array Design	
Combustion Engineering 15 x 15 Palisades	39.7 kg
Exxon/ANF 15 x 15 CE	31.4 kg
Combustion Engineering 16 x 16 Array Design	
Combustion Engineering 16 x 16 St. Lucie 2	34.8 kg
Combustion Engineering 16 x 16 Ark. Nucl. 2	40.1 kg
Combustion Engineering 16 x 16 San Onofre	42.6 kg
Combustion Engineering 16 x 16 System 80	44.0 kg
Westinghouse 14 x 14 Array Design	
Westinghouse 14 x 14 Std/SC	28.9 kg
Westinghouse 14 x 14 Std/ZCA	32.0 kg
Westinghouse 14 x 14 Std/ZCB	31.8 kg
Westinghouse 14 x 14 OFA	32.1 kg
Babcock & Wilcox 14 x 14 Ginna	N/A
Exxon/ANF 14 x 14 WE	28.4 kg
Exxon/ANF 14 x 14 Top Rod	24.6 kg
Westinghouse 15 x 15 Array Design	
Westinghouse 15 x 15 Std/SC	33.4 kg
Westinghouse 15 x 15 Std/ZC	35.8 kg
Westinghouse 15 x 15 OFA	32.6 kg
Babcock & Wilcox 15 x 15 St. Steel	28.4 kg
Exxon/ANF 15 x 15 WE	27.3 kg
Westinghouse 15 x 16 Array Design	
Westinghouse 15 x 16	N/A
Combustion Engineering 15 x 16 Yankee-Rowe	35.0 kg
Exxon/ANF 15 x 16 WE	30.4 kg
Westinghouse 17 x 17 Array Design	
Westinghouse 17 x 17 Std	29.6 kg
Westinghouse 17 x 17 OFA	32.3 kg
Westinghouse 17 x 17 Vantage 5	N/A
Westinghouse 17 x 17 X-long	N/A
Babcock & Wilcox 17 x 17 Mark BW	N/A
Exxon/ANF 17 x 17 WE	34.6 kg

Table 2.7.2 Spent Fuel Disassembly Hardware for Different BWR Assembly Types. Listing by Array Design.

Advanced Nuclear Fuels 9 x 9 Array Design	
Exxon/ANF 9 x 9 JP-3	9.3 kg
Exxon/ANF 9 x 9 JP-4,5	9.3 kg
Allis Chalmers 10 x 10 Array Design	
Allis Chalmers 10 x 10	N/A
Exxon/ANF 10 x 10	16.8 kg
General Electric 6 x 6 Dresden-1 Array Design	
General Electric 6 x 6	N/A
Exxon/ANF 6 x 6 GE	9.4 kg
United Nuclear 6 x 6	N/A
General Electric 6 x 6 Hum. Bay Array Design	
General Electric 6 x 6	N/A
Exxon/ANF 6 x 6	N/A
General Electric 7 x 7 Array Design	
General Electric 7 x 7 /2,3:V.1	8.0 kg
General Electric 7 x 7 /2,3:V.2	8.0 kg
General Electric 7 x 7 /4,5	8.0 kg
Exxon/ANF 7 x 7 GE	13.6 kg
General Electric 8 x 8 Array Design	
General Electric 8 x 8 /2,3	N/A
General Electric 8 x 8 /4-6:V.1	13.1 kg
General Electric 8 x 8 /4-6:V.2	14.8 kg
Exxon/ANF 8 x 8 JP-3	8.1 kg
Exxon/ANF 8 x 8 JP-4,5	9.0 kg
General Electric 9 x 9 Array Design	
General Electric 9 x 9	N/A
Exxon/ANF 9 x 9 Big Rock Point	N/A
General Electric 11 x 11 Array Design	
General Electric 11 x 11	N/A
Exxon/ANF 11 x 11	10.5 kg
Westinghouse 8 x 8 Array Design	
Westinghouse 8 x 8 QUAD+	N/A

Table 2.7.3 ORIGEN2 Runs Made for SFD Hardware Characterization.

Reactor Type	Material	Neutron Exposure Zone	Burnup
PWR	Zircaloy-4	Top End	Std.,High
PWR	Zircaloy-4	Gas Plenum	Std.,High
PWR	Zircaloy-4	Incore	Std.,High
PWR	Inconel 625	Incore	Std.,High
PWR	Inconel 718	Top End	Std.,High
PWR	Inconel 718	Gas Plenum	Std.,High
PWR	Inconel 718	Incore	Std.,High
PWR	Inconel 718	Bottom End	Std.,High
PWR	Inconel X-750	Top End	Std.,High
PWR	Inconel X-750	Incore	Std.,High
PWR	St. Steel 304	Top End	Std.,High
PWR	St. Steel 304	Incore	Std.,High
PWR	St. Steel 304	Bottom End	Std.,High
BWR	Zircaloy-2	Top End	Std.
BWR	Zircaloy-2	Incore	Std.
BWR	Zircaloy-2	Bottom End	Std.
BWR	Zircaloy-4	Top End	Std.
BWR	Zircaloy-4	Incore	Std.
BWR	Inconel 718	Top End	Std.
BWR	Inconel 718	Incore	Std.
BWR	Inconel X-750	Top End	Std.
BWR	Inconel X-750	Incore	Std.
BWR	St. Steel 304	Top End	Std.
BWR	St. Steel 304	Incore	Std.
BWR	St. Steel 304	Bottom End	Std.

Table 2.7.4 Material Radiological Description Report for Zircaloy-2.

Radiological Report (Curies)
for
Zircaloy-2

Boiling Water Reactor
In Core Zone
Standard (27,500) 15 Years After Discharge

ALL VALUES ARE PER KILOGRAM OF IRRADIATED MATERIAL

Isotope	Curies
C 14	1.500E-03
Fe 55	1.256E-02
Co 60	1.357E-01
Ni 59	2.279E-04
Ni 63	3.123E-02
Sr 90	3.184E-06
Y 90	3.185E-06
Zr 93	5.915E-04
Nb 93m	3.242E-04
Nb 94	1.768E-04
Tc 99	1.702E-08
Sn121m	2.010E-03
Sb125	1.378E-01
Te125m	3.361E-02

Table 2.7.5 Material Radiological Description Report for St. Steel 304.

Radiological Report (Curies)
for
Stainless Steel 304

Boiling Water Reactor
In Core Zone
Standard (27,500) 15 Years After Discharge

ALL VALUES ARE PER KILOGRAM OF IRRADIATED MATERIAL

Isotope	Curies
Be 10	2.304E-09
C 14	2.436E-02
Cl 36	9.060E-09
Fe 55	5.699E+00
Co 60	1.094E+01
Ni 59	4.065E-02
Ni 63	5.565E+00
Zr 93	8.639E-10
Nb 93m	4.735E-10
Nb 94	2.122E-04

Table 2.7.6 Material Radiological Description Report for Inconel-718.

Radiological Report (Curies)
for
Inconel-718

Boiling Water Reactor
In Core Zone
Standard (27,500) 15 Years After Discharge

ALL VALUES ARE PER KILOGRAM OF IRRADIATED MATERIAL

Isotope	Curies
Be 10	1.152E-09
C 14	2.436E-02
Fe 55	1.712E+00
Co 60	6.394E+01
Ni 59	2.368E-01
Ni 63	3.242E+01
Zr 93	4.197E-07
Nb 93m	2.300E-07
Nb 94	9.808E-02
Mo 93	1.895E-03
Tc 99	3.586E-04

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark BZ Fuel Assembly.

Spent Fuel Disassembly Hardware — Page: 1
for a
Babcock & Wilcox 15 X 15 Mark BZ

Top Zone — Standard Burnup — 10 years after discharge — Units are Curies

Component Name	Inconel-718	St.Steel 304	Isotope Total
HOLDDOWN SPRING	(1.8000 kg)		
C 14	2.146E-03		2.146E-03
Fe 55	1.501E-01		1.501E-01
Co 60	8.573E+00		8.573E+00
Ni 59	2.882E-02		2.882E-02
Ni 63	3.271E+00		3.271E+00
Zr 93	1.216E-08		1.216E-08
Nb 93m	5.125E-09		5.125E-09
Nb 94	3.289E-03		3.289E-03
Mo 93	3.825E-05		3.825E-05
Tc 99	1.780E-06		1.780E-06
	1.202E+01		
SPRING RETAINER	(0.9100 kg)		
Be 10		3.180E-11	3.180E-11
C 14		1.085E-03	1.085E-03
Mn 54		1.681E-04	1.681E-04
Fe 55		2.423E-01	2.423E-01
Co 60		7.378E-01	7.378E-01
Ni 59		2.499E-03	2.499E-03
Ni 63		2.836E-01	2.836E-01
Zr 93		1.059E-11	1.059E-11
Nb 93m		4.463E-12	4.463E-12
Nb 94		2.996E-06	2.996E-06
		1.267E+00	
TOP NOZZLE	(7.4800 kg)		
Be 10		2.614E-10	2.614E-10
C 14		8.916E-03	8.916E-03
Mn 54		1.382E-03	1.382E-03
Fe 55		1.992E+00	1.992E+00
Co 60		6.065E+00	6.065E+00
Ni 59		2.054E-02	2.054E-02
Ni 63		2.332E+00	2.332E+00
Zr 93		8.707E-11	8.707E-11
Nb 93m		3.668E-11	3.668E-11
Nb 94		2.462E-05	2.462E-05
		1.042E+01	

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark BZ Fuel Assembly. (cont.)

Spent Fuel Disassembly Hardware for a Babcock & Wilcox 15 X 15 Mark BZ

Page: 2

Top Zone — Standard Burnup — 10 years after discharge — Units are Curies

TOP NOZZLE	Inconel-718	St.Steel 304	Isotope Total
UPPER NUT	(0.5100 kg)		
Be 10		1.782E-11	1.782E-11
C 14		6.079E-04	6.079E-04
Mn 54		9.420E-05	9.420E-05
Fe 55		1.358E-01	1.358E-01
Co 60		4.135E-01	4.135E-01
Ni 59		1.400E-03	1.400E-03
Ni 63		1.590E-01	1.590E-01
Zr 93		5.936E-12	5.936E-12
Nb 93m		2.501E-12	2.501E-12
Nb 94		1.679E-06	1.679E-06
		7.104E-01	
UPPER END PLUG	(0.0600 kg)		
Be 10		2.096E-12	2.096E-12
C 14		7.152E-05	7.152E-05
Mn 54		1.108E-05	1.108E-05
Fe 55		1.598E-02	1.598E-02
Co 60		4.865E-02	4.865E-02
Ni 59		1.648E-04	1.648E-04
Ni 63		1.870E-02	1.870E-02
Zr 93		6.984E-13	6.984E-13
Nb 93m		2.942E-13	2.942E-13
Nb 94		1.975E-07	1.975E-07
		8.357E-02	

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark BZ Fuel Assembly. (cont.)

Spent Fuel Disassembly Hardware Page: 3
for a
Babcock & Wilcox 15 X 15 Mark BZ

Top Zone Standard Burnup 10 years after discharge Units are Curies

Total zone weight: 10.760000 kg. Total zone volume: 0.0013370 cu m.

Top Zone Totals by isotope:

Isotope	Curies	Curies/kg	Curies/cu m	Class C Limit	Ratio
Be 10	3.131E-10	2.910E-11	2.342E-07		
C 14	1.283E-02	1.192E-03	9.596E+00	80.00	1.200E-01
Mn 54	1.655E-03	1.538E-04	1.238E+00		
Fe 55	2.536E+00	2.357E-01	1.897E+03		
Co 60	1.584E+01	1.472E+00	1.185E+04		
Ni 59	5.342E-02	4.965E-03	3.996E+01	220.00	1.816E-01
Ni 63	6.065E+00	5.637E-01	4.536E+03	7000.00	6.480E-01
Zr 93	1.227E-08	1.140E-09	9.177E-06		
Nb 93m	5.169E-09	4.804E-10	3.866E-06		
Nb 94	3.319E-03	3.085E-04	2.482E+00	0.20	1.241E+01
Mo 93	3.825E-05	3.555E-06	2.861E-02		
Tc 99	1.780E-06	1.654E-07	1.331E-03		

Top Zone Totals by material:

Inconel-718	St.Steel 304
1.202E+01	1.248E+01

Top Zone Totals by material in kilograms:

Inconel-718	St.Steel 304
1.8000	8.9600

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark BZ Fuel Assembly. (cont.)

Spent Fuel Disassembly Hardware Page: 4
for a
Babcock & Wilcox 15 X 15 Mark BZ

Gas Plenum Zone Standard Burnup 10 years after discharge Units are Curies

Component Name	Inconel-718	Isotope Total
SPACER-PLENUM	(1.0400 kg)	
C 14	1.369E-02	1.369E-02
Fe 55	6.541E-01	6.541E-01
Co 60	5.195E+01	5.195E+01
Ni 59	1.805E-01	1.805E-01
Ni 63	2.056E+01	2.056E+01
Zr 93	5.296E-08	5.296E-08
Nb 93m	2.231E-08	2.231E-08
Nb 94	1.873E-02	1.873E-02
Mo 93	1.667E-04	1.667E-04
Tc 99	7.698E-06	7.698E-06
	7.338E+01	

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark BZ Fuel Assembly. (cont.)

Spent Fuel Disassembly Hardware Page: 5
for a
Babcock & Wilcox 15 X 15 Mark BZ

Gas Plenum Zone Standard Burnup 10 years after discharge Units are Curies

Total zone weight: 1.040000 kg. Total zone volume: 0.0001270 cu m.

Gas Plenum Zone Totals by isotope:

Isotope	Curies	Curies/kg	Curies/cu m	Class C Limit	Ratio
C 14	1.369E-02	1.316E-02	1.078E+02	80.00	1.348E+00
Fe 55	6.541E-01	6.289E-01	5.151E+03		
Co 60	5.195E+01	4.995E+01	4.091E+05		
Ni 59	1.805E-01	1.736E-01	1.421E+03	220.00	6.461E+00
Ni 63	2.056E+01	1.977E+01	1.619E+05	7000.00	2.313E+01
Zr 93	5.296E-08	5.092E-08	4.171E-04		
Nb 93m	2.231E-08	2.145E-08	1.757E-04		
Nb 94	1.873E-02	1.801E-02	1.475E+02	0.20	7.375E+02
Mo 93	1.667E-04	1.603E-04	1.313E+00		
Tc 99	7.698E-06	7.402E-06	6.062E-02		

Gas Plenum Zone Totals by material:

Inconel-718

7.338E+01

Gas Plenum Zone Totals by material in kilograms:

Inconel-718

1.0400

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark BZ Fuel Assembly. (cont.)

Spent Fuel Disassembly Hardware — Page: 6
for a
Babcock & Wilcox 15 X 15 Mark BZ

In Core Zone — Standard Burnup — 10 years after discharge — Units are Curies

Component Name	Zircaloy-4	Isotope Total
GRID SUPPORTS	(0.6400 kg)	
C 14	9.901E-04	9.901E-04
Mn 54	3.475E-05	3.475E-05
Fe 55	5.004E-02	5.004E-02
Co 60	1.784E-01	1.784E-01
Ni 59	6.733E-06	6.733E-06
Ni 63	8.819E-04	8.819E-04
Sr 90	3.114E-06	3.114E-06
Y 90	3.115E-06	3.115E-06
Zr 93	3.558E-04	3.558E-04
Nb 93m	1.500E-04	1.500E-04
Nb 94	1.357E-04	1.357E-04
Sn119m	4.768E-04	4.768E-04
Sn121m	1.410E-03	1.410E-03
Sb125	3.396E-01	3.396E-01
Te125m	8.288E-02	8.288E-02
	6.554E-01	
GUIDE TUBES	(8.0000 kg)	
C 14	1.238E-02	1.238E-02
Mn 54	4.344E-04	4.344E-04
Fe 55	6.254E-01	6.254E-01
Co 60	2.230E+00	2.230E+00
Ni 59	8.416E-05	8.416E-05
Ni 63	1.102E-02	1.102E-02
Sr 90	3.893E-05	3.893E-05
Y 90	3.894E-05	3.894E-05
Zr 93	4.448E-03	4.448E-03
Nb 93m	1.874E-03	1.874E-03
Nb 94	1.697E-03	1.697E-03
Sn119m	5.960E-03	5.960E-03
Sn121m	1.762E-02	1.762E-02
Sb125	4.245E+00	4.245E+00
Te125m	1.036E+00	1.036E+00
	8.192E+00	

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark BZ Fuel Assembly. (cont.)

Spent Fuel Disassembly Hardware Page: 7
for a
Babcock & Wilcox 15 X 15 Mark BZ

In Core Zone Standard Burnup 10 years after discharge Units are Curies

Component Name	Zircaloy-4	Isotope Total
INSTRUMENT TUBE	(0.6400 kg)	
C 14	9.901E-04	9.901E-04
Mn 54	3.475E-05	3.475E-05
Fe 55	5.004E-02	5.004E-02
Co 60	1.784E-01	1.784E-01
Ni 59	6.733E-06	6.733E-06
Ni 63	8.819E-04	8.819E-04
Sr 90	3.114E-06	3.114E-06
Y 90	3.115E-06	3.115E-06
Zr 93	3.558E-04	3.558E-04
Nb 93m	1.500E-04	1.500E-04
Nb 94	1.357E-04	1.357E-04
Sn119m	4.768E-04	4.768E-04
Sn121m	1.410E-03	1.410E-03
Sb125	3.396E-01	3.396E-01
Te125m	8.288E-02	8.288E-02
	6.554E-01	
SPACER-INCORE	(4.9000 kg)	
C 14	7.580E-03	7.580E-03
Mn 54	2.661E-04	2.661E-04
Fe 55	3.831E-01	3.831E-01
Co 60	1.366E+00	1.366E+00
Ni 59	5.155E-05	5.155E-05
Ni 63	6.752E-03	6.752E-03
Sr 90	2.384E-05	2.384E-05
Y 90	2.385E-05	2.385E-05
Zr 93	2.724E-03	2.724E-03
Nb 93m	1.148E-03	1.148E-03
Nb 94	1.039E-03	1.039E-03
Sn119m	3.650E-03	3.650E-03
Sn121m	1.079E-02	1.079E-02
Sb125	2.600E+00	2.600E+00
Te125m	6.345E-01	6.345E-01
	5.018E+00	

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark BZ Fuel Assembly. (cont.)

Spent Fuel Disassembly Hardware Page: 8
for a
Babcock & Wilcox 15 X 15 Mark BZ

In Core Zone Standard Burnup 10 years after discharge Units are Curies

Total zone weight: 14.180000 kg. Total zone volume: 0.0021616 cu m.

In Core Zone Totals by isotope:

Isotope	Curies	Curies/kg	Curies/cu m	Class C Limit	Ratio
C 14	2.194E-02	1.547E-03	1.015E+01	80.00	1.269E-01
Mn 54	7.699E-04	5.429E-05	3.562E-01		
Fe 55	1.108E+00	7.814E-02	5.126E+02		
Co 60	3.952E+00	2.787E-01	1.828E+03		
Ni 59	1.492E-04	1.052E-05	6.902E-02	220.00	3.137E-04
Ni 63	1.953E-02	1.377E-03	9.035E+00	7000.00	1.291E-03
Sr 90	6.899E-05	4.865E-06	3.192E-02		
Y 90	6.901E-05	4.867E-06	3.193E-02		
Zr 93	7.884E-03	5.560E-04	3.647E+00		
Nb 93m	3.322E-03	2.343E-04	1.537E+00		
Nb 94	3.008E-03	2.121E-04	1.392E+00	0.20	6.958E+00
Sn119m	1.056E-02	7.447E-04	4.885E+00		
Sn121m	3.123E-02	2.202E-03	1.445E+01		
Sb125	7.525E+00	5.307E-01	3.481E+03		
Te125m	1.836E+00	1.295E-01	8.494E+02		

In Core Zone Totals by material:

Zircaloy-4

1.452E+01

In Core Zone Totals by material in kilograms:

Zircaloy-4

14.1800

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark BZ Fuel Assembly. (cont.)

Spent Fuel Disassembly Hardware Page: 9
for a
Babcock & Wilcox 15 X 15 Mark BZ

Bottom Zone Standard Burnup 10 years after discharge Units are Curies

Component Name	Inconel-718	St.Steel 304	Isotope Total
SPACER-BOTTOM	(1.3000 kg)		
C 14	8.869E-03		8.869E-03
Fe 55	6.207E-01		6.207E-01
Co 60	3.622E+01		3.622E+01
Ni 59	1.181E-01		1.181E-01
Ni 63	1.345E+01		1.345E+01
Zr 93	5.027E-08		5.027E-08
Nb 93m	2.118E-08		2.118E-08
Nb 94	1.440E-02		1.440E-02
Mo 93	1.582E-04		1.582E-04
Tc 99	7.320E-06		7.320E-06
	5.043E+01		
BOTTOM NOZZLE	(8.1600 kg)		
Be 10		1.633E-09	1.633E-09
C 14		5.567E-02	5.567E-02
Mn 54		8.625E-03	8.625E-03
Fe 55		1.244E+01	1.244E+01
Co 60		3.871E+01	3.871E+01
Ni 59		1.272E-01	1.272E-01
Ni 63		1.449E+01	1.449E+01
Zr 93		5.434E-10	5.434E-10
Nb 93m		2.290E-10	2.290E-10
Nb 94		1.630E-04	1.630E-04
		6.583E+01	
LOWER NUT	(0.1500 kg)		
Be 10		3.001E-11	3.001E-11
C 14		1.023E-03	1.023E-03
Mn 54		1.585E-04	1.585E-04
Fe 55		2.287E-01	2.287E-01
Co 60		7.116E-01	7.116E-01
Ni 59		2.338E-03	2.338E-03
Ni 63		2.664E-01	2.664E-01
Zr 93		9.988E-12	9.988E-12
Nb 93m		4.209E-12	4.209E-12
Nb 94		2.995E-06	2.995E-06
		1.210E+00	

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark B2 Fuel Assembly. (cont.)

Spent Fuel Disassembly Hardware Page: 10
for a
Babcock & Wilcox 15 X 15 Mark B2

Bottom Zone Standard Burnup 10 years after discharge Units are Curies

Total zone weight: 9.610000 kg. Total zone volume: 0.0011949 cu m.

Bottom Zone Totals by isotope:

Isotope	Curies	Curies/kg	Curies/cu m	Class C Limit	Ratio
Be 10	1.663E-09	1.730E-10	1.392E-06		
C 14	6.556E-02	6.822E-03	5.487E+01	80.00	6.858E-01
Mn 54	8.784E-03	9.140E-04	7.351E+00		
Fe 55	1.329E+01	1.383E+00	1.112E+04		
Co 60	7.564E+01	7.871E+00	6.330E+04		
Ni 59	2.476E-01	2.576E-02	2.072E+02	220.00	9.419E-01
Ni 63	2.821E+01	2.935E+00	2.361E+04	7000.00	3.373E+00
Zr 93	5.082E-08	5.288E-09	4.253E-05		
Nb 93m	2.141E-08	2.228E-09	1.792E-05		
Nb 94	1.456E-02	1.515E-03	1.219E+01	0.20	6.093E+01
Mo 93	1.582E-04	1.646E-05	1.324E-01		
Tc 99	7.320E-06	7.617E-07	6.126E-03		

Bottom Zone Totals by material:

Inconel-718	St.Steel 304
5.043E+01	6.704E+01

Bottom Zone Totals by material in kilograms:

Inconel-718	St.Steel 304
1.3000	8.3100

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark BZ Fuel Assembly. (cont.)

Spent Fuel Disassembly Hardware Page: 11
for a
Babcock & Wilcox 15 X 15 Mark BZ

All Zones Standard Burnup 10 years after discharge Units are Curies

Total zones weight: 35.590000 kg. Total zones volume: 0.0048204 cu m.

Totals by isotope for all zones:

Isotope	Curies	Curies/kg	Curies/cu m	Class C Limit	Ratio
Be 10	1.976E-09	5.552E-11	4.099E-07		
C 14	1.140E-01	3.203E-03	2.365E+01	80.00	2.956E-01
Mn 54	1.121E-02	3.150E-04	2.326E+00		
Fe 55	1.759E+01	4.942E-01	3.649E+03		
Co 60	1.474E+02	4.142E+00	3.058E+04		
Ni 59	4.816E-01	1.353E-02	9.991E+01	220.00	4.541E-01
Ni 63	5.485E+01	1.541E+00	1.138E+04	7000.00	1.626E+00
Sr 90	6.899E-05	1.938E-06	1.431E-02		
Y 90	6.901E-05	1.939E-06	1.432E-02		
Zr 93	7.884E-03	2.215E-04	1.636E+00		
Nb 93m	3.322E-03	9.334E-05	6.891E-01		
Nb 94	3.962E-02	1.113E-03	8.219E+00	0.20	4.110E+01
Mo 93	3.631E-04	1.020E-05	7.532E-02		
Tc 99	1.680E-05	4.720E-07	3.485E-03		
Sn119m	1.056E-02	2.967E-04	2.191E+00		
Sn121m	3.123E-02	8.775E-04	6.479E+00		
Sb125	7.525E+00	2.114E-01	1.561E+03		
Te125m	1.836E+00	5.159E-02	3.809E+02		

Assembly totals by material:

Zircaloy-4	Inconel-718	St.Steel 304
1.452E+01	1.358E+02	7.952E+01

Assembly grand total in Curies	Curies/kg	Curies/cu m
2.298E+02	6.458E+00	4.768E+04

*Some radiological data based on a slightly different elemental composition than actual material.

2.8 NONFUEL ASSEMBLY HARDWARE

2.8.1 Overview

This category deals with reactor hardware that is not necessarily tied in with fuel assemblies on a one-to-one basis, as SFD hardware is. Nonfuel Assembly (NFA) hardware is generally used within or between assemblies but is not permanently attached to an assembly. These NFA hardware components are usually retired from service on a schedule that is different from that of the fuel assemblies. Physical and radiological characterization of these components is being done as part of the LWR NFA Hardware Data Base. Appendix 2E contains physical description reports for pieces of NFA hardware as described by the vendors. Table 2.8.1 gives an example of a physical description report from the LWR NFA Hardware Data Base. Appendix 2F is the user's guide to the LWR NFA Hardware Data Base. Because NFA hardware may remain in the reactor for many cycles, radiological characterization of the NFA hardware will require somewhat different treatment than that of SFD hardware. The methodology of using four neutron exposure zones (see Section 2.7.3) is continued with NFA hardware; likewise, the structure of the data file describing the materials used, weight, etc., remains essentially unchanged. Generating a different set of ORIGEN2 outputs is required.

The major contributors to this waste category are BWR fuel channels, BWR control blades, PWR control rods, and PWR burnable poison assemblies. Other contributors to NFA hardware are neutron sources, in-core instrumentation, and guide-tube thimbles or orifice rods.

Historically, BWR fuel channels have been discharged on a one-to-one basis with the BWR assemblies. They are normally attached to the assemblies and could be considered as SFD hardware except that actual fuel disassembly is not required.

The BWR control blades (cruciforms) and PWR control rods also contribute significantly to the volume of NFA hardware waste. The BWR cruciforms have an estimated lifetime of 3 to 25 years, depending on their service mode. The PWR control rods have an estimated lifetime of

up to the full reactor lifetime because they are normally in a withdrawn position. In the past, they have been replaced halfway through reactor lifetime.

Burnable poison assemblies, especially from B&W reactors, are the third large contributor to the NFA hardware waste category. They are typically used for only one cycle and then changed out. Since they are used in the core of the reactor, their levels of neutron activation are high.

Guide-tube thimbles or orifice rods should be a minor contributor to NFA hardware quantities. Typically stainless steel tubes designed to inhibit the flow of water through otherwise empty guide tubes, these rods are generally outside the core zone. They are hollow and short; so they are relatively lightweight. They are not routinely replaced except when damaged; so the number being changed out is small.

In-core power instrumentation should also be a minor contributor to NFA hardware quantities. The detectors/emitters on these pieces of hardware are in the reactor core and may be highly activated, but the majority of the mass of these pieces is outside the in-core zone and may be far less activated.

Neutron sources may also pose a problem from the standpoint of TRU wastes. Primary neutron sources are typically polonium- or plutonium-beryllium alloys, or californium. Secondary neutron sources are typically antimony-beryllium alloys. Quantities of these wastes should be small.

2.8.2 Babcock and Wilcox Non-Fuel Assembly Hardware

2.8.2.1 Control Rod Assemblies

Control rod assemblies consist of 16 individual rods with their upper ends fastened to a spider assembly. The control rod drive mechanism engages the spider assembly to withdraw and position the control rod assembly. The spider assembly consists of seven pounds of CF3M, 304, and 316 stainless steels. The control rods are about 13 feet long and 0.440 inch in diameter and the cladding is 304 stainless steel

with 304 or 308 stainless steel end plugs. The thickness is not given, but the weight of the 16 tubes with lower end plugs is 18.38 pounds. The nuts, upper end plugs, and spring spacers weigh 7.5 pounds. The Ag-In-Cd alloy weighs 95 pounds, and the total weight of the assembly is 130 pounds.

Axial power shaping assemblies have a similar spider assembly that weighs 7.8 pounds. The axial power shaping rods are the same length and diameter as control rods and use 304 stainless steel for cladding, whereas end plugs, intermediate plugs, and nuts may be 304 or 308 stainless steel. The stainless steel parts weigh 24.7 pounds. The Ag-In-Cd absorber must be shorter, is assumed to be at the bottom of the rod, and weighs 23.4 pounds. The overall assembly weight is 57 pounds.

Gray axial power shaping rod assemblies also have a similar spider assembly that weighs 7.5 pounds. The gray axial power shaping rods are 155.56 inches long and 0.440 inch in diameter and use 304 stainless steel for cladding while end plugs, intermediate plugs, and nuts may be of 304 or 308 stainless steel. The stainless steel parts weigh 29.8 pounds. The absorber is Inconel 600. It is 63.25 inches long and weighs 33.8 pounds. The total weight of the assembly is 71 pounds. The overall length of the assembly is 159.75 inches.

2.8.2.2 Neutron Sources

Primary neutron source clusters

The description is incomplete. Weights are given for three of the ten components, and no overall weight is given. The shroud tube, intermediate plug, and lower end plug are made of 304 stainless steel and weigh one pound. No description of the source is given except that it is Am-Be-Cu and is B&W proprietary. It is not known if it is part of a control assembly or an orifice rod assembly.

Regenerative neutron source clusters

These clusters consist of a coupling spider assembly and eight rods. There is no mention of orifice plugs for the other eight locations. The spider assembly is made of CF3M, 304, and 316 stainless steel and weighs 7.8 pounds. The clad and end plugs for the rods are

made of 304 stainless steel, but the weights are not given. The length of the source rods is about 11 feet 8 inches and the diameter is 0.440 inch. By comparison to other types of rods, it may be assumed that the weight of stainless steel is about 12 pounds; thus, the antimony-beryllium source weighs 26.4 pounds. The source composition is B&W proprietary. The total weight of the cluster is 46.3 pounds.

2.8.2.3 Burnable Poison Assemblies

The spider for these assemblies is made of CF3M, 304, and 316 stainless steel and weighs 7.8 pounds. The 16 burnable poison rods use Zircaloy-4 for cladding, end fittings, and the nuts to fasten the rods to the spider. Each rod is about 12 feet 6 inches long and 0.430 inch in diameter. The hold-down spring is 302 or 304 stainless steel. The burnable poison is a B&W proprietary mixture of Al_2O_3 and B_4C. The poison weighs 20.8 pounds, the Zircaloy 25.2 pounds, and the springs 2.1 pounds. The overall weight of the assembly is 57 pounds.

2.8.2.4 Orifice Rod Assembly

The spider assembly is made of CF3M and 304 stainless steel and weighs 7.8 pounds. The orifice rods are about 12 inches long and 0.480 inch in diameter. The 16 rods and associated nuts weigh 7.7 pounds. The rods are made of 304 stainless steel whereas the nuts are made of 304 or 308 stainless steel. B&W's submittal indicates that this assembly should have "orifice plugs" made of 304 stainless steel but does not assign them a weight. The total assembly weight is given as 15.8 pounds; this is 0.3 pound heavier than the sum of the other components.

2.8.2.5 In-Core Instrumentation

B&W provides in-core instrumentation but none was described in the information supplied.

2.8.3 Combustion Engineering Non-Fuel Assembly Hardware

2.8.3.1 Control Element Assemblies

Control element assemblies (CEA's) provide a means of controlling core reactivity. CE designed control element assemblies consist of

tubes or "fingers" filled with neutron absorbing materials. The geometry of the fingers allows them to fit inside the guide tubes of fuel assemblies. During normal operation, the fingers are fully withdrawn from the fueled zone into the upper guide structure of the reactor vessel. In this position the lower tips of the CEA fingers are approximately two inches above the fueled zone. It is sometimes necessary to control the shape of the power distribution by using the CEA's. This is accomplished by inserting designated blanks of CEA's several inches into the fueled zone. This technique is used for relatively short-term power shaping. CEA's that have reached the end of their usable life (4000 FPD) are housed in spent fuel assemblies which are in the utility's spent fuel pool.

Palisades is a special case; the control assemblies are in the form of cruciform blades and are expected to last for the lifetime of the unit. There are 45 assemblies in the reactor with an overall length of 151 inches and a weight of 214 pounds. Each of the four blades extends 6.125 inches from the center line and ranges from 0.32 inch thick at the root to 0.18 inch thick at the edge. The absorber is Ag-In-Cd clad in 304 stainless steel. The control rod drive mechanism engages a hanger to withdraw and position the CEA. The hanger is made of 304 and 308 stainless steel. The stainless steel weighs 62.2 pounds and the absorber 151.8 pounds.

For the remaining reactors, the CEA's consist of four or five fingers fastened to a spider at their upper ends. The control rod drive mechanism engages the spider assembly to withdraw and position the CEA. The three Palo Verde reactors also have 48 CEA's that consist of twelve fingers and their associated spider. There are eight variations of the number of CEA's in the 14 reactors built by CE involving totals of 45, 81, 89, and 91. Of these totals, 4, 8, 12, or 13 CEA's may have part-length rods with the remainder having full-length rods. Six of the reactors use 14 x 14 fuel. The CEA's for these reactors use Inconel 625 clad control rods 0.948 inch in diameter with a 0.040-inch wall thickness. The pellet diameter is 0.86 inch in the CEA's for all six

reactors. One reactor uses CEA's 152 inches long; this reactor uses 128 inches of B_4C in all five fingers of the full-length CEA's and the center finger of the part-length CEA's. The four outside fingers of the part-length CEA's have 32 inches of B_4C. The full-length assembly contains 7.5 pounds of stainless steel, 34 pounds of Inconel, and 25 pounds of B_4C. The other five reactors use 161 inch CEA's and 134 inches of B_4C in the center fingers of the full-length rods. The outside fingers have 2.6 inches of Inconel on the tip, 8.0 inches of Ag-In-Cd alloy and 124 inches of B_4C.

The part-length CEA's have several variations in the number and kinds of control material that they employ. Arrangements include one B_4C rod and four stainless steel rods; three stainless steel rods and two rods with 8 inches of Ag-In-Cd and 124 inches of B_4C; one rod of Al_2O_3 and four rods of silver alloy and B_4C; one rod of Al_2O_3, two rods of silver alloy and B_4C, and two rods having 10 inches of stainless steel and 124 inches of Al_2O_3; and one rod of B_4C, two rods of silver alloy and B_4C, and two rods of stainless steel and Al_2O_3. The full-length assemblies contain 7.5 pounds of stainless steel, 39 pounds of Inconel, 24.2 pounds of B_4C, and 6.1 pounds of silver alloy. The total weight of CEA's for 14 x 14 fuel ranges from 63 to 105 pounds.

The CEA's for the eight reactors with 16 x 16 fuel are also of varying length - one at 162.8 inches, one at 180.8 inches and three each at 181.3 and 253.0 inches. The 253-inch CEA's and four of the 181.3-inch CEA's have only four fingers, but all others have five fingers. All are clad in Inconel 625. The cladding is 0.816 inch in diameter with a 0.035 inch wall thickness. The poison materials are 0.737 inch in diameter, although the Ag-In-Cd alloy may be annular. The poison arrangement also varies. The full-length control rods may be all B_4C; 12.5 inches of Ag-In-Cd, or 9.2 inches of Inconel 625 may be substituted for some of the B_4C near the tip. In all cases, a 0.5-inch stainless steel spacer is used above and below the poison and to separate the poisons when there are two. The end plug and spacer hold the poison 1.125 to 1.25 inches from the tip. The part-length CEA's may

have 68.5 to 75 inches of Inconel 625 near the tip and 14 to 16 inches of B_4C above it or 16 inches of B_4C near the tip with 75 inches of Inconel above it. The plenum springs are 302 stainless steel. The 12-finger CEA's at the three Palo Verde units have 19.5 pounds of stainless steel, 123 pounds of Inconel, and 49.7 pounds of B_4C. The total weight is 192.2 pounds. The full-length CEA's for the other five reactors have 8 pounds of stainless steel, 31.8 to 37.1 pounds of Inconel, 17.4 to 19.3 pounds of B_4C, and 6.9 to 8.6 pounds of silver alloy. The total weight ranges from 65.8 to 72 pounds. The total weight of the part-length CEA's ranges from 83 to 95 pounds. The part-length CEA's consist of about 8 pounds of stainless steel, 2.0 to 2.3 pounds of B_4C, and the balance Inconel. It is assumed that through December 1987 CE has manufactured only the original set of CEA's.

2.8.3.2 Neutron Source Assemblies

Two neutron source assemblies, placed in guide tubes of perimeter assemblies on opposite sides of the core, are used in each core. Their life expectancy is 3500 FPD for the 16 x 16 fuel and 4000 FPD for the other fuel. They are stationary fixtures with an upper shoulder resting on a post of the fuel assembly and held down by a plunger and spring.

Palisades is again a special case and uses two startup sources and two sustaining sources. They are about 115 inches long, 0.34 inch in diameter, and clad with 304 stainless steel. The sustaining source is made of antimony-beryllium; it is 0.286 inch in diameter and 72 inches long. The startup source is the same size but consists of 12 inches of polonium-beryllium in the center with 30 inches of antimony-beryllium above and below. Each source assembly consists of 4.1 pounds of stainless steel and 0.2 pound of beryllium. The sustaining source contains 0.2 pound of antimony and nearly 0.1 pound of polonium. The total weight of each is 4.5 pounds.

The assemblies for 14 x 14 fuel and 16 x 16 fuel are quite similar. The assembly for 16 x 16 fuel is described as follows. The assembly consists of two subassemblies. The lower subassembly is 42.5 inches long and contains the sources, a tubular spacer at the bottom, and a

hold-down spring at the top. There are 15.65 inches of 0.654-inch-diameter antimony-beryllium pellets in the center with a plutonium-beryllium capsule 6.0 inches long and 0.654 inch in diameter above and below them. The upper subassembly consists of the upper fitting, a coupler to connect to the lower subassembly, and a tube of the proper length to center the lower assembly in the active zone of the core. The plunger, upper subassembly, cladding, and spacer for the lower subassembly are 316 stainless steel. The cladding and tubing diameter is 0.812 inch. The assemblies contain 7.2 pounds of stainless steel, 0.4 pound of beryllium, and 0.2 pound each of nickel-based alloy springs, plutonium, and antimony. The assembly for 14 x 14 fuel is 0.875 inch in diameter and about 20 to 30 percent heavier.

2.8.3.3 Burnable Poison Assemblies

Combustion Engineering uses integral burnable poisons; they do not manufacture burnable poison NFA Hardware.

2.8.3.4 Orifice Rod Assemblies

Combustion Engineering does not describe any Orifice Rod Assemblies in CEND-428.

2.8.3.5 In-Core Instrumentation

The in-core instrument assemblies are located strategically about the reactor core in positions not designated for control element assemblies. The emitters are rhodium attached to Inconel 600 lead wires surrounded by aluminum oxide insulator and sheathed in Inconel 600. The emitter is 0.018 inches in diameter and 15.75 inches long. The sheath is 0.064 inches in diameter and 30 to 116 feet long. Four or five emitters, a calibration tube, a background detector, and an outlet thermocouple are enclosed in a housing tube which is 0.45 inch maximum diameter for Palo Verde and 0.35 inch for all other reactors. The data sheets state that the housing material is Inconel 600 for all reactors, however, it may be stainless steel for those reactors using 14x14 fuel.

The 128 to 150 inches of each assembly in the active core includes 0.01 pounds of rhodium and 0.1 to 0.3 pounds of Al_2O_3. The assemblies for 14x14 fuel have 0.3 to 0.4 pounds of Inconel and 2.0 to 2.9 pounds of stainless steel. The assemblies for 16x16 fuel have 1.7 to 3.1 pounds of Inconel and no stainless steel. There are 28 to 61 assemblies per reactor.

2.8.4 General Electric Nonfuel Assembly Hardware

Preliminary data of GE NFA hardware have been obtained from Safety Analysis Reports for selected reactors. Efforts to obtain more complete data are ongoing.

2.8.4.1 Control Element Assemblies

General Electric use cruciform blades made of stainless steel 304 with a boron carbide absorber contained in 84 stainless steel 304 tubes for control elements in their reactors. The boron carbide is packed into the tubes at about 70% of its theoretical density. The cruciform blades, which are 9.75 inches wide and have a control length of 143 inches, are positioned below the active fuel zone. The weights of the cruciforms have not been given in the Safety Analysis Reports, but estimates based on the materials of construction indicate that cruciforms weigh about 225 pounds. BWR cruciforms have an estimated lifetime of 3 to 25 years. The cruciforms have a unique radiological feature: their end bearings are made of an alloy high in cobalt (Stellite-3 or Haynes-25). The natural cobalt activates to ^{60}Co to a level high enough that these bearings are sometimes removed and packaged for use as ^{60}Co sources. The heat output from these bearings can present a distorted picture of the heat output from the cruciform as a whole.

2.8.4.2 Neutron Sources

General Electric uses five to seven antimony-beryllium neutron source rods per reactor. Each source rod consists of two irradiated antimony rods within a single beryllium cylinder. Both the antimony and beryllium are encased in stainless steel tubes. No weights or dimensions are available at this time.

2.8.4.3 Burnable Poison Assemblies

Temporary poison curtains made of borated (3800 to 5400 ppm boron) stainless steel sheets were used to control reactivity in the initial core of early reactors. These curtains, which were 141.25 inches long, 9.20 inches wide, and 0.0625 inch thick, were placed between fuel assemblies in water gaps without control rods. The weight is not given, but density considerations indicate a mass of about 25 pounds. Currently, Ge incorporates integral gadolinia poisons to provide this reactivity control.

2.8.4.4 Orifice Rod Assemblies

Orifice rod assemblies are not applicable to GE reactors.

2.8.4.5 In-Core Instrumentation

General Electric uses three types of in-core instrumentation: source range monitors, intermediate range monitors, and local power range monitors. Most reactors seem to have four source range monitors and four intermediate range monitors. The number of local power range monitors in dependent of the number of control-rod groups. One four-element local power range monitor is at the center of each four-control-rod group. No weights or dimensions are available at this time.

2.8.4.6 BWR Channels

BWR fuel channels for all BWR/2, 3, 4, and 5 reactors have a square cross section with a 5.278-inch inside width. The nominal length is 162 inches for BWR/2 and 3 reactors and 167 inches for BWR/4 and 5 reactors. Three channel thicknesses have been produced: 80, 100, and 120 mil. Estimates of the weights for these channels for BWR/4 and 5 reactors are 67, 83, and 100 pounds, respectively.

Recently, suggestions have been made that a BWR channel might be reused with a second fuel assembly. This approach could decrease the number of BWR channels by up to 50%; however, the channel would almost

surely have to be thicker. Thus the total mass of the channels might not be greatly affected. Higher burnup fuels may also reduce the utility of this approach.

2.8.5 Westinghouse Non-Fuel Assembly Hardware

2.8.5.1 Control Rod Assemblies

Control rod assemblies for reactors using 17 x 17 fuel consist of 24 individual rods fastened on their upper ends to a spider assembly. Reactors using 15 x 15 fuel have 20 control rods, and reactors using 14 x 14 fuel have 16. The control-rod-drive mechanism engages the spider assembly to withdraw and position the control rod assembly. The spider assembly consists of 4.8 to 7.6 pounds of 304 and 308 stainless steel and 1.65 pounds of Inconel 718 springs for use with 17 x 17 fuel. The assembly for 15 x 15 fuel uses 7.6 pounds of 304 and 308 stainless steel and 1.61 pounds of Inconel X-750. The assemblies for 14 x 14 fuel use 6.25 pounds of 304 and 308 stainless steel for the longer assemblies and 4.25 pounds for the short ones. All use 1.61 pounds of Inconel X-750.

The control rods for 17 x 17 fuel are 151.885 inches long and 0.385 inch in diameter. The cladding is 304 stainless steel with 308 stainless steel end plugs. The cladding is 0.0185 inch thick except for hybrid rods, which have a 0.038 inch thick cladding. The full-length rods contain 142 inches of Ag-In-Cd alloy or hafnium. Part-length rods contain 36 inches of Ag-In-Cd alloy with 106 inches of aluminum oxide spacer above the absorber. Hybrid rods contain 40 inches of Ag-In-Cd alloy with 102 inches of B_4C above it. The part-length and full-length rods contain hold-down springs of 302 stainless steel, whereas the hybrid rods may use Inconel 718. The cladding and end plugs for the hybrid assembly weigh 45 pounds, whereas for the other assemblies they weigh 24 or 25 pounds. The assembly of part-length rods contains 29 pounds of Ag-In-Cd, whereas the hybrid contains 25 pounds and that of the full-length contains 114 pounds. The hybrid assembly also contains about 14 pounds of B_4C. The hafnium version of the full-length assembly

contains 144 pounds of hafnium. The stainless steel springs weigh 1.2 pounds, whereas the Inconel springs weigh 0.6 pound. The total assembly weights range from 93 to 180 pounds. The overall length of the assemblies is 161 inches.

The control rods for 15 x 15 fuel are 150.92 to 152.75 inches long and 0.443 inch in diameter. The 304 stainless steel cladding is 0.020 inch thick and the end plugs are 308 stainless steel. The cladding and end plugs weigh 28 pounds. The absorber is 142 inches of Ag-In-Cd alloy and adds 129 pounds to the weight of the assembly. The hold-down springs in the rods are carbon steel and weigh 0.3 pound. The total assembly weight is 165 pounds and the overall length is 156.6 to 158.5 inches.

The control rods for 14 x 14 fuel are 120 to 153 inches long and 0.435 inch in diameter. The 304 stainless steel cladding is 0.0185 inch thick and the end plugs are 308 stainless steel. The cladding and end plugs weigh 18 to 21 pounds. The absorber is 118 to 142 inches of Ag-In-Cd weighing from 83 to 100 pounds. The hold-down springs in the longer rods are carbon steel weighing 0.3 pound whereas the short rods have 0.4 pound of Inconel X-750 springs. The total assembly weights are 109 to 128 pounds, and the overall length is 134 to 158 inches.

Through June 30, 1986, Westinghouse has manufactured 2,215 control rod assemblies compared to 18,032 fuel assemblies.

2.8.5.2 Neutron Sources

Primary Source Assemblies

The fuel assemblies that do not have control assemblies may be fitted with primary source assemblies, although presumably only one or two per reactor and then probably only for the first cycle or two. (Westinghouse has only manufactured 92, although they describe 106 configurations in WSTD-TME-148.) All these configurations have only one primary source rod. They indicate that six configurations have 20 thimble plugs but no other rods in addition to the source rod. There are 23 configurations with a primary source rod and one, three, or four

secondary source rods and 0, 12, or 16 burnable poison rods with the balance thimble plugs. All others had a primary source rod and 12 to 23 burnable poison rods, with the balance, if any, being thimble plugs. The overall weight ranges from 15 to 52 pounds. The 16 to 24 rods and plugs are attached to a spider pack or hold-down assembly that is held down by the same hold-down plate that keeps the fuel assemblies in place. The hold-down assemblies and spider packs consist of 3.7 to 7.8 pounds of 304 or 308 stainless and 0.4 pound to 1.4 pounds of Inconel springs. Alternate springs included 0.02 pound of carbon steel, 0.8 pound of 302 stainless steel and 0.5 pound to 1.1 pound of Inconel X-750. Overall length ranged from 116.2 inches to 158.8 inches.

Primary source rods for 17 x 17 fuel are clad with 304 stainless tubing and 308 stainless end plugs. The tubing weighs 0.92 pound to 1.07 pounds and the end plugs weigh 0.07 pound. The diameter is 0.385 inch, 0.019-inch wall thickness, 153.7 inches long. The source is californium with a strength of 2×10^8 to 1.2×10^9 Ci. The source rests on 32.5 inches of aluminum oxide spacer and the plenum length ranges from 0.6 inch to 2.65 inches. The total weight of a rod is about three pounds. For the other fuels, the cladding may be PDS 10708BN or 304 stainless steel with PDS 10708BN or 308 stainless steel end caps. The tubing weighs 0.83 pound to 1.1 pounds and the end plugs weigh 0.07 to 0.12 pound. The diameter is 0.371 to 0.466 inch and the wall thickness is 0.016 to 0.024 inch. The length is 110.4 to 152.5 inches. The sources are plutonium-beryllium, polonium-beryllium or californium. Plutonium-beryllium source strengths were 50 Ci, polonium-beryllium 200 Ci, and californium 4×10^8 Ci. The plutonium-beryllium sources rest on 8.3 inches of antimony-beryllium secondary source and has 96.3 inches of antimony-beryllium above it. The polonium-beryllium sources rest on 11.8 inches of Al_2O_3 or 14.8 inches of antimony-beryllium with 100.9 inches of Al_2O_3 or 101.0 inches of antimony-beryllium above it. The californium sources rest on 32.5 to 32.9 inches of Al_2O_3 or 304 stainless steel spacer and some have a spacer above them. The plenum length is 1.25 to 2.65 inches. The total weight of a rod ranges from 2.7 to 3.6 pounds.

Secondary source rods for 17 x 17 fuel are also clad in 304 stainless with 308 stainless end plugs. The tubing weighs 0.92 pound to 1.07 pounds, the end plugs weigh 0.07 pound, and the length is 152.3 inches. The source material is an alloy of 22% beryllium and 77% antimony. It is said to be 88 inches long with no spacer material and a plenum of 1.06 inches. The overall weight of the rod is 2.05 pounds. The 23 configurations that contain both primary and secondary sources might be left in the reactor more cycles than those containing only primary source rods. For the other fuels, the cladding may be PDS 10708BN or 304 stainless steel with PDS 10708BN or 308 stainless steel end caps. The tubing weighs 0.83 pound to 1.06 pounds and the end caps weigh 0.18 pound where data are given. The diameter ranges from 0.431 to 0.474 inch. The source material is antimony-beryllium and ranges in length from 67.1 to 121.65 inches. Again the data indicates that there is no spacer material, but the plenum space indicated that the end plugs must be 3 to 9 inches long. Rod weights range from 2.4 to 3.1 pounds.

Burnable poison rods for 17 x 17 fuel are clad with 0.92 pounds of 304 stainless with 0.08 pound of 308 stainless end plugs. The length is 152.4 inches, and the diameter is 0.385 inch. There is also 304 stainless radial spacer material 142.28 inches long weighing 0.11 pound. The absorber material is borosilicate glass tubing with 12.5% natural B_2O_3. The tubing is 142 inches long, 0.073 inch thick and weighs 0.70 pound. For the other fuel, 304 stainless steel cladding weighs from 1.10 to 1.18 pounds, and the 308 stainless end plugs weigh from 0.08 to 0.17 pound. The diameter ranges from 0.437 to 0.445 inch and the length from 150.4 to 152.8 inches. The 304 stainless steel spacer material is 142.95 inches long and weighs 0.19 pound. The absorber material is borosilicate glass tubing with 12.5% natural B_2O_3. The tubing is 0.072 to 0.076 inch thick, 141.6 to 142.7 inches long, and weighs 0.83 pound.

The thimble plugs range in length from 5.15 to 9.16 inches, from 0.424 to 0.502 inch in diameter, and 0.16 to 0.49 pound in weight.

Secondary Source Assemblies

Westinghouse manufactured 120 secondary source assemblies of 95 different configurations. One had three secondary source rods, 12 burnable poison rods and left 5 guide tubes vacant. One configuration had 4 secondary source rods and 16 burnable poison rods, leaving 4 guide tubes vacant. One configuration had 4 secondary source rods; 12 burnable poison rods, and three thimble plugs again leaving 5 guide tubes vacant. Two configurations had 4 secondary source rods; 16 guide tubes were left vacant. Fifty-five configurations had four secondary source rods and all the remaining locations filled with thimble plugs. Seven configurations consisted of four secondary source rods with the remaining locations filled with burnable poison rods. The remaining 28 configurations had 4 secondary source rods and 7 to 16 burnable poison rods; the remaining 4 to 12 locations were thimble plugs. The 16 to 24 rods and plugs are attached to a spider pack or hold-down assembly which is held down by the same hold-down plate which keeps the fuel assemblies in place. The hold-down assemblies and spider packs consist of 3.7 to 7.8 pounds of 304 or 308 stainless and 0.4 pound to 1.4 pounds of Inconel 718 springs. Alternate springs included 0.02 pound of carbon steel, 0.8 pound of 302 stainless steel and 0.5 pound to 1.1 pounds of Inconel X-750.

Secondary source rods for 17 x 17 fuel are clad in 304 stainless with 308 stainless end plugs. The tubing weighs 0.92 pound to 1.07 pounds, the end plugs weigh 0.07 pound, and the length is 152.3 inches. The source material is an alloy of 22% beryllium and 77% antimony. It is said to be 88 inches long with no spacer material and a plenum of 1.06 inches. The overall weight of the rod is 2.05 pounds. For the other fuels, the cladding may be PDS 10708BN or 304 stainless steel with PDS 10708BN or 308 stainless steel end caps. The tubing weighs 0.83 pound to 1.06 pounds, the end caps weigh 0.18 pound where data are given, and the diameter ranges from 0.431 to 0.474 inch. The source material, which is antimony-beryllium, ranges in length from 67.1 to 121.65 inches. Again the data indicate that there is no spacer

material, but the plenum space indicated that the end plugs must be 3 to 9 inches long. Rod weights range from 2.4 to 3.1 pounds.

Burnable poison rods for 17 x 17 fuel are clad with 0.92 pound of 304 stainless with 0.08 pound of 308 stainless end plugs. The length is 152.4 inches and the diameter is 0.385 inch. The 304 stainless steel radial spacer material is 142.28 inches long and weighs 0.11 pound. The absorber material is borosilicate glass tubing with 12.5% natural B_2O_3. The tubing is 142 inches long, 0.073 inch thick and weighs 0.70 pound. For the other fuel, the 304 stainless steel cladding weighs from 1.10 to 1.18 pounds and the 308 stainless end plugs weigh from 0.08 to 0.17 pound. The diameter ranges from 0.437 to 0.445 inch and the length from 150.4 to 152.8 inches. The 304 stainless steel spacer material is 142.95 inches long and weighs 0.19 pound. The tubing is 0.072 to 0.076 inch thick, 141.6 to 142.7 inches long, and weighs 0.83 pound.

The thimble plugs range in length from 5.15 to 9.16 inches, from 0.424 to 0.502 inch in diameter and 0.16 to 0.49 pound in weight.

The secondary source assemblies range in weight from 15 to 52 pounds and 91.1 to 158.8 inches in overall length.

2.8.5.3 Neutron Poisons

Burnable Poison Assemblies

Westinghouse has fabricated 5003 burnable poison assemblies in 329 different configurations. They had from 1 to 24 burnable poison rods with the remaining locations occupied by thimble plugs. The 16 to 24 rods and plugs are attached to a spider pack or hold-down assembly which is held down by the same hold-down plate that keeps the fuel assemblies in place. The hold-down assemblies and spider packs consist of 3.7 to 7.8 pounds of 304 or 308 stainless steel and 0.4 pound to 1.4 pounds of Inconel 718 springs. Alternate springs included 0.02 pound of carbon steel, 0.8 pound of 302 stainless steel and 0.5 pound to 1.1 pounds of Inconel X-750.

Burnable poison rods for 17 x 17 fuel are clad with 0.92 pound of 304 stainless steel with 0.08 pound of 308 stainless steel end plugs. The length is 152.4 inches and the diameter is 0.385 inch. The 304 stainless steel radial spacer material is 142.28 inches long and weighs 0.11 pound. The absorber material is borosilicate glass tubing with 12.5% natural B_2O_3. The tubing is 142 inches long, 0.073 inch thick and weighs 0.70 pound. For the other fuel, 304 stainless steel cladding weighs from 1.10 to 1.18 pounds and the 308 stainless end plugs weigh from 0.08 to 0.17 pound. The diameter ranges from 0.437 to 0.445 inch and the length from 150.4 to 152.8 inches. The 304 stainless steel spacer material is 142.95 inches long and weighs 0.19 pound. The absorber material is borosilicate glass tubing with 12.5% natural B_2O_3. The tubing is 0.072 to 0.076 inch thick, 141.6 to 142.7 inches long, and weighs 0.83 pound.

The thimble plugs range from 5.15 to 9.16 inches in length, from 0.424 to 0.502 inch in diameter, and from 0.16 to 0.49 pound in weight.

The overall weight of the assemblies ranges from 11 to 54 pounds and the overall length ranges from 151.6 to 156.9 inches.

Wet Annular Burnable Absorber Assemblies

Westinghouse described 143 configurations using 3 to 24 wet annular burnable absorber rods. All these configurations except one used thimble plugs in the remaining locations. The hold-down assemblies use 3.75 to 5.10 pounds of 304 stainless steel and 0.62 to 0.92 pound of Inconel 718 springs. The thimble plugs are 5.15 to 8.08 inches long and 0.424 to 0.498 inch in diameter. Their weight ranges from 0.16 to 0.30 pound. The absorber rods are 143.1 to 150.1 inches long and contain 105.5 to 134 inches of B4C-Al203 as annular pellets. The pellet wall thickness is 0.070 inch and is clad inside and outside with Zircaloy-4. Rods with shorter absorber lengths have up to 13.5 inches of Zircaloy-4 spacer. The inner cladding has a wall thickness of 0.020 inch, whereas the wall thickness of the outer tube is 0.026 inch. The cladding diameter is 0.381 inch. The top connector is made of 304 stainless steel. Each rod weighs 1.9 pounds and the assembly weighs from 15 to 52 pounds. If any of these assemblies have been built, they must be included in the 5003 burnable poison assemblies described above.

2.8.5.4 Thimble Plug Assemblies

Westinghouse has fabricated 8208 thimble plug assemblies from the thimble plugs and hold-down assemblies described above. The overall weight is expected to range from 9 to 19 pounds. These assemblies may stay in the reactor for more than one cycle. The 16 to 24 plugs are attached to a spider pack or hold-down assembly which is held down by the same hold-down plate which keeps the fuel assemblies in place. The hold-down assemblies and the spider packs consist of 3.7 to 7.8 pounds of 304 or 308 stainless and 0.4 pound to 1.4 pounds of Inconel 718 springs. Alternate springs included 0.02 pound of carbon steel, 0.8 pound of 302 stainless steel and 0.5 pound to 1.1 pounds of Inconel X-750.

The thimble plugs range from 5.15 to 9.16 inches in length, from 0.424 to 0.502 inch in diameter, and from 0.16 to 0.49 pound in weight.

2.8.5.5 In-Core Instrumentation

Westinghouse provides in-core instrumentation, but none was described in WSTD-TME-148.

2.8.6 Radiological Characterization

Because most nonfuel assembly hardware components remain in use for more than one assembly lifetime, they are expected to be more highly activated than corresponding pieces of spent fuel disassembly hardware, for example. Notable exceptions may be BWR fuel channels (if not reused), PWR burnable poison assemblies that are used for only one cycle, and BWR poison curtains. These components, depending on the concentration of niobium in the materials of construction, may qualify as Class C low-level waste. Other components, such as instrumentation, may also qualify as Class C waste because the majority of the component is far removed from the core of the reactor. The LWR NFA Hardware Data Base provides radiological characterization of nonfuel hardware components. A sample radiological description report is shown in Table 2.8.2. Preliminary estimates of the amounts of Greater than Class C Waste from both NFA and SFD hardware are given in section 5.1.1.

2.8.7 References for Section 2.8

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Cooper 1986. Letter from R. G. Cooper, Babcock & Wilcox, to A. R. Irvine, Oak Ridge National Laboratory, dated August 28, 1986.

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Hayduk 1987. D. M. Hayduk, Reference Manual of Core Components Fabricated by Combustion Engineering, Combustion Engineering Document CEND-428, March 1987.

Illinois Power. Illinois Power Company, Clinton Power Station Units I & 2, Preliminary Safety Analysis Report, DOCKET 50-461-3.

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Public Service. Public Service Electric and Gas Company, Newbold Island Nuclear Generating Station, Preliminary Safety Analysis Report, DOCKET 50-354-2.

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Table 2.8.1. Sample Physical Description Report from LWR NFA Hardware Data Base.

Physical Description Report Page: 1

Combustion Enigneering SYSTEM80 12-Rod Full-Length Control Element

Designed for:
Fuel Assembly with array size: 16 x 16
Pressurized Water Reactor

Dimensions:

Total Length:	253	inches
Total Weight:	192.2	pounds

Cladding:

Material:	Inconel 625	
Outer Diameter:	0.816	inches
Wall Thickness:	0.035	inches
Diametral Gap:	0.009	inches

Poison:

Primary Material:	Boron Carbide (CE)	
Poison Length:	148	inches
Pellet Diameter:	0.737	inches

Plenum Spring Material: St. Steel 302

Spider Material: St. Steel 304

Number of Control Rods: 12

Life Expectancy: 4000 EFPD

Table 2.8.1. Sample Physical Description Report from LWR NFA Hardware Data Base. (cont.)

Physical Description Report Page: 2

Combustion Enigneering SYSTEM80 12-Rod Full-Length Control Element

Composition:

Material	Total Weight(kg)	Neutron Zone
St.Steel 304	8.17	Top
Inconel 625	53.62	Top
Boron Carbide (CE)	20.90	Top
St.Steel 304	0.68	Gas Plenum
Inconel 625	2.20	Gas Plenum
Boron Carbide (CE)	1.60	Gas Plenum

Used at the Following Reactors:

Reactor	Number in Core
Palo Verde 1	48
Palo Verde 2	48
Palo Verde 3	48

Used with the Following Fuel Assembly Types:

Vendor	Array	Version
Combustion Engineering	16 x 16	System 80

Table 2.8.2. Sample Radiological Description Report from the LWR NFA Hardware Data Base.

Radiological Description Report Page 1

Combustion Engineering SYSTEM80 12-Rod Full-Length Control Element

ISOTOPIC COMPOSITION

Used for 7 cycles (77,000 MWd/MTIHM) 5 years after discharge
Weight: 97.170 kg Volume of metal: 0.013289 Cu. Meters

Isotope	Grams	Watts	Curies	Curies/m3	Class C Limit	Class C Ratio
C-14	5.348E-04	6.994E-07	2.384E-05	5.311E-03	80	0.6
Ni-59	2.474E-01	7.447E-07	1.876E-02	4.179E+02	220	1.9
Ni-63	3.583E-02	2.227E-04	2.211E+02	4.926E+04	7000	7.0
Co-60	9.512E-03	1.659E-03	1.068E+01	2.397E+03	N/A	N/A
Nb-94	9.760E-03	1.865E-05	1.831E+00	4.097E+02	0.2	220
Total	5.490E+00	1.535E+00	8.349E+03	2.465E+06	N/A	N/A

Used for 10 cycles (111,000 MWd/MTIHM) 5 years after discharge
Weight: 97.170 kg Volume of metal: 0.013289 Cu. Meters

Isotope	Grams	Watts	Curies	Curies/m3	Class C Limit	Class C Ratio
C-14	5.348E-04	6.994E-07	2.384E-05	5.311E-03	80	0.6
Ni-59	2.474E-01	7.447E-07	1.876E-02	4.179E+02	220	1.9
Ni-63	3.583E-02	2.227E-04	2.211E+02	4.926E+04	7000	7.0
Co-60	9.512E-03	1.659E-03	1.068E+01	2.397E+03	N/A	N/A
Nb-94	9.760E-03	1.865E-05	1.831E+00	4.097E+02	0.2	220
Total	5.490E+00	1.535E+00	8.349E+03	2.465E+06	N/A	N/A

NOTE: The data presented here is only for the purpose of illustrating the form of the Radiological Description Report. It is not intended to be used for any purpose other than that illustration.

Table 2.8.2. Sample Radiological Description Report from the LWR NFA Hardware Data Base (cont.).

Radiological Description Report Page 2

Combustion Engineering SYSTEM80 12-Rod Full-Length Control Element

PHOTON SPECTRA

Mean Energy(MeV)	Photons/second (77,000 MWd/MTIHM)	Photons/second (110,000 MWd/MTIHM)
0.0100	2.162E+10	3.569E+10
0.0250	3.674E+09	6.063E+09
0.0375	2.088E+09	3.444E+09
0.0575	2.397E+09	3.874E+09
0.0850	9.237E+08	1.524E+09
0.1250	3.548E+08	7.851E+08
0.2250	1.167E+08	1.925E+08
0.3750	3.272E+07	5.396E+07
0.5750	1.879E+06	3.099E+06
0.8500	6.411E+08	9.650E+08
1.2500	7.960E+11	1.313E+12
1.7500	2.253E+01	2.768E+01
2.2500	4.219E+06	6.956E+06
2.7500	1.306E+04	2.152E+04

METALLIC COMPOSITION
(Materials modeled to obtain this report)

Material	Total Weight (kg)	Zone
Inconel 625	53.620	Top
Boron Carbide	20.900	Top
Stainless Steel 304	8.170	Top
Inconel 625	2.200	Gas Plenum
Boron Carbide	1.600	Gas Plenum
Stainless Steel 304	0.680	Gas Plenum

NOTE: The data presented here is only for the purpose of illustrating the form of the Radiological Description Report. It is not intended to be used for any purpose other than that illustration.

3. IMMOBILIZED HIGH-LEVEL WASTE

3.1 SUMMARY

Canisters of high-level waste (HLW) immobilized in borosilicate glass or glass-ceramic mixtures are to be produced at West Valley Demonstration Project (WVDP), Savannah River Plant (SRP), Hanford (HANF), and Idaho National Engineering Laboratory (INEL) for shipment to one or more geologic repositories. Data are presented in this section on the estimated physical characteristics and production schedules of the canisters of immobilized waste through the year 2020.

3.1.1 Canister Dimensions and Weights

Table 3.1.1 summarizes the estimated dimensions and weights of the canisters for the four sites. Three of the sites (WVDP, SRP, and HANF) plan to use cylindrical stainless steel canisters, 61 cm (24 in.) in diameter and 300 cm (118 in.) high, filled with borosilicate glass to about 85% of the canister volume. The 85% figure refers to the glass volume at filling temperature, which is about 825°C (average) in the canister as filled. According to SRP experiments, cooling the canister to ambient temperature does not reduce the glass level in the canister appreciably. The designs are similar but not identical; wall thickness and filler neck dimensions vary. The weight of a loaded canister is about 2150 to 2180 kg, of which about 1650 to 1895 kg is HLW glass.

For INEL, neither the canister dimensions nor the waste form have been fixed. Borosilicate glass and hot-isostatic-pressed (HIP) glass-ceramic forms are being considered; the glass-ceramic form requires a considerably smaller number of canisters than the glass form for a given amount of waste. The estimates in Table 3.1.1 for INEL are based on information from INEL (Berreth 1987), which, in turn, is based on the assumptions that the glass-ceramic form will be used for immobilization and that the external dimensions of the canister will be the same as those used for WVDP, SRP, and HANF.

Table 3.1.1 also summarizes the estimated maximum radioactivity and thermal power (curies and watts) per canister at the time of filling,

based on the most highly radioactive immobilized waste composition at each site. Curies and watts as functions of decay time after filling are given in subsequent sections on the individual sites.

3.1.2 Canister Production Schedules

The total number of HLW canisters to be produced at each site by the year 2020 is not yet completely established. The estimate for the WVDP site is about 275 to 300 canisters; this estimate should be fairly accurate, because the amount of waste at WVDP is a known quantity. For the three defense sites, there are several possible scenarios and options that can lead to different total numbers of canisters. This report will present one such scenario, which will be referred to as the base case. Other possible cases that give larger numbers of canisters are also discussed, and the assumptions and processing options that go into each case are described.

Table 3.1.2 shows the estimated production schedule of canisters at each site in the base case. Both the annual number and the cumulative number of canisters are shown for each year through the year 2020. In this scenario, based on the assumptions used in this report, it is estimated that a total of about 18,000 canisters will have been produced by the end of 2020. Table 3.1.3 summarizes the assumptions on which the estimates in Table 3.1.2 are based. All projections were obtained from the respective sites. Startup dates are based on current plans, which call for initial vitrification to begin at WVDP and SRP in 1990 and at HANF in 1996. For INEL, the actual strategy and process for disposal of HLW will not be decided until the 1990s; however, for planning purposes it is assumed that the glass-ceramic waste form will be used for immobilization and that immobilization will start in 2011. The immobilized waste generation schedule for INEL shown in Table 3.1.2 assumes that during the first three years of operation the immobilization plant will operate at a reduced rate (500 to 700 canisters per year), which is consistent with the annual fuel reprocessing rate. After the third year, a production rate of 1000 canisters/year is assumed to allow for working off the backlog of stored calcine over a period of about 30 years (Berreth 1987).

The production schedules detailed in Table 3.1.2 are shown graphically in Figs. 3.1.1 and 3.1.2. Figure 3.1.1 shows the cumulative numbers of canisters produced at each of the four individual sites through the year 2020, and Figure 3.1.2 shows the cumulative number of canisters summed for all sites.

3.1.2.1 Comparison with Previous Projections

A recent report on defense HLW repository fee calculation (DOE/RL-86-10, 1986) gave estimated defense HLW canister production for three cases referred to (in order of increasing number of canisters) as the base case, augmented case, and maximum case. These cases were described as follows: The base case gave the quantity of defense HLW from SRP, HANF, and INEL that was expected to go to the geologic repositories, assuming that INEL waste was in the low-volume glass-ceramic form with removal of inerts prior to immobilization. The augmented case was the same as the base case except that the INEL waste was assumed to be converted to glass-ceramic form without removal of inerts prior to immobilization. The maximum case was the augmented case plus vitrified HANF single-shell tank waste and overpacked strontium and cesium capsules; in this case, each overpack was counted as a canister. The number of canisters produced in each of these cases is shown in Table 3.1.4, and the year-by-year production schedule of canisters for the base case is shown in Table 3.1.5. Table 3.1.6 summarizes the assumptions that were used to produce the data in Tables 3.1.4 and 3.1.5.

In DOE/RL-86-10, the total number of canisters produced by a given year was stated in terms of "equivalent" canisters. This means that all of the HLW produced by that year was included in the calculation of the number of canisters, although this total quantity of waste would not actually be canistered until several years later. Thus the cumulative equivalent numbers of canisters shown in Table 3.1.4 for a given year are not readily comparable to those calculated in the present report (Table 3.1.2), which are based on cumulative actual canisters produced by a given year. However, Table 3.1.5, which shows the DOE/RL-86-10 base-case canister production schedule, can be directly compared with

the base case schedule in our Table 3.1.2, since both are in terms of actual canisters. The total number of defense HLW canisters produced through the year 2020 is about 17,500 in this report and about 14,000 in the base case of DOE/RL-86-10. Most of the difference is in the production at INEL: 8,800 canisters in this report compared with 4,350 in DOE/RL-86-10. This in turn was due to the assumption of inerts removal prior to immobilization in DOE/RL-86-10, which gave an immobilization rate of 335 canisters/yr vs 1,000 canisters/yr in this report. INEL considers that the assumption of inerts removal prior to immobilization is based on unproven technology and gives no cost advantage, and therefore INEL did not use the assumption of inerts removal in making their own estimates of the number of canisters produced (Berreth and Knecht 1986, Berreth 1987a). Thus the augmented case in DOE/RL-86-10 (no inerts removal prior to immobilization) is more comparable to the base case in the present report; the augmented case would have an INEL production rate of about 1,000 canisters/yr for a total production at INEL of 13,000 canisters by 2020, and an overall total DHLW production of about 22,600 canisters by 2020. This is about 5,000 canisters more than the 17,500 estimated in this report. Most of this difference is accounted for by the recently revised INEL startup schedule (Berreth 1987), which shows a total production of 8,800 canisters by the end of 2020 based on starting up in year 2011, rather than the 13,000 canisters that would be produced in the DOE/RL-86-10 augmented case based on startup at full rate in year 2008. Taking this adjustment into account, the remaining difference in total canister production in the two reports is about 900 canisters by year 2020. This is accounted for by differences in SRP and HANF production in the two reports. For SRP, the present report shows 6,800 canisters versus 7,500 in DOE/RL-86-10, and for HANF the corresponding estimates were 1900 canisters in this report and 2100 in DOE/RL-86-10. The DOE/RL-86-10 projections include "future" production of 1100 canisters at SRP and HANF, split about equally between the two sites. In the present report there is no category of "future" production; updated projections from the defense sites through the year 2020 were used directly. As mentioned previously, the totals presented here do not include any canisters produced after the year 2020.

In DOE/RL-86-10, it was estimated that vitrification of HANF single-shell tank waste for repository emplacement, if required, would result in an additional 21,500 canisters. Based on the current design throughput of the HANF vitrification plant (145 canisters/yr), production of this number of canisters would take many years. Referring to the base case production schedule shown in Table 3.1.2, the maximum additional number of canisters that could be produced at HANF from year 2011 to year 2020 would be about 1500 canisters. Thus the maximum case for this report (i.e., including vitrification of single-shell tank waste) would be the base case plus 1500 canisters, or a total of 19,300 canisters by the end of year 2020. This assumes that the present HANF vitrification plant design throughput is not increased; however, it appears likely that this throughput would be revised upward if vitrification of single-shell tank waste became necessary. Also, it appears likely that the requirement of 21,500 canisters for this waste could be considerably reduced by pretreatment.

3.1.3 Radiological Properties

To the extent possible, data were obtained from the sites on the projected radionuclide content of the canisters at the time of filling. These data represent estimated maximum radioactivities per canister for the three defense sites and both maximum and average radioactivity per canister for WVDP. Where current information from the sites was unavailable, estimates were made based on previously published data. Using this information, ORIGEN2 decay calculations were made to determine the grams, curies, and thermal power (watts) of each nuclide, and the total per canister, for decay times up to 10^6 years after filling. This information is summarized in this chapter for each of the sites and is presented in more detailed form in Appendix 3A. These data are also available in magnetic diskette form; the diskette data also show alpha curies and photon energy distributions.

Cumulative average radioactivities per canister have also been estimated for the defense sites based on their projections of cumulative radioactivity in glass and cumulative number of canisters produced.

These cumulative average radioactivities per canister are in general much lower than the maximum radioactivities per canister previously discussed and should be more useful than the maximum values for the calculation of total repository thermal loading at time points subsequent to the year 2020. However, the year-by-year cumulative average radioactivities and thermal power per canister have been calculated from preliminary data that cannot take into account actual tankage allocations and detailed processing schedules; thus these averages should not be used for detailed short-term calculations.

3.1.4 Assessment of Data

At WVDP, the radionuclide content per canister is fairly well established. Reprocessing of fuel was discontinued in 1972; thus the waste to be immobilized is a fixed quantity and its composition is known. The radiological properties per canister can therefore be readily calculated, and the only variation of these properties with time is that due to the process of radioactive decay. These calculations require only a single ORIGEN2 decay run, which starts with a single fixed composition and tracks the resulting grams, curies, and watts for any desired series of decay times.

At the three defense sites, however, the situation is more complex. Plants at these sites continue to process fuel, so the wastes in storage are a mixture of old, well-aged waste and newly processed waste of much higher radioactivity. When vitrification begins, it might be desirable to try to work off the older waste first; however, this may not be feasible because of tankage constraints. Thus, the proportions of old and new waste fed to the melter will vary from year to year. In addition, the composition of the freshly produced waste may undergo changes. However, even if this latter variation does not occur, the radiological properties of a canister from a defense site will depend on the melter feed composition in the year in which the canister was filled as well as on the decay time elapsed since filling. A complete characterization of

the canisters produced from such a site would require a schedule of melter feed composition versus time and a separate decay calculation for each melter feed composition and decay time.

Little data are available on estimated melter feed compositions and on the estimated variation of these compositions following startup of the vitrification plants. Two defense sites (SRP and HANF) have each released compositions representing feeds of estimated maximum activity. HANF has also issued four compositions representing the estimated variation of radionuclide contents between 1996 and 2000 (Mitchell 1986). INEL, because of security restrictions, has released no data on radionuclide compositions. To provide preliminary estimates of maximum radioactivity and decay heat per canister as a function of decay time for INEL HLW, a composition based on 1982 data was used. Assessment of the data presented in this report pinpoints the variation of radionuclide compositions of melter feeds with time as an area requiring additional information and analysis.

Equally important in the assessment of the data in this report, it should be recognized that various strategies and processing alternatives for immobilized waste production are still under consideration. Also, future defense production requirements may change. This report presents a scenario that appears likely at this time; however, changes in the canister production schedule, the radiological properties of canisters, and the total number of canisters produced are still possible. It must also be kept in mind that this report does not present any information on the number of defense HLW canisters produced after the year 2020. Based on the quantities of HLW remaining uncanistered at the end of 2020, it is clear that several thousand additional canisters will be required after that time.

3.1.5 Interim Forms of High-Level Waste

At present, the high-level wastes stored at the sites are in various interim forms such as liquids, slurries, sludges, calcine, etc.

The quantities and compositions of these interim forms and their conversion to final forms are discussed in Appendix 3B, which thus serves to provide the detailed backup data for the information presented in this section, as well as additional details on the processing of the waste.

3.1.6 Order of Presentation

The remainder of this section is arranged according to site location and is presented in the following order: WVDP, SRP, HANF, and INEL. For each site, the data are presented in a fixed order, as follows: (1) types of waste produced, (2) canister dimensions and weights, (3) canister production schedule, (4) radionuclide content per canister at time of filling, (5) radiological properties (curies and watts) per canister as a function of time after filling, (6) chemical composition of waste form, and (7) assessment of data.

3.1.7 References for Section 3.1

Berreth 1987. Letters from J. Berreth, INEL, to J. E. Solecki, DOE/IDO, March 19, 1987 and April 1, 1987.

Chandler 1987. Letter from R. L. Chandler to M. W. Shupe, HLW Lead Office, Richland, transmitting SRP input to DHLW Integrated Data Base, April 1, 1987.

Coony 1987. F. M. Coony, Rockwell Hanford, submission of Hanford HLW data to IDB, March 1987.

DOE/RL-86-10, 1986. Defense High-Level Waste Technology Program Office, Perspective on Methods to Calculate a Fee for Disposal of Defense High-Level Waste in Combined (Civilian/Defense) Repositories, December 1986.

McDonell and Goodlett 1984. W. R. McDonell and C. B. Goodlett, Systems Costs for Disposal of Savannah River High-Level Waste Sludge and Salt, DP-MS-83-121, August 1984.

Mitchell 1986. D. E. Mitchell, Hanford Waste Vitrification Plant, Preliminary Description of Waste Form and Canister, RHO-RE-SR-55P, August 1986.

Rykken 1987. Telephone conversation, L. E. Rykken, WVDP, and R. Salmon, ORNL, March 25, 1987.

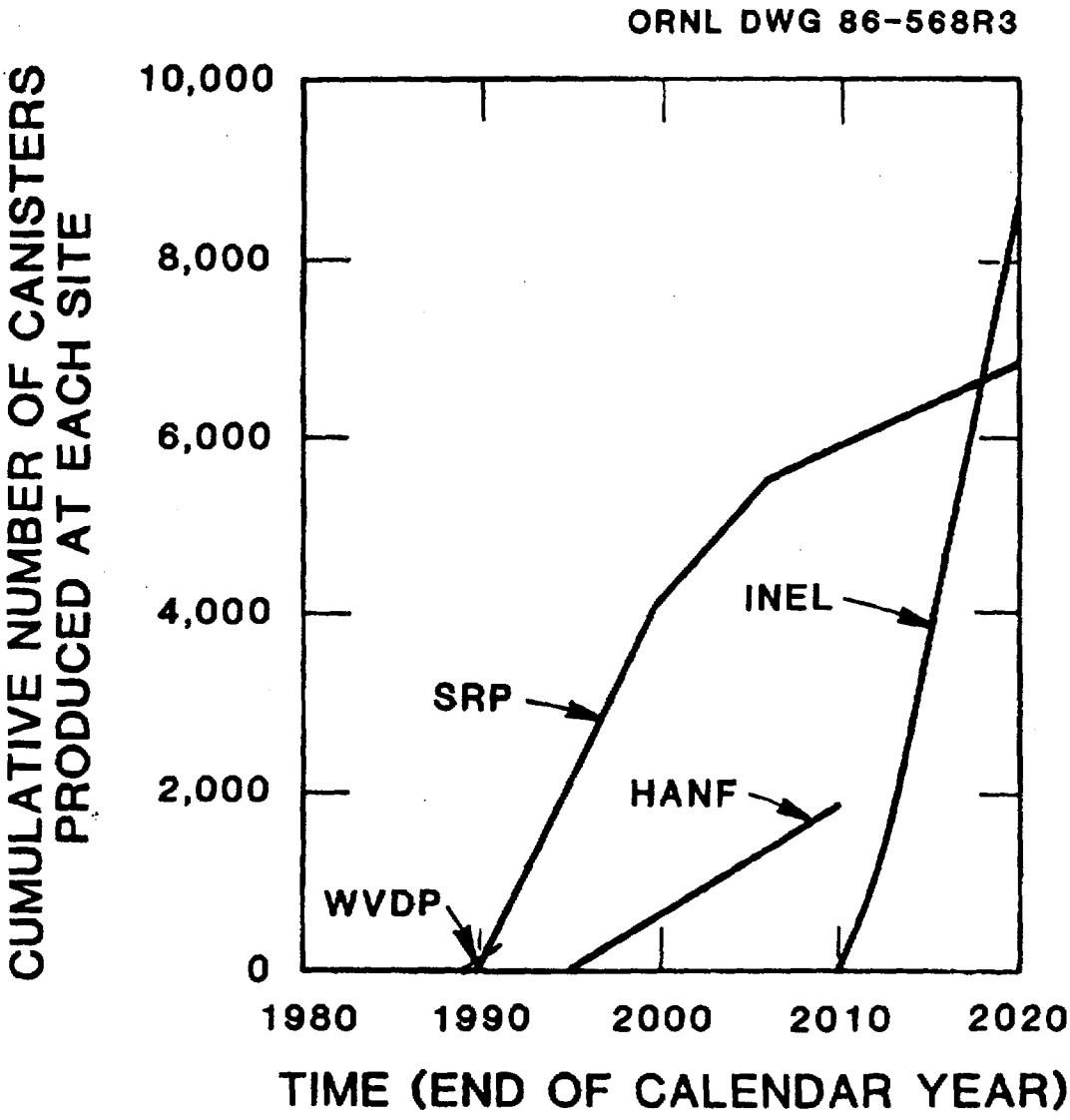

Fig. 3.1.1. Cumulative number of canisters of HLW produced at each individual site. Base case, as shown in Table 3.1.2.

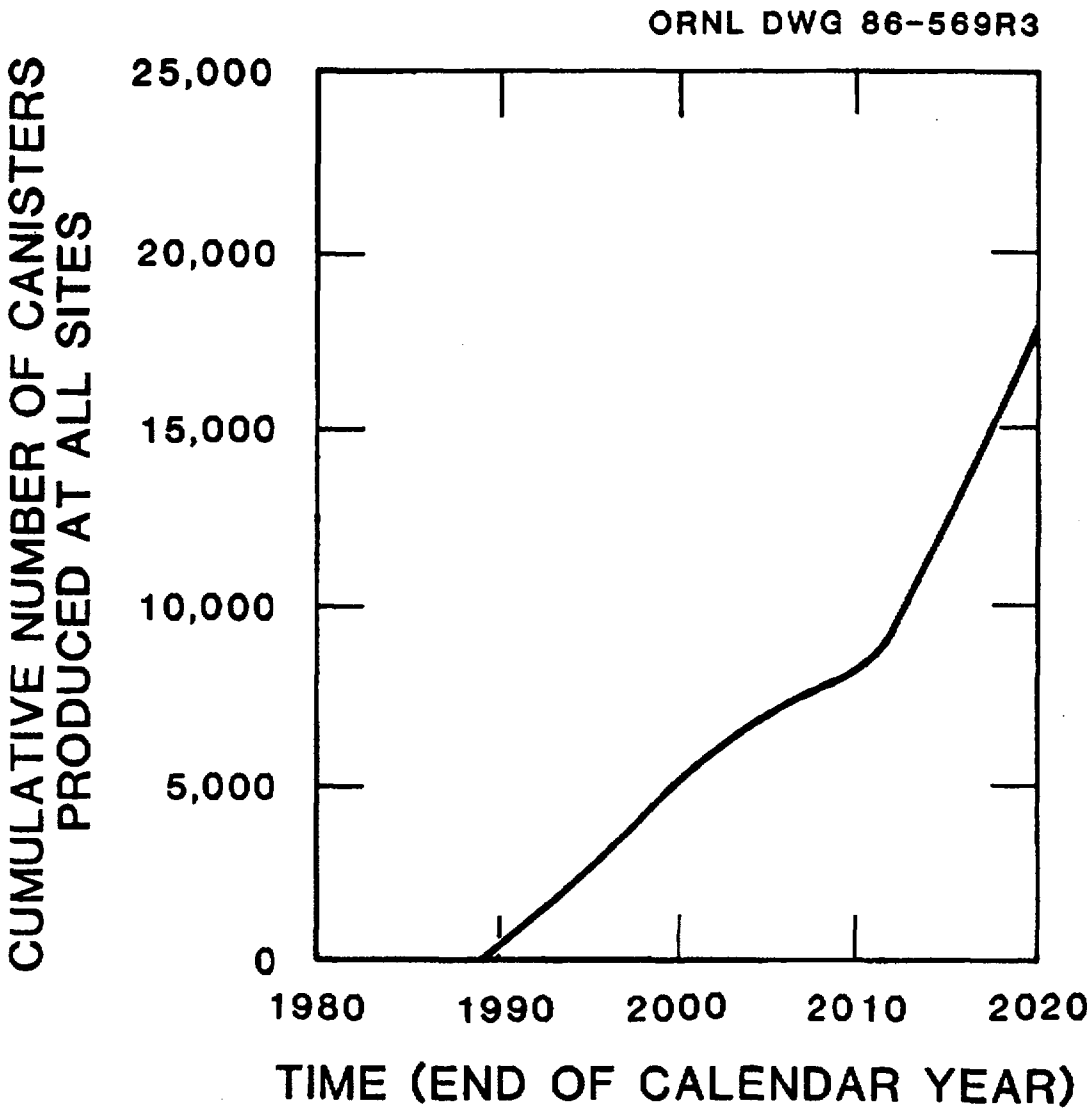

Fig. 3.1.2. Cumulative number of canisters of HLW produced, total for all four sites. Base case, as shown in Table 3.1.2.

Table 3.1.1. Dimensions, weights, and radioactivity of canisters (summary)

	West Valley Demonstration Project	Savannah River Plant	Hanford	Idaho National Engineering Laboratory
Outside diameter, cm	61	61	61	61
Overall height, cm	300	300	300	300
Material	SS	SS	SS	SS
Wall thickness, cm	0.34	0.95	0.95	0.95
Weights (kg)				
Canister	252	500	500	500
Glass or ceramic	1895	1682	1650	1825
Total	2147	2182	2150	2325
Curies per canister[a]	125,200	234,400	416,000	143,000
Watts per canister[a]	382	709	1158	446

[a]These are estimated maximum values from ORIGEN2 calculations based on radionuclide compositions supplied by the sites. Curies and watts shown are at time of filling the canister, except for West Valley Demonstration Project where the values shown are for the end of year 1990. Maximum values for the defense sites do not represent initial operations; canisters of maximum activity will not be produced until after several years of operation.

Table 3.1.2. Number of canisters of immobilized HLW produced at all sites[a,b]

End of calendar year	West Valley Demonstration Project		Savannah River Plant		Hanford Operations		Idaho National Engineering Laboratory		Total for all four sites	
	Number of canisters produced during year	Cumulative number of canisters produced	Number of canisters produced during year	Cumulative number of canisters produced	Number of canisters produced during year	Cumulative number of canisters produced	Number of canisters produced during year	Cumulative number of canisters produced	Number of canisters produced during year	Cumulative number of canisters produced
1988	0	0	0	0	0	0	0	0	0	0
1989	0	0	0	0	0	0	0	0	0	0
1990	130	130	102	102	0	0	0	0	232	232
1991	145	275	410	512	0	0	0	0	555	787
1992	0	275	410	922	0	0	0	0	410	1197
1993	0	275	410	1332	0	0	0	0	410	1607
1994	0	275	410	1742	0	0	0	0	410	2017
1995	0	275	410	2152	0	0	0	0	410	2427
1996	0	275	410	2562	145	145	0	0	555	2982
1997	0	275	410	2972	145	290	0	0	555	3537
1998	0	275	410	3382	145	435	0	0	555	4092
1999	0	275	410	3792	73	508	0	0	483	4575
2000	0	275	410	4202	145	653	0	0	555	5130
2001	0	275	220	4422	145	798	0	0	365	5495
2002	0	275	220	4642	72	870	0	0	292	5787
2003	0	275	220	4862	145	1015	0	0	365	6152
2004	0	275	220	5082	145	1160	0	0	365	6517
2005	0	275	220	5302	145	1305	0	0	365	6882
2006	0	275	220	5522	73	1378	0	0	293	7175
2007	0	275	92	5614	145	1523	0	0	237	7412
2008	0	275	92	5706	145	1668	0	0	237	7649
2009	0	275	92	5798	72	1740	0	0	164	7813
2010	0	275	92	5890	120	1860	0	0	212	8025
2011	0	275	92	5982	0[c]	1860	500	500	592	8617
2012	0	275	92	6074	0	1860	600	1100	692	9309
2013	0	275	92	6166	0	1860	700	1800	792	10101
2014	0	275	92	6258	0	1860	1000	2800	1092	11193
2015	0	275	92	6350	0	1860	1000	3800	1092	12285
2016	0	275	92	6442	0	1860	1000	4800	1092	13377
2017	0	275	92	6534	0	1860	1000	5800	1092	14469
2018	0	275	92	6626	0	1860	1000	6800	1092	15561
2019	0	275	92	6718	0	1860	1000	7800	1092	16653
2020	0	275	92	6810	0	1860	1000	8800	1092	17745

[a]Sources: WVDP - Rykken 1987.
SRP - Chandler 1987 (IDB submittal).
HANF - Coony 1987.
INEL - Barrath 1987.

[b]For assumptions used in compiling this table see Table 3.1.3. This table represents the base case for this report. Canisters produced after 2020 are not included here.

[c]The Hanford schedule is based on the assumption that there will be no fuel reprocessing operations after calendar year 2001. Some planning scenarios do project such operations beyond CY 2001. Each additional year of fuel reprocessing would generate about 50 canisters after CY 2010 (Coony 1987).

Table 3.1.3. Assumptions used in base case of this report

1. Canister dimensions 61 cm diameter by 300 cm length; 85% fill assumed at filling temperature.

2. Maximum immobilization throughputs of the various sites, in canisters per year, are as follows: WVDP, 200; SRP, 410; HANF, 145; INEL, 1000.

3. Production of canisters of HLW starts at WVDP and SRP in 1990, at HANF in 1996, and at INEL in 2011. Canister production is shown through the end of year 2020 and does not include any waste canistered after 2020.

4. WVDP canister production is based on 520,000 kg of total glass loaded at 1890 kg/canister (Rykken 1987).

5. SRP canister production is based on SRP 1987 IDB submittal (Boore 1987). Waste production rates are based on operation of three reactors throughout the projection period (2020); last year's forecast was based on a four-reactor case. The DWPF is assumed to reach full production in 4QFY 1990. It is assumed that sludge and liquid inventories will reach steady state in 2006. Steady-state volumes must be maintained to allow waste to age before it is processed.

6. HANF canister production is based on HANF "reference alternative," in which single-shell tank waste is not vitrified but is immobilized in place. About 21,000 m^3 of single-shell tank liquid is transferred to double-shell tanks between 1985 and 1996. HANF strontium/cesium capsules are not reprocessed and vitrified but are overpacked for emplacement in a repository; overpacks are not included in canister production figures. It is assumed that the N Reactor operates through the year 2000 and the Purex plant operates through the year 2001. It is assumed in this projection that there will be no fuel reprocessing operations after year 2001. However, there are some planning scenarios that do project such operations beyond year 2001.

7. INEL canister production is based on the schedule given in Berreth 1987a. The ceramic-based waste form is used for immobilization. No removal of inerts prior to immobilization was assumed. A canister load is 1825 kg of ceramic, equivalent to 1277 kg of calcine. Density of ceramic is 3200 kg/m^3. The maximum production rate is 1000 canisters/year; this permits working off the stored calcine over about a 30-year period.

Table 3.1.4. Defense high-level waste quantities and characteristics used in report DOE/RL-86-10.[a]

Site/waste form	Cumulative number of equivalent canisters to: 1986	2000	2020
Base case			
Savannah River Plant - glass	4,900	7,000	7,000
Hanford - glass	800	1,500	1,500
Future production (2000-2020)[b] - glass	-	-	1,500
Idaho - ceramic (with inerts removal)[c]	900	2,500	6,000
Total of base case	6,600	11,000	16,000
Augmented			
Base case without Idaho	5,700	8,500	10,000
Idaho - ceramic (without inerts removal)	3,000	9,000	22,000
Total augmented case	8,700	17,500	32,000
Maximum			
Augmented quantity	8,700	17,500	32,000
Hanford additional - glass[d]	21,500	21,500	21,500
Hanford Cs and Sr capsules - overpacked	500	500	500
Total maximum case	30,700	39,500	54,000

[a]This table is excerpted from Table 4.1 of DOE/RL-86-10, December 1986; cumulative radioactivity and thermal power have been deleted. All values shown are totals for the year shown; that is, they are not additive in the horizontal direction. "Equivalent canisters" means the number of canisters that would result if all of the waste produced by a given year were immobilized.

[b]Assumed equivalent to two Savannah River reactors on sites to be determined.

[c]This case is based on a postulated process for removing inerts prior to immobilization.

[d]Quantity if all single-shell-tank (SST) waste must go to geologic disposal.

Table 3.1.5. Defense high-level waste canister production schedule used in base case of report DOE/RL-86-10[a]

Year	Savannah River		Hanford		Future production[b]		Idaho[c]		DHLW cumm. total
	Rate	Total	Rate	Total	Rate	Total	Rate	Total	
1990	405	405							405
1991	405	810							810
1992	405	1,215							1,215
1993	405	1,620							1,620
1994	405	2,025							2,025
1995	405	2,430							2,430
1996	405	2,835							2,835
1997	405	3,240	135	135					3,375
1998	405	3,645	135	270					3,915
1999	405	4,050	135	405					4,455
2000	405	4,455	135	540					4,995
2001	405	4,860	135	675					5,535
2002	405	5,265	135	810					6,075
2003	105	5,370	135	945					6,315
2004	105	5,475	135	1,080					6,555
2005	105	5,580	135	1,215					6,795
2006	105	5,685	135	1,350	75	75			7,110
2007	105	5,790	150	1,500	75	150			7,440
2008	105	5,895			75	225	335	335	7,955
2009	105	6,000			75	300	335	670	8,470
2010	105	6,105			75	375	335	1,005	8,985
2011	105	6,210			75	450	335	1,340	9,500
2012	105	6,315			75	525	335	1,675	10,015
2013	105	6,420			75	600	335	2,010	10,530
2014	105	6,525			75	675	335	2,345	11,045
2015	105	6,630			75	750	335	2,680	11,560
2016	105	6,735			75	825	335	3,015	12,075
2017	105	6,840			75	900	335	3,350	12,590
2018	105	6,945			75	975	335	3,685	13,105
2019	55	7,000			75	1,050	335	4,020	13,570
2020					75	1,125	335	4,355	13,980
Total canisters produced		7,000		1,500		1,125		4,355	13,980
Total canisters from waste generated through 2020		7,000		1,500		1,500		6,000	16,000

[a]This table is the same as Table 7.1 of DOE/RL-86-10, December 1986. This schedule is representative of the many possible production scenarios.

[b]A level rate for 20 years was assumed.

[c]Processing for Future and Idaho will continue after 2020 to allow for cooling of wastes produced from 2016 to 2020. The additional waste that will be produced at Idaho after 2020 is not considered here.

Table 3.1.6. Assumptions used in estimating number of defense HLW canisters in report DOE/RL-86-10, December 1986

1. Definitions of cases:

   Base case: The quantity of immobilized defense HLW expected to go to geologic repositories, with INEL waste in the low-volume ceramic form with inerts removed prior to immobilization.

   Augmented case: Same as base case except that INEL waste volume is increased because inerts are not removed.

   Maximum case: Same as augmented case except that HANF single-shell tank waste is vitrified and HANF strontium-cesium capsules are overpacked for shipment to repository.

2. (All cases) canister dimensions 61 cm diameter x 300 cm overall length.

3. (All cases) production of HWL canisters begins at SRP in 1990, at HANF in 1997, and at INEL in 2008.

4. (All cases) canister production in Table 3.1.4 is given as number of equivalent canisters; this means the number of canisters that would be required if all of the waste produced through a given year were immobilized.

5. (Base case) SRP production is based on McDonell and Goodlett, 1984.

6. (Base case) HANF production is based on the HANF "reference alternative," in which single-shell tank waste is not vitrified, but is immobilized in place.

7. (Base case) Future defense production (i.e. production from 2000 to 2020) is shown separately; but it is likely that this production will be split between SRP and HANF. Future production is assumed to be equivalent to two SRP reactors and is assumed to give a constant rate of canister production of 75 canisters/year for 20 years.

8. (Maximum case) HANF single-shell tank waste is vitrified, requiring an additional 21,500 canisters. The strontium and cesium capsules produced at HANF are placed in overpacks for disposal at the repository. The total number of overpacks is 500; each of these is counted as a canister.

3.2 WEST VALLEY DEMONSTRATION PROJECT

3.2.1 Introduction

Approximately 660 metric tons of irradiated fuel were processed at the commercial fuel reprocessing plant at West Valley, New York, from 1966 to 1972; the reprocessing plant was then shut down. The West Valley Demonstration Project (WVDP), jointly funded by the U.S. DOE and the New York State Energy Research and Development Agency, was started in 1982 with the objective of solidifying the HLW remaining from the commercial reprocessing operations into a form suitable for transportation and disposal in a federal repository.

3.2.2 Types of HLW Produced

Only one type of immobilized HLW will be produced at WVDP, i.e. HLW immobilized in borosilicate glass encased in stainless steel canisters.

3.2.3 Physical Description

Figure 3.2.1 and Table 3.2.1 show details of the HLW glass canister planned for use at the WVDP vitrification facility. The canister is approximately 0.61 m in diameter and 3.0 m in height and is welded shut after filling. The top closure is a cap made of flat plate about 0.95 cm thick. The expected fill volume is 85% of capacity $\pm$5%. The empty canister weighs about 234 kg. When filled to 85% of capacity, each canister will contain 0.70 m^3 (about 1895 kg) of vitrified waste and will weigh about 2147 kg. The density of the solidified waste glass is approximately 2.7 g/cm^3 at 25°C (Rykken 1986a,b,c, Eisenstatt 1986).

3.2.4 Inventory and Production Schedule

Cold operations at the vitrification plant are scheduled to start in 1989 and be completed in April 1990. Vitrification of waste is scheduled to begin in April 1990 and to be completed about October 1991. A total of about 520,000 kg of vitrified wsste (about 275 canisters) will be produced during this period (Rykken 1987). Table 3.2.2 shows the estimated schedule of production; this schedule is based on a single

campaign with a duration of 18 months, starting April 1990 and ending in October 1991. This allows about 20% offstream time for scheduled and unscheduled shutdowns, producing a total of 109 batches at 100 onstream hours per batch. On this basis, 130 canisters of waste will be produced in 1990 and 145 canisters in 1991. This will account for the entire quantity of HLW at WVDP.

3.2.5 Radionuclide Content per Canister

The initial radionuclide contents per canister of the glass were taken from Eisenstatt 1986, which gives average, maximum, and minimum values. These compositions, expressed as curies of each radionuclide per canister, are shown in Table 3.2.3. Data are for the year 1990.

3.2.6 Radioactivity and Thermal Power

Table 3.2.4 shows calculated radioactivity and thermal power per canister as a function of decay time for maximum and average canisters for periods up to 1,000,000 years. The decay calculations were made with the ORIGEN2 code using compositional data for the year 1990 as the starting point; thus the year 1990 represents the zero point for decay time. As shown in Table 3.2.4, a maximum-activity canister produced in 1990 would have a radioactivity of 125,200 Ci and a thermal power of 382 W. The same canister in 1995, or a new canister filled in 1995 from this same batch of waste, would have a maximum radioactivity of 111,100 Ci and a thermal power of 341 W. The corresponding average values of radioactivity and thermal power per canister are 112,700 Ci and 339 W for year 1990 and 100,200 Ci and 303 W for year 1995.

More detailed tables, showing the contributions of individual radionuclides, are given in Appendix 3A (Tables 3A.1-3A.6).

3.2.7 Chemical Composition

Table 3.2.5 shows the expected chemical composition of the HLW glass to be produced at WVDP and the possible range of variation of the concentrations of individual components (Eisenstatt 1986).

3.2.8 Assessment of Data

The radionuclide composition and quantity of the waste at WVDP and of the glass made from that waste are well established. Estimates of

the number of canisters to be produced range from 275 (Rykken 1987) to 300 (Bixby 1987). We used 275 since that gives a higher value for radioactivity per canister; however, the 300 estimate is more conservative from the standpoint of space requirement for the repository.

Melter feed batches are prepared individually and thus may have some variation in composition. The fill level of individual canisters also may vary. For these reasons, the maximum initial activity per canister can exceed the average by an amount estimated at 11%, as indicated in Table 3.2.4.

3.2.9 References for Section 3.2

Bixby 1987. Letter from W. W. Bixby, West Valley Project Office, to S. N. Storch, ORNL, February 27, 1987.

Eisenstatt 1986. L. R. Eisenstatt, Description of the West Valley Demonstration Project Reference High-Level Waste Form and Canister, WVDP-056, July 1986.

Rykken 1986a. Letter from L. E. Rykken, WVDP, to R. Salmon, ORNL, April 11, 1986.

Rykken 1986b. Telephone conversation, L. E. Rykken, WVDP, and R. Salmon, ORNL, April 16, 1986.

Rykken 1986c. Telephone conversation, L. E. Rykken, WVDP, and R. Salmon, ORNL, April 23, 1986.

Rykken 1987. Telephone conversation, L. E. Rykken, WVDP, and R. Salmon, ORNL, March 25, 1987.

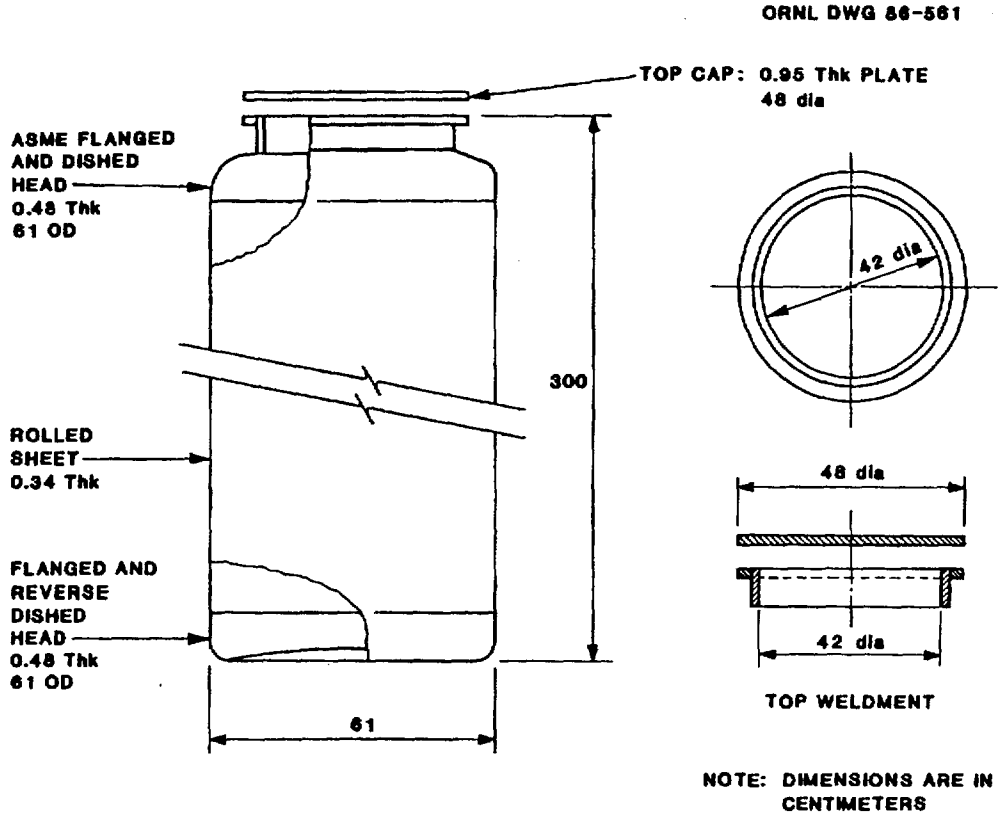

Fig. 3.2.1. West Valley Demonstration Project HLW canister. Source: Rykken 1986b.

Table 3.2.1. West Valley Demonstration Project.
High-level waste form and canister characteristics.[a]

Waste form	Borosilicate glass in closed canister
Canister material	Stainless steel type 304L
Borosilicate glass density, g/cm^3 at 25°C	2.7
Weights per canister:	
Empty canister, kg	234
Cover, kg	18
Borosilicate glass, kg	1,895
Total loaded weight, kg	2,147
Canister dimensions:	
Outside diameter, cm	61
Height overall, cm	300
Wall thickness, cm	0.34
Radionuclide content, curies per canister (1990)[b]	
Average	112,700
Maximum	125,200
Thermal power, watts per canister (1990)[b]	
Average	339
Maximum	382

[a]Source: Eisenstatt 1986 and ORNL calculations.

[b]Quantities shown are at 85% fill. Curies and watts per canister are for the year 1990.

Table 3.2.2. West Valley Demonstration Project. Estimated production schedule of canisters of HLW glass.[a]

End of calendar year	Number of canisters produced during year	Cumulative total number of canisters produced	Cumulative total glass produced (kg)
1989	0	0	0
1990	130	130	246,000
1991	145	275	520,000

[a]Based on Rykken, 1987. Canister fill volume is assumed to be 85%. Each canister contains about 1895 kg of glass.

Table 3.2.3. West Valley Demonstration Project.
Radioisotope content per HLW canister[a]

Radionuclide	Radioactivity (Ci)		
	Nominal	Range	
Fe-55	1.9E+0	1.7E+0	2.1E+0
Ni-59	3.2E-1	2.9E-1	3.6E-1
Ni-63	2.5E+1	2.3E+1	2.7E+1
Co-60	3.2E+0	2.8E+0	3.6E+0
Se-79	1.5E-2	1.3E-2	1.6E-2
Sr-90	2.7E+4	2.4E+4	3.0E+4
Y-90	2.7E+4	2.4E+4	3.0E+4
Zr-93	9.5E-1	8.5E-1	1.1E+0
Nb-93m	7.8E-1	7.0E-1	8.6E-1
Tc-99	6.7E+0	6.0E+0	7.4E+0
Ru-106	3.2E-2	2.9E-2	3.6E-2
Rh-106	3.2E-2	2.9E-2	3.6E-2
Pd-107	4.7E-3	4.2E-3	5.3E-3
Sb-125	8.4E+0	7.5E+0	9.3E+0
Te-125m	1.9E+0	1.7E+0	2.1E+0
Sn-126	1.6E-1	1.4E-1	1.8E-1
Sb-126m	1.6E-1	1.4E-1	1.8E-1
Sb-126	2.2E-1	2.0E-1	2.5E-1
Cs-134	1.5E+1	1.3E+1	1.6E+1
Cs-135	6.3E-1	5.6E-1	7.0E-1
Cs-137	2.9E+4	2.5E+4	3.2E+4
Ba-137m	2.7E+4	2.4E+4	3.0E+4
Ce-144	3.8E-3	3.4E-3	4.3E-3
Pr-144	3.8E-3	3.4E-3	4.3E-3
Pm-147	5.4E+2	5.0E+2	6.3E+2
Sm-151	8.1E+2	7.2E+2	9.0E+2
Eu-152	1.5E+0	1.3E+0	1.6E+0
Eu-154	4.0E+2	3.6E+2	4.5E+2
Eu-155	5.9E+1	5.3E+1	6.5E+1
Th-232	6.3E-3	4.7E-3	8.0E-3
U-233	3.8E-2	3.4E-2	4.2E-2
U-234	1.7E-2	1.6E-2	1.9E-2
U-235	3.9E-4	3.5E-4	4.4E-4
U-236	1.1E-3	9.9E-4	1.2E-3
U-238	3.1E-3	2.8E-3	3.5E-3
Np-237	4.3E-2	2.0E-2	6.9E-2
Np-239	9.4E+0	4.4E+0	1.5E+1
Pu-238	2.7E+1	2.4E+1	3.0E+1
Pu-239	6.8E+0	6.1E+0	7.6E+0
Pu-240	1.5E+1	8.7E+0	1.9E+1
Pu-241	3.0E+2	2.6E+2	3.3E+2
Pu-242	6.8E-3	6.0E-3	7.5E-3
Am-241	3.4E+2	1.7E+2	5.0E+2
Am-242	8.3E-2	3.8E-2	1.3E-1
Am-242m	8.3E-2	3.8E-2	1.3E-1
Am-243	9.4E+0	4.4E+0	1.5E+1
Cm-242	8.3E-2	3.8E-2	1.3E-1
Cm-243	6.2E-1	3.0E-1	1.0E+0
Cm-244	7.8E+1	3.7E+1	1.2E+2
Cm-245	3.9E-2	1.9E-2	6.3E-2
Cm-246	1.7E-2	8.0E-3	2.7E-2

[a]Source: Eisenstatt 1986. Quantities shown are for the year 1990 and are based on a canister containing 1895 kg of HLW glass.

Table 3.2.4. West Valley Demonstration Project. Calculated radioactivity and thermal power per HLW canister.[a]

Decay time after 1990, years	Radioactivity per canister (Ci)		Thermal power per canister (W)	
	Average	Maximum	Average	Maximum
0	112,700	125,200	339	382
1	110,400	122,400	333	375
2	107,700	119,500	325	366
5	100,200	111,100	303	341
10	89,040	98,750	270	305
15	79,190	87,840	241	272
20	70,480	78,180	216	244
30	55,870	62,000	173	196
50	35,220	39,120	112	128
100	11,350	12,690	42	50
200	1,510	1,770	12.6	17.2
300	450	587	8.6	12.3
350	339	457	7.8	11.2
500	226	317	6.2	9.0
1,000	119	168	3.3	4.7
1,050	113	159	3.1	4.4
2,000	59.3	80.8	1.4	1.9
5,000	38.0	50.2	0.72	0.96
10,000	29.5	38.1	0.51	0.69
20,000	20.3	25.3	0.30	0.39
50,000	12.3	14.4	0.11	0.14
100,000	9.2	10.7	0.05	0.06
500,000	4.6	5.6	0.03	0.04
1,000,000	3.0	3.8	0.03	0.04

[a]Calculations made with ORIGEN2 code based on data supplied by WVDP (Eisenstatt 1986). Canister contains 1895 kg of HLW glass. Initial time point (0 years) is at year 1990.

Table 3.2.5. West Valley Demonstration Project. Chemical composition of reference HLW glass[a]

Component	Nominal composition (wt %)	Range (wt %)	
AgO	0.0001	-	-
Al_2O_3	2.8295	1.19	7.15
AmO_2	0.0073	-	-
BaO	0.0540	0.04	0.08
B_2O_3	9.9516	9.33	10.66
CaO	0.5993	0.39	0.93
CdO	0.0003	-	-
CeO_2	0.0670	0.04	0.10
CmO_2	0.0001	-	-
CoO	0.0002	-	-
Cr_2O_3	0.3112	0.21	0.48
Cs_2O	0.0826	0.05	0.13
CuO	0.0001	-	-
Eu_2O_3	0.0014	-	-
Fe_2O_3	12.1573	8.32	18.50
Gd_2O_3	0.0003	-	-
In_2O_3	0.0001	-	-
K_2O	3.5733	3.36	3.84
La_2O_3	0.0337	0.02	0.05
Li_2O	3.0315	2.84	3.25
MgO	1.3032	1.22	1.39
MnO_2	1.3107	0.84	1.96
MoO_3	0.0088	-	0.01
NaCl	0.0183	0.01	0.03
NaF	0.0013	-	-
Na_2O	10.9335	10.25	11.71
Nd_2O_3	0.1209	0.08	0.19
NiO	0.3358	0.22	0.52
NpO_2	0.0224	0.01	0.03
P_2O_5	2.5084	0.21	3.16
PdO	0.0062	-	-
Pm_2O_3	0.0003	-	-
Pr_6O_{11}	0.0321	0.02	0.05
PuO_2	0.0076	-	-
Rb_2O	0.0005	-	-
RhO_2	0.0136	0.01	0.02
RuO_2	0.0759	0.05	0.12
SO_3	0.2164	0.14	0.33
Sb_2O_3	0.0001	-	-
SeO_2	0.0005	-	-
SiO_2	44.8770	42.08	48.10
Sm_2O_3	0.0267	0.02	0.04
SnO_2	0.0006	-	-
SrO	0.0269	0.02	0.04
Tc_2O_7	0.0021	-	-
ThO_2	3.5844	1.83	6.56
TeO_2	0.0028	-	-
TiO_2	0.9800	0.92	1.05
UO_2	0.5605	0.37	0.87
Y_2O_3	0.0177	0.01	0.03
ZnO	0.0010	-	-
ZrO_2	0.2943	0.19	0.45
Insolubles	0.0080	-	-

[a]Source: Eisenstatt 1986. Reference glass composition is WV-205.

3.3 SAVANNAH RIVER PLANT (SRP)

3.3.1 Introduction

Interim forms of high-level waste now in storage at SRP have been produced since 1954 by the reprocessing of defense reactor fuels. Neutralization and settling of the HLW have resulted in the formation of sludge and supernatant liquid. Subsequent evaporation of the supernatant liquid, which contains almost all of the Cs-137 activity, has produced a saturated salt solution and a saltcake consisting of the salts crystallized out of the saturated solution. Starting in 1990, the sludge and most of the radioactivity in the supernatant liquid will be processed in the Defense Waste Processing Facility (DWPF) to produce canisters of borosilicate glass in which the HLW is dispersed and immobilized. Processing of decontaminated salt solution into saltstone will be started in 1988; the saltstone is low-level waste and will go to onsite engineered storage.

3.3.2 Types of HLW Produced

The glass to be produced at the DWPF is referred to as sludge-precipitate glass and will be made from a blend of (1) washed sludge, (2) washed precipitate made by treating the salt solution to precipitate cesium together with smaller quantities of other radionuclides, and (3) glass frit. The salt solution will include salts redissolved out of the saltcake phase; thus the washed precipitate will contain essentially all of the radioactivity originally in the supernate. A more complete description of the feed preparation process is given in Appendix 3B.

3.3.3 Physical Description

Design details of the DWPF canister are shown in Figs. 3.3.1 and 3.3.2 and Table 3.3.1. The main body of the canister is made of schedule 20 type 304L stainless steel pipe with an outside diameter of 61 cm and a wall thickness of 0.95 cm. The overall length of the canister is

300 cm (9 ft 10 in.). The nominal inside volume is about 0.74 m^3, and the weight of the empty canister is about 500 kg (1100 lb). Each canister will contain 0.626 m^3 of glass, or about 1680 kg (3710 lb), when loaded to about 85% of its volume at an average glass temperature of 825°C. The density of the reference glass is about 2.7 g/cm^3 at this temperature; the density at 25°C is about 2.85 g/cm^3. The total weight of a loaded canister is therefore about 2180 kg (4810 lb), and the volume of glass in a loaded canister at 25°C, based on density ratio, would be about 0.59 m^3; however, the actual glass level in the canister is essentially unchanged (Kelker 1986; DPSP 80-1033, Rev. 91).

3.3.4 Inventory and Production Schedule

Table 3.3.2 shows a preliminary projection of glass production from 1990 to 2020 estimated by SRP for the 1987 Integrated Data Base submittal (Boore 1987). As shown in the table, the initial production of glass at SRP is scheduled to start in 1990, and it is estimated that about 102 canisters will be produced in that year. A total of about 6800 canisters will have been produced by the end of 2020. All canisters produced will be stored on site until a repository becomes available.

3.3.5 Radionuclide Content per Canister

Existing tanks at SRP contain blends of waste of all ages. From the standpoint of minimizing the radioactivity of the glass, it would be desirable to vitrify the waste in the oldest tanks first. However, this is not feasible because of practical contraints in the waste tank farms. Some of the tanks currently receiving fresh waste from the fuel reprocessing facilities will be nearly full by the time the DWPF begins operation. Since it is essential to have tankage space available to receive current production, it will be necessary to process some of the fresher waste first. It has not yet been possible for SRP to prepare a complete characterization of the feed to the vitrification plant in terms of an estimated schedule of radionuclide content versus time. It is clear, however, that the oldest waste will not be vitrified first.

Although present plans are to prepare the initial feed batch from sludge of lower activity, the activity of the cesium precipitate feed will probably be close to the DWPF flowsheet maximum because of processing constraints in the tank farm.

The radionuclide composition estimated by SRP to represent the most highly radioactive glass likely to be made from sludge-supernate processing is shown in Table 3.3.3; this was based on data in DPSP 80-1033, Rev. 91, and is the best current estimate of maximum activity per canister. Table 3.3.3 is based on sludge aged an average of 5 years and a cesium-containing precipitate derived from supernate aged an average of 15 years. The radionuclide content of sludge-precipitate glass is shown in terms of curies and grams per canister; this was based on 1682 kg (3710 lb) of sludge-precipitate glass at the reference-case fill level of 85%.

3.3.6 Radioactivity and Thermal Power

The maximum expected values of radioactivity and thermal power per canister as a function of decay time after filling were determined by ORIGEN2 calculations based on the radionuclide content per canister shown in Table 3.3.3. The results are shown in Table 3.3.4 in summary form. The total activity and decay heat at the time of filling are 234,400 Ci and 709 W per canister. Detailed tables showing the contributions of individual radionuclides to total curies and watts per canister over a time span of 0 to 10^6 years are given in Appendix 3A.

Recent curie balances indicate that the glass produced during the first five years of operation will not exceed an activity of about 154,000 Ci/canister and a heat generation rate of about 460 W/canister (Baxter 1986). However, the detailed radionuclide composition of this glass will not be available until just before it is processed. Washed sludge for the initial feed blend has just been collected but has not been analyzed, and the washed precipitate for initial feed blending has not yet been produced. Estimates of the analysis of the first feed batch should be available by 1988 and about one year prior to feeding for subsequent feed batches.

Table 3.3.5 shows the estimated average radioactivity and thermal power per canister on a cumulative year-by-year basis. The average radioactivity of canisters produced through the year 2020 is considerably less than the maximum radioactivity per canister shown in Table 3.3.3. For example, at the end of year 2020 the total cumulative radioactivity in glass is 404.2×10^6 Ci. Dividing this by the total number of canisters produced (6810), the resulting average is 59,400 Ci/canister. The average thermal power, determined in the same way, is 169 W/canister. The SRP projections on which Table 3.3.5 was based were given in Chandler 1987. The reader is cautioned that these projections were not intended to represent actual processing schedules and tankage allocations and therefore should not be used to calculate radioactivity or thermal power per canister in any specific year. However, the long-term cumulative averages shown should be useful for repository calculations, since it is clear that the averages should give better estimates of overall heat loads than would be obtained by multiplying the total number of canisters by the maximum heat load per canister.

3.3.7 Chemical Composition

Table 3.3.6 shows the approximate chemical composition of a simulated average borosilicate glass from SRP (Chandler 1987).

3.3.8 Assessment of Data

The data in DPSP 80-1033, Rev. 91, are the best available at present for the estimation of maximum radioactivity and thermal power per canister. Based on these data and a canister loading of 1682 kg, it appears that the maximum values of radioactivity and thermal power per canister will not exceed those shown in Table 3.3.4. For repository design and other purposes, it would be useful to have an estimated schedule of the radionuclide content of the vitrification plant feed as a function of the year of operation; but, as already indicated, such estimates will not be available until about one year before feeding to

the vitrification plant. It appears likely that the glass produced during the first 5 years of operation will not exceed about 154,000 Ci/canister and 460 W/canister. SRP is continually working to update waste treatment and vitrification process flowsheets, mass balances, and curie balances, so the estimates given here are subject to revision as new data become available.

3.3.9 References for Section 3.3

Baxter 1986. Telephone conversation, R. G. Baxter (SRP) and R. Salmon (ORNL), May 16, 1986.

Baxter 1987. Letter from R. G. Baxter, SRP, to Royes Salmon, ORNL, February 18, 1987.

Boore 1987. Letter from W. B. Boore, SRP, to M. G. O'Rear, SRO, March 10, 1987.

Chandler 1987. Letter from R. L. Chandler to M. W. Shupe, HLW Lead Office, Richland, transmitting SRP input to DHLW Integrated Data Base, April 1, 1987.

DPSP 80-1033, Rev. 91. DWPF Basic Data Report, DPSP-80-1033, Rev. 91, April 1985.

Kelker 1986. J. W. Kelker, Jr., Development of the DWPF Canister Temporary Shrink-fit Seal, DP-1720, April 1986.

SRP 1987. Data transmittal at meeting at SRP, March 10, 1987.

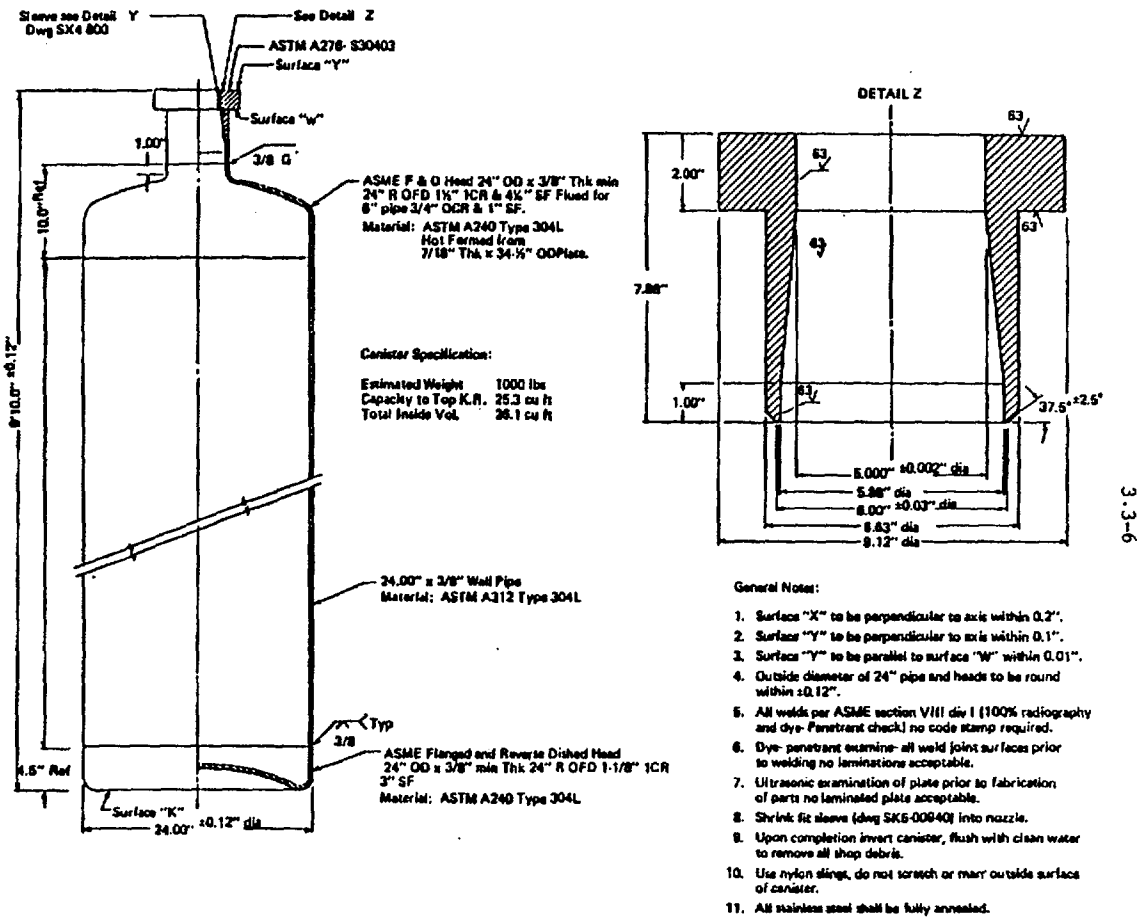

Fig. 3.3.1. Savannah River Plant HLW canister. Source: DuPont report DP-1720, April 1986.

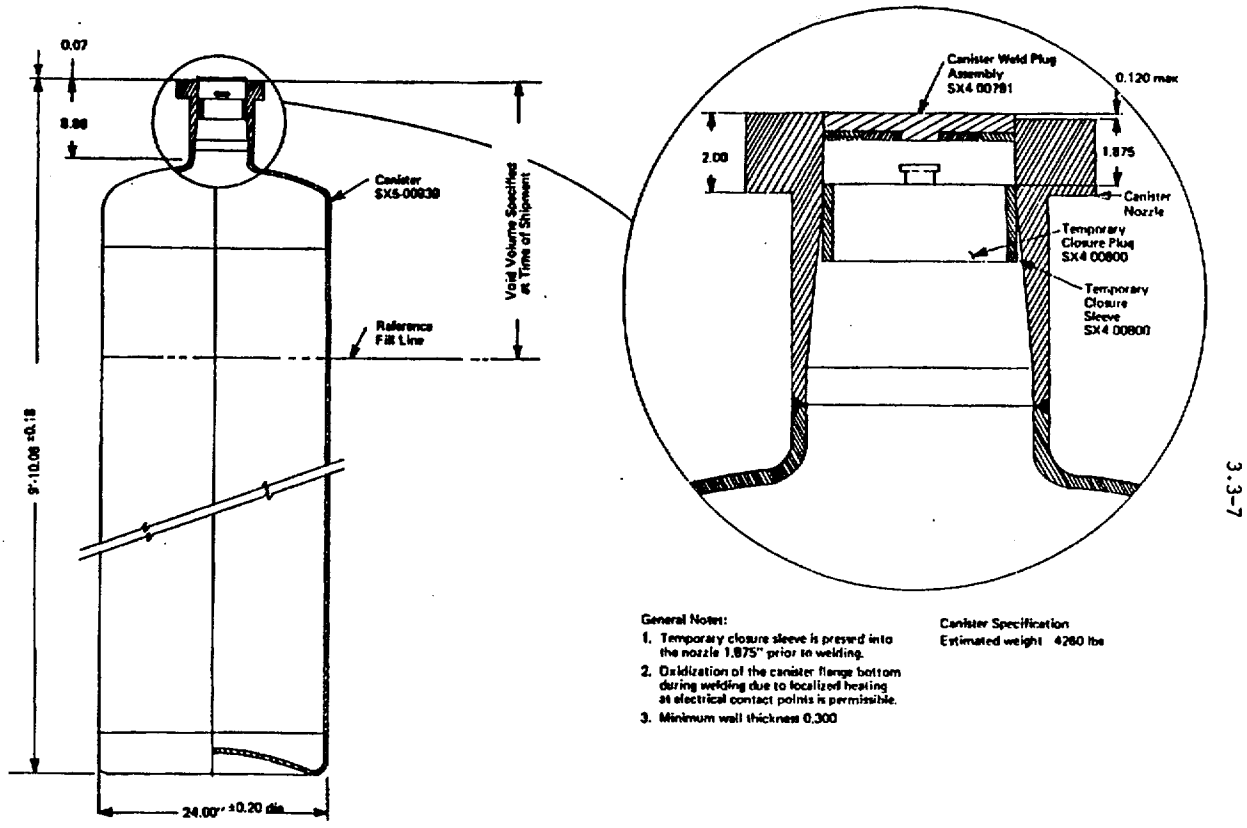

Fig. 3.3.2. Savannah River Plant HLW canister closure. Source: DuPont report DP-1720, April 1986.

Table 3.3.1. Savannah River Plant. High-level waste form and canister characteristics.[a]

	Canister 85% fill
Canister inside volume, m^3	0.736
Glass volume at average fill temperature (see note b), m^3	0.626
Glass density at average fill temperature (see note b), g/cm^3	2.69
Glass weight, kg	1,682
Canister weight, kg	500
Gross weight, kg	2,182
Total activity, curies	234,000[c]
Decay heat, watts	690[c]

[a]Source: DWPF Basic Data Report, DPSP 80-1033, Rev. 91, April 1985.

[b]The average fill temperature (i.e. the average temperature of the glass upon completion of filling to 85% of canister volume) is 825°C. The glass volume per canister when cooled to 25°C is about 0.59 m^3. The density of the glass is about 2.69 g/cm^3 at 825°C and 2.85 g/cm^3 at 25°C (SRP 1987).

[c]These figures are the ones given in DPSP 80-1033, Rev. 91. The corresponding figures calculated by ORIGEN2 are 234,400 Ci and 709W, as shown in Table 3.1. Activity and decay heat (thermal power) are at the time of filling the canister and are based on the maximum case, i.e. 5-yr old sludge and 15-yr old supernate.

Table 3.3.2. DWPF, Savannah River Plant.
Estimated production schedule of canisters of HLW glass.[a]

End of calendar year	Number of canisters produced during year	Cumulative number of canisters produced	Cumulative volume of glass produced $10^3 m^3$
1987	0	0	0
1988	0	0	0
1989	0	0	0
1990	102	102	0.06
1991	410	512	0.32
1992	410	922	0.58
1993	410	1332	0.84
1994	410	1742	1.10
1995	410	2152	1.36
1996	410	2562	1.62
1997	410	2972	1.88
1998	410	3382	2.14
1999	410	3792	2.40
2000	410	4202	2.66
2001	220	4422	2.80
2002	220	4642	2.94
2003	220	4862	3.08
2004	220	5082	3.22
2005	220	5302	3.36
2006	220	5522	3.50
2007	92	5614	3.56
2008	92	5706	3.62
2009	92	5798	3.68
2010	92	5890	3.74
2011	92	5982	3.80
2012	92	6074	3.86
2013	92	6166	3.92
2014	92	6258	3.98
2015	92	6350	4.04
2016	92	6442	4.10
2017	92	6534	4.16
2018	92	6626	4.22
2019	92	6718	4.28
2020	92	6810	4.34

[a]Production shown is based on a glass melt rate of 228 lb/hr and 75% attainment. Canisters (2-ft diameter x 10-ft long) are assumed to be filled to 85% capacity with a glass waste form incorporating 28 wt% waste sludge oxides, 8 wt% residues from waste salt, and 64 wt% oxides from a nonradioactive frit. Volumes reported are for the glass waste form and not the canisters. Source: Chandler 1987.

Table 3.3.3. Savannah River Plant. Radioisotope content per HLW canister[a]

	Isotope	Curies/canister	Grams/canister
1	Cr-51	0.9312E-16	0.1008E-20
2	Co-60	0.1699E+03	0.1502E+00
3	Ni-59	0.2397E-01	0.3163E+00
4	Ni-63	0.2975E+01	0.4824E-01
5	Ti-208	0.1128E-02	0.3829E-11
6	U-232	0.1339E-01	0.6256E-03
7	U-233	0.1584E-05	0.1636E-03
8	U-234	0.3428E-01	0.5485E+01
9	U-235	0.1573E-03	0.7278E+02
10	U-236	0.1128E-02	0.1742E+02
11	U-238	0.1050E-01	0.3122E+05
12	Np-236	0.1744E-07	0.1323E-05
13	Np-237	0.8904E-02	0.1263E+02
14	Pu-236	0.1221E+00	0.2297E-03
15	Pu-237	0.8941E-11	0.7401E-15
16	Pu-238	0.1484E+04	0.8667E+02
17	Pu-239	0.1291E+02	0.2076E+03
18	Pu-240	0.8681E+01	0.3809E+02
19	Pu-241	0.1670E+04	0.1620E+02
20	Pu-242	0.1224E-01	0.3206E+01
21	Am-241	0.1102E+02	0.3210E+01
22	Am-242	0.1436E-01	0.1776E-07
23	Am-242m	0.1447E-01	0.1488E-02
24	Am-243	0.5788E-02	0.2902E-01
25	Cm-242	0.3495E-01	0.1057E-04
26	Cm-243	0.5565E-02	0.1078E-03
27	Cm-244	0.1076E+03	0.1329E+01
28	Cm-245	0.6715E-05	0.3910E-04
29	Cm-246	0.5342E-06	0.1739E-05
30	Cm-247	0.6604E-12	0.7116E-08
31	Cm-248	0.6864E-12	0.1614E-09
32	Se-79	0.1699E+00	0.2439E+01
33	Rb-87	0.8719E-06	0.9961E+01
34	Sr-89	0.4267E-04	0.1470E-08
35	Sr-90	0.4675E+05	0.3426E+03
36	Y-90	0.4786E+05	0.8795E-01
37	Y-91	0.7568E-03	0.3085E-07
38	Zr-93	0.1117E+01	0.4443E+03
39	Zr-95	0.1005E-01	0.4680E-06
40	Nb-94	0.9646E-04	0.5147E-03
41	Nb-95	0.2115E-01	0.5407E-06
42	Nb-95m	0.1247E-03	0.3272E-09
43	Tc-99	0.3079E+01	0.1816E+03
44	Ru-103	0.1684E-07	0.5217E-12
45	Ru-106	0.2252E+04	0.6729E+00

Table 3.3.3 (continued)

	Isotope	Curies/canister	Grams/canister
46	Rh-103m	0.1636E-07	0.5028E-15
47	Rb-106	0.2259E+04	0.6346E-06
48	Pd-107	0.1473E-01	0.2863E+02
49	Ag-110m	0.1258E+00	0.2647E-04
50	Cd-113	0.5009E-13	0.1472E+00
51	Cd-115m	0.1213E-08	0.4763E-13
52	Sn-121m	0.7902E-01	0.1336E-02
53	Sn-123	0.2549E+00	0.3101E-04
54	Sn-126	0.4415E+00	0.1556E+02
55	Sb-124	0.7123E-07	0.4071E-11
56	Sb-125	0.8496E+03	0.8226E+00
57	Sb-126	0.6159E-01	0.7365E-06
58	Sb-126m	0.4415E+00	0.5619E-08
59	Te-125m	0.2760E+03	0.1532E-01
60	Te-127	0.1202E+00	0.4555E-07
61	Te-127m	0.1228E+00	0.1302E-04
62	Te-129	0.3053E-11	0.1457E-18
63	Te-129m	0.4749E-11	0.1576E-15
64	Cs-134	0.3372E+03	0.2606E+00
65	Cs-135	0.9943E-01	0.8633E+02
66	Cs-136	0.7828E-39	0.1068E-43
67	Cs-137	0.4341E+05	0.4989E+03
68	Ba-136m	0.8607E-38	0.3195E-49
69	Ba-137m	0.4155E+05	0.7724E-04
70	Ba-140	0.1024E-35	0.1404E-40
71	La-140	0.4304E-36	0.7734E-42
72	Ce-141	0.3591E-10	0.1260E-14
73	Ce-142	0.9609E-05	0.4005E+03
74	Ce-144	0.9869E+04	0.3093E+01
75	Pr-143	0.1198E-33	0.1780E-38
76	Pr-144	0.9869E+04	0.1306E-03
77	Pr-144m	0.1187E+03	0.6545E-06
78	Nd-144	0.4860E-09	0.4110E+03
79	Nd-147	0.1261E-43	0.1570E-48
80	Pm-147	0.2419E+05	0.2609E+02
81	Pm-148	0.6975E-10	0.4243E-15
82	Pm-148m	0.1009E-08	0.4722E-13
83	Sm-147	0.2000E-05	0.8796E+02
84	Sm-148	0.5788E-11	0.1916E+02
85	Sm-149	0.1781E-11	0.7420E+01
86	Sm-151	0.2478E+03	0.9418E+01
87	Eu-152	0.3688E+01	0.2132E-01
88	Eu-154	0.6196E+03	0.2295E+01
89	Eu-155	0.4749E+03	0.1021E+01
90	Eu-156	0.5231E-31	0.9489E-36
91	Tb-160	0.1120E-05	0.9923E-10
	Total	0.2344E+06	0.3427E+05

[a]Quantities shown are for sludge + supernate glass and are based on the DWPF Basic Data Report, DPSP 80-1033, Rev. 91, April 1985, assuming sludge aged an average of 5 years and supernate aged an average of 15 years, with a canister load of 3710 lb of glass (1683 kg).

Table 3.3.4. Savannah River Plant. Calculated radioactivity and thermal power per HLW canister.[a]

Decay time, years[b]	Radioactivity per canister (Ci)	Thermal power per canister (W)
0	234,400	709
1	208,500	627
2	193,800	586
5	169,300	527
10	145,800	467
15	128,400	418
20	113,900	374
30	90,000	301
50	56,500	198
100	17,900	75
200	2,100	17
300	390	7.2
350	227	5.2
500	95	2.7
1,000	42	1.1
1,050	41	1.1
2,000	29	0.72
5,000	24	0.54
10,000	20	0.43
20,000	16	0.30
50,000	11	0.16
100,000	9.2	0.11
500,000	4.8	0.05
1,000,000	2.4	0.02

[a]Based on 5-yr cooled sludge and 15-yr cooled supernate. Calculations made by ORIGEN2 code based on data supplied by SRP (Basic Data Report, DPSP-80-1033, Rev. 91, April 1985). Canister is filled to 85% of capacity and contains 1683 kg of glass.

[b]Years after vitrification.

Table 3.3.5. Savannah River Plant. Estimated cumulative average radioactivity and thermal power per canister of HLW glass[a]

End of calendar year	Cumulative number of canisters produced	Cumulative radioactivity		Cumulative thermal power	
		Total (10^6 Ci)	per canister (Ci)	Total (10^3 W)	per canister (W)
1987	0	0	0	0	0
1988	0	0	0	0	0
1989	0	0	0	0	0
1990	102	6.1	59,800	17.3	170
1991	512	27.0	52,700	78.6	154
1992	922	50.8	55,100	143.3	155
1993	1,332	74.2	55,700	208.4	156
1994	1,742	98.4	56,500	277.3	159
1995	2,152	127.9	59,400	359.9	167
1996	2,562	156.5	61,100	438.5	171
1997	2,972	171.7	57,800	482.7	162
1998	3,382	192.8	57,000	541.9	160
1999	3,792	214.3	56,500	602.6	159
2000	4,202	242.0	57,600	681.1	162
2001	4,422	249.0	56,300	701.6	159
2002	4,642	251.3	54,100	708.2	153
2003	4,862	256.9	52,800	724.8	149
2004	5,082	261.4	51,400	738.3	145
2005	5,302	268.5	50,600	759.7	143
2006	5,522	276.7	50,100	784.1	142
2007	5,614	277.9	49,500	787.6	140
2008	5,706	281.4	49,300	798.1	140
2009	5,798	283.8	49,000	805.2	139
2010	5,890	287.1	48,700	815.3	138
2011	5,982	287.7	48,100	817.1	137
2012	6,074	288.3	47,500	818.9	135
2013	6,166	288.9	46,900	820.7	133
2014	6,258	289.4	46,200	822.2	131
2015	6,350	291.1	45,800	825.4	130
2016	6,442	292.0	45,300	827.9	129
2017	6,534	292.6	44,800	829.6	127
2018	6,626	293.1	44,200	831.1	125
2019	6,718	293.7	43,700	832.9	124
2020	6,810	294.3	43,200	834.6	123

[a]Calculated from estimates given in Chandler 1987. Year-by-year radioactivity and thermal power per canister do not necessarily represent actual processing schedules and tankage allocations and should not be used for design purposes. Radioactivity and thermal power shown are for fission products only. Radioactivity will be about 1% higher and thermal power about 6% higher when actinides are included.

Table 3.3.6. Savannah River Plant. Chemical composition of HLW glass[a]

Component	wt %
SiO_2	45.6
Na_2O	11.0
B_2O_3	10.3
Fe_2O_3	7.0
Al_2O_3	4.0
K_2O	3.6
Li_2O	3.2
FeO	3.1
U_3O_8	2.2
MnO	2.0
Other	8.0
Total	100.0

[a]Source: Chandler 1987.

3.4 HANFORD SITE (HANF)

3.4.1 Introduction

The HLW currently stored at HANF was generated by the reprocessing of irradiated fuel from production reactors for the recovery of uranium, plutonium, and other elements. The Hanford Waste Vitrification Plant (HWVP) is now in the preliminary conceptual design stage. Procurement and construction are scheduled to begin in 1989, and hot startup is scheduled for 1996. The plant will vitrify pretreated HLW in a borosilicate glass which will be cast into stainless steel canisters. Maximum use will be made of existing technology, such as that developed in the design of the Defense Waste Processing Facility at SRP.

3.4.2 Types of HLW Produced

Current plans are that the HWVP will produce vitrified waste of three different compositions, corresponding to three different feeds. These are known as neutralized current acid waste (NCAW), complexant concentrate (CC), and plutonium finishing plant waste (PFP). The NCAW has a much higher activity than the CC/PFP and may therefore be considered as the design basis feed for repository purposes. It is possible that the CC and PFP may be combined and run as a single feed. The two or three HLW borosilicate glasses from these operations are the major HLW forms with which the repository will be concerned. The only other HLW forms produced at HANF of possible interest to the repository are strontium and cesium capsules. These are discussed in Section 5.5 and in Appendix 3B.

It is assumed here that the HANF reference plan will be followed. In this plan, the single-shell tank wastes are not vitrified but are immobilized in place. The decision as to whether the single-shell tank (SST) wastes are vitrified is dependent on the outcome of the Environmental Impact Statement (EIS) process. If the SST wastes are vitrified, these canisters would also go to a repository.

3.4.3 Physical Description

The canisters are made of type 304L stainless steel pipe with an outside diameter of 61 cm and a length of 300 cm. Figures 3.4.1 and 3.4.2 are sketches of the canister and neck detail showing relevant dimensions. The canister is essentially identical to that planned for use at the Savannah River DWPF. Additional descriptive information on the canister and HLW glass is given in Table 3.4.1. The fill level of the HWVP canister is approximately 85% of the available internal canister fill volume, resulting in a canister glass volume of 0.62 m^3 (22 ft^3) which is equivalent to a glass height of 2.3 m (7.5 ft). A 15% void volume minimizes the potential of canister overfill.

The density of the HWVP glass is 2.64 g/cm^3 (165 lb/ft^3). A glass volume of 0.62 m^3 (22 ft^3) corresponds to a glass weight of 1650 kg (3630 lb) (White 1986). The total weight of the filled canister is approximately 2150 kg (4740 lb), assuming that the empty canister weighs 500 kg, in accordance with SRP's estimate.

3.4.4 Inventory and Production Schedule

Estimated annual canister production rates for the vitrified waste are shown in Table 3.4.2. The HWVP is planned to start up in FY 1996. The HWVP design throughput is 145 canisters per year. After every three years of HWVP operations, there is a six-month shutdown for melter change-out. Table 3.4.2 indicates half of a normal year's production in the years in which melter change-out occurs. Rockwell is currently planning a production of 930 canisters of vitrified NCAW, 580 canisters of complexant concentrate, and 350 canisters of plutonium finishing plant waste (Coony 1987).

3.4.5 Radionuclide Content per Canister

Radioisotopic data describing the composition of a canister of glass made from NCAW were supplied by Rockwell Hanford (White 1986). These data, shown in Table 3.4.3, were intended as the upper bound of activity and thermal power and represent the most active waste expected

to be fed to the vitrification plant; this type of feed would not be encountered before 1999 (Watrous, 1986). The radionuclide composition of the NCAW glass (upper bound case) given in the 1987 Integrated Data Base submittal (Coony 1987) is identical to that given in White 1986. Because of the upper-bound conservatism in both the values of maximum curies per canister and number of canisters, the product of these two values will be higher than the values for curies given in Coony 1987.

The current reference plan at HANF is not to produce any more Sr and Cs capsules; none of these have been produced since 1985. The Sr and Cs in the HLW will become part of the NCAW HLW glass. The radionuclide composition shown in Table 3.4.3 is based on this assumption. Currently there are no plans to vitrify any Sr and Cs capsules; current plans provide for enclosing these capsules in overpacks for repository emplacement, as described in Sect. 5.5. At present there are 640 Sr and 1576 Cs capsules.

3.4.6 Radioactivity and Thermal Power

Based on the upper-bound isotopic data supplied by Rockwell Hanford (Coony 1987), ORIGEN2 calculations were made to determine the estimated radioactivity and thermal power per canister of HLW glass made from NCAW. Table 3.4.4 shows the calculated radioactivity and thermal power per canister for decay times ranging from 0 to 10^6 years.

Because radionuclide compositions of the glasses produced from CC and PFP are not available, no calculations of radioactivity and thermal power per canister as functions of decay time have been made for glasses produced from those streams.

More detailed tables showing the contributions of individual radionuclides to the radioactivity and thermal power of the upper-bound NCAW glass on a per-canister basis for decay times from 0 to 10^6 years are given in Appendix 3A.

During the first three years of vitrification plant operation, it is not expected that the radioactivity of the NCAW glass will be as high

as that estimated for the upper bound case. At present the best information on the glass produced during initial operations is that decay heat loads per canister will be in the range of 400 to 800 watts rather than the 1150 watts shown for the upper-bound case (Mitchell 1986).

Hanford has also provided estimates of annual and cumulative radioactivities of the vitrified waste on a year-by-year basis from 1996 to 2020. These estimates are shown in Tables 3.4.5 and 3.4.6. Table 3.4.5 shows the average radioactivity per canister on an annual "as produced" basis, and Table 3.4.6 shows the average radioactivity per canister on a cumulative basis; the amounts of radioactivity per canister were calculated by dividing Hanford's estimates of annual or cumulative radioactivity in vitrified form by the annual or cumulative number of canisters. The table shows a maximum of 324,000 Ci/canister for NCAW in 1996, about 20% less than the "upper bound" estimate of 416,500 Ci/canister. However, the radioactivities per canister shown for specific years should probably not be taken too literally since it is doubtful that actual melter feed batch scheduling could be projected accurately over the time span indicated. Complexant concentrate and plutonium finishing plant wastes appear to have radioactivities of about 220 and 60 Ci/canister, respectively.

3.4.7 Chemical Composition

The reference NCAW glass composition, designated HW-39, is shown in Table 3.4.7. Because of the radioactive nature of the waste, glass formulation and process development studies were conducted with a simulated or substituted NCAW. The elements that were substituted or deleted, as well as the glass frit composition, are indicated in the table. The final glass composition is based on 25 wt % waste oxides and 75 wt % glass frit. The frit composition will be modified as necessary to accommodate variations in NCAW composition (Mitchell 1986).

3.4.8 Assessment of Data

The upper-bound and average radioactivities of the NCAW glass have been established to the extent possible at the present time. Additional

information on the glass made from plutonium finishing plant waste and complexant concentrate would also be useful, since approximately 900 canisters of this glass will be produced; however, as indicated in Table 3.4.5, the radioactivity per canister is very low for these glasses (about 220 Ci/canister for CC waste and 60 Ci/canister for PFP waste). Maximum thermal power per canister of NCAW glass has been determined, but average thermal power has not. As a rough preliminary approximation, it could be assumed that the average and maximum thermal power are in the same ratio as the average and maximum radioactivities. However, this approximation is not recommended for long decay times because of changes in the relative importance of fission products and actinides.

3.4.9 References for Section 3.4

Coony 1986. Telephone conversation, M. R. Coony (Rockwell Hanford) and R. Salmon, ORNL, June 18, 1986.

Coony 1987. F. M. Coony, Rockwell Hanford, submission of Hanford HLW data to IDB, March 1987.

Mitchell 1986. D. E. Mitchell, Hanford Waste Vitrification Plant, Preliminary Description of Waste Form and Canister, RHO-RE-SR-55P, August 1986.

Watrous 1986. Telephone conversation, R. L. Watrous (Rockwell Hanford) and R. Salmon, ORNL, July 23, 1986.

White 1986. Letter from J. D. White, Richland Operations Office, to W. R. Bibb, DOE/ORO, dated July 3, 1986.

Wolfe 1985a. Personal communication, B. A. Wolfe, Rockwell Hanford Operations, to R. Salmon, ORNL, November 8, 1985.

Wolfe 1985b. Personal communication, B. A. Wolfe, Rockwell Hanford Operations, to R. Salmon, ORNL, November 12, 1985.

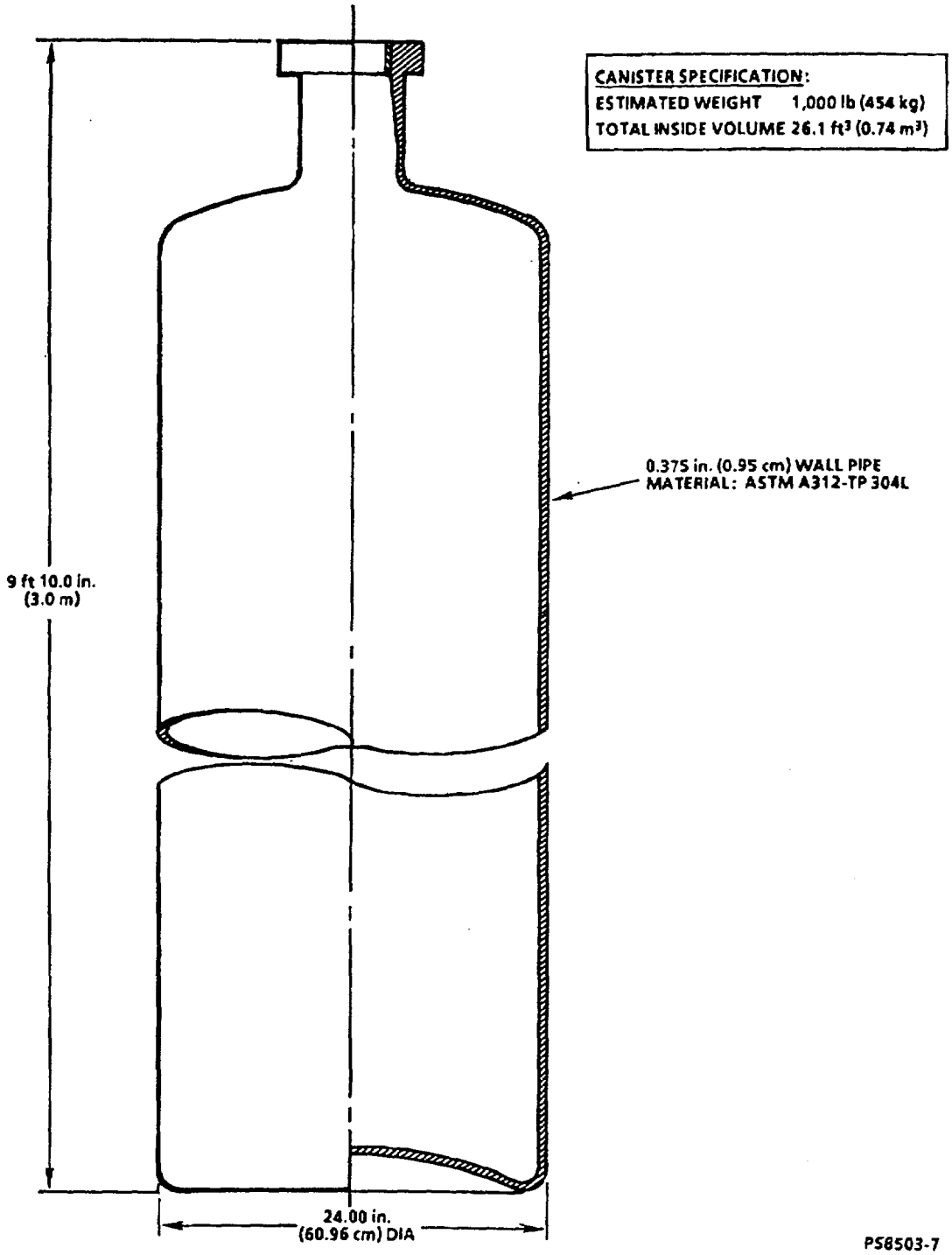

Fig. 3.4.1. Hanford HLW canister. Source: White 1986.

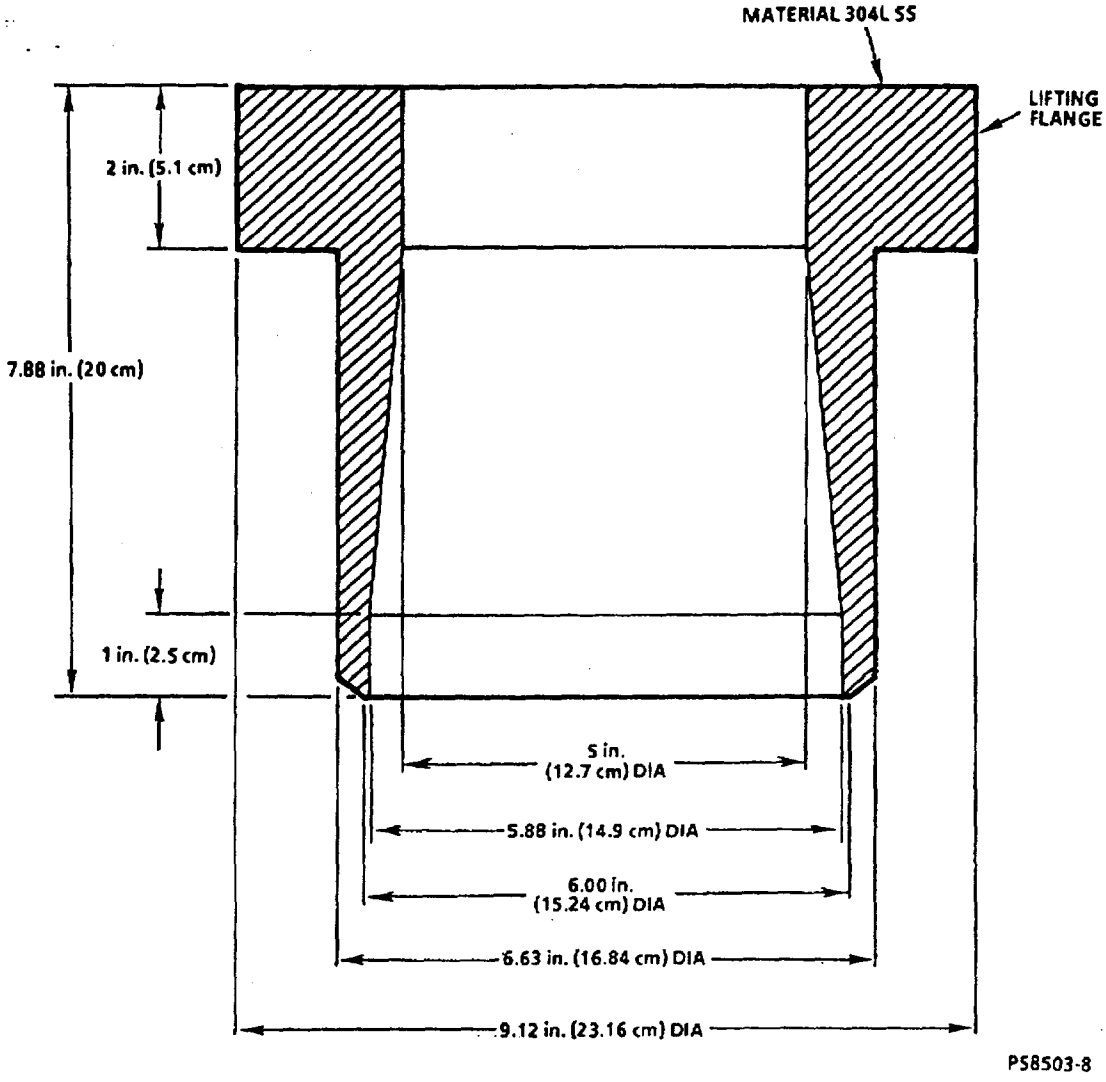

Fig. 3.4.2. Hanford HLW canister neck detail. Source: White 1986.

Table 3.4.1. Hanford Operations. High-level waste form and canister characteristics[a]

Waste form	Borosilicate glass in closed steel canister
Canister material	Type 304L stainless steel
Weights per canister	
Empty canister, kg	500
Borosilicate glass, kg	1650
Total loaded weight, kg	2150
Canister dimensions	
Outside diameter, cm	61
Height overall, cm	300
Wall thickness, cm	0.95
Inside volume, m^3	0.736
Glass volume at average fill temperature, m^3	0.626[b]
Radionuclide content, curies per canister[c]	126,000 - 478,000
Thermal power, watts per canister[c]	354 - 1750

[a]Sources: Wolfe 1985, White 1986, Mitchell 1986.

[b]Canister is filled to 85% of volume at average fill temperature of 825°C.

[c]All values shown are based on NCAW reference feed (neutralized current acid waste) with 25% wt waste oxide in glass. Activities and thermal power are at time of filling canister. Range of values shown is from Mitchell 1986 in which estimated activities and radionuclide compositions were given for four feeds typical of production during the period from 1996 to 2000. Radionuclide compositions are shown in Table 3.20.

Table 3.4.2. Hanford Operations. Estimated production schedule of canisters of HLW glass[a]

End of calendar year	Number of canisters produced during year	Cumulative number of canisters produced
1990	0	0
1991	0	0
1992	0	0
1993	0	0
1994	0	0
1995	0	0
1996	145	145
1997	145	290
1998	145	435
1999	73	508
2000	145	653
2001	145	798
2002	72	870
2003	145	1015
2004	145	1160
2005	145	1305
2006	73	1378
2007	145	1523
2008	145	1668
2009	72	1740
2010	120	1860
2011	0	1860
2012	0	1860
2013	0	1860
2014	0	1860
2015	0	1860
2016	0	1860
2017	0	1860
2018	0	1860
2019	0	1860
2020	0	1860

[a]Source: Coony 1987. It was assumed there that no fuel reprocessing takes place after CY 2001. If reprocessing does extend beyond 2001, each additional year of reprocessing would produce about 50 additional canisters.

Table 3.4.3. Hanford Operations. Radioisotope content per HLW canister (NCAW glass)[a]

	Isotope	Curies/canister	Grams/canister
1	C-14	0.9590E-01	0.2151E-01
2	Fe-55	0.1260E+02	0.5039E-02
3	Ni-59	0.1030E-01	0.1360E+00
4	Ni-63	0.2380E+01	0.3858E-01
5	Co-60	0.3580E+02	0.3166E-01
6	Zr-93	0.3430E-01	0.1365E+02
7	Nb-93m	0.7460E-02	0.2639E-04
8	In-113m	0.2870E-01	0.1716E-08
9	Sn-113	0.2870E-01	0.2858E-05
10	Sn-119m	0.2460E+02	0.5492E-02
11	Sn-121m	0.1310E+00	0.2215E-02
12	Sb-125	0.1100E+03	0.1065E+00
13	Te-125m	0.2690E+02	0.1493E-02
14	U-234	0.1750E-04	0.2800E-02
15	U-235	0.3250E-03	0.1504E+03
16	U-236	0.8190E-03	0.1265E+02
17	U-238	0.5880E-02	0.1749E+05
18	Np-237	0.3120E+00	0.4425E+03
19	Pu-238	0.4110E+00	0.2400E-01
20	Pu-239	0.3600E+01	0.5789E+02
21	Pu-240	0.1180E+01	0.5177E+01
22	Pu-241	0.3740E+02	0.3630E+00
23	Pu-242	0.6500E-04	0.1702E-01
24	Am-241	0.1030E+04	0.3000E+03
25	Am-242m	0.5960E+00	0.6131E-01
26	Am-243	0.4400E+00	0.2227E+01
27	Cm-242	0.1230E+02	0.3719E-02
28	Cm-244	0.9700E+00	0.1198E-01
29	C-14	0.9400E-04	0.2109E-04
30	Se-79	0.5260E+00	0.7550E+01
31	Sr-89	0.1150E-01	0.3961E-06
32	Sr-90	0.7310E+05	0.5357E+03
33	Y-90	0.7310E+05	0.1343E+00
34	Y-91	0.1280E+00	0.5218E-05
35	Zr-93	0.2440E+01	0.9709E+03
36	Zr-95	0.4910E+00	0.2285E-04
37	Nb-93m	0.1220E+01	0.4315E-02
38	Nb-95	0.1080E+01	0.2762E-04
39	Tc-99	0.1760E+02	0.1038E+04
40	Ru-103	0.1040E-03	0.3221E-08
41	Ru-106	0.5960E+04	0.1781E+01
42	Rh-103m	0.1040E-03	0.3196E-11
43	Rh-106	0.5960E+04	0.1674E-05
44	Pd-107	0.6960E-01	0.1353E+03
45	Ag-110	0.1120E-03	0.2686E-13

Table 3.4.3 (continued)

	Isotope	Curies/canister	Grams/canister
46	Ag-110m	0.8430E-02	0.1774E-05
47	Cd-113m	0.2240E+02	0.1033E+00
48	Cd-115m	0.1570E-05	0.6164E-10
49	Sn-119m	0.1980E+01	0.4421E-03
50	Sn-121m	0.1460E+00	0.2469E-02
51	Sn-123	0.6100E+00	0.7422E-04
52	Sn-126	0.8290E+00	0.2921E+02
53	Sb-124	0.6330E-05	0.3618E-09
54	Sb-125	0.2350E+04	0.2275E+01
55	Sb-126	0.1160E+00	0.1387E-05
56	Sb-126m	0.8290E+00	0.1055E-07
57	Te-123m	0.8890E-31	0.1002E-34
58	Te-125m	0.5750E+03	0.3192E-01
59	Te-127	0.5760E+00	0.2183E-06
60	Te-127m	0.5850E+00	0.6200E-04
61	Te-129	0.6170E-07	0.2945E-14
62	Te-127m	0.9810E-07	0.3256E-11
63	I-129	0.2140E-02	0.1212E+02
64	Cs-134	0.1360E+04	0.1051E+01
65	Cs-135	0.4650E+00	0.4038E+03
66	Cs-137	0.8730E+05	0.1003E+04
67	Ba-137m	0.8250E+05	0.1534E-03
68	Ce-141	0.1880E-05	0.6598E-10
69	Ce-144	0.1370E+05	0.4294E+01
70	Pr-143	0.6440E-21	0.9565E-26
71	Pr-144	0.1370E+05	0.1813E-03
72	Pr-144m	0.1960E+03	0.1080E-05
73	Pm-147	0.5270E+05	0.5685E+02
74	Pm-148	0.1100E-06	0.6692E-12
75	Pm-148m	0.2280E-05	0.1067E-09
76	Sm-151	0.1550E+04	0.5891E+02
77	Eu-152	0.3560E+01	0.2058E-01
78	Eu-154	0.4190E+03	0.1552E+01
79	Eu-155	0.6610E+03	0.1421E+01
80	Gd-153	0.4400E-02	0.1247E-05
81	Tb-160	0.2010E-04	0.1780E-08
Total		0.4165E+06	0.2273E+05

[a]Source: Coony 1987. Based on 1650 kg of HLW glass per canister. This is the upper bound case for HANF NCAW glass.

Table 3.4.4. Hanford Operations. Calculated radioactivity and thermal power per HLW canister.[a]

Decay time, years[b]	Radioactivity per canister (Ci)	Thermal power per canister (W)
0	416,500	1,159
1	373,600	1,034
2	346,400	964
5	300,300	861
10	257,000	759
15	226,200	676
20	200,600	604
30	158,900	484
50	100,000	314
100	32,000	117
200	4,040	33
300	1,110	22
350	813	20
500	530	16
1,000	240	7
1,050	224	6.6
2,000	73	1.6
5,000	31	0.18
10,000	29	0.15
20,000	27	0.12
50,000	24	0.08
100,000	22	0.08
500,000	13	0.11
1,000,000	9	0.11

[a]Calculations made by ORIGEN2 code based on data supplied by HANF (White, 1986). Canister is filled to 85% of capacity and contains 1650 kg of HLW glass made from neutralized current acid waste (NCAW). Data shown represent the "upper bound" case, i.e., the maximum expected activity.

[b]Years after vitrification.

Table 3.4.5. Hanford Operations. Estimated annual average radioactivity per canister of HLW glass[a]

End of calendar year	Annual number of canisters	Annual radioactivity curies	Annual average curies per canister
1995	0	0	0
1996	145	4.7E+07	324,100
1997	145	3.8E+07	262,000
1998	145	3.4E+07	234,500
1999	73	1.6E+07	219,200
2000	145	3.0E+07	206,900
2001	145	2.9E+07	200,000
2002	72	1.4E+07	194,400
2003	145	1.1E+07	75,900
2004	145	3.3E+04	228
2005	145	3.2E+04	221
2006	73	1.6E+04	219
2007	145	2.9E+04	200
2008	145	8.8E+03	61
2009	72	4.4E+03	61
2010	120	7.3E+03	61
2011	0	0	0
2012	0	0	0
2013	0	0	0
2014	0	0	0
2015	0	0	0
2016	0	0	0
2017	0	0	0
2018	0	0	0
2019	0	0	0
2020	0	0	0

[a]Calculated from Table 4 of Coony 1987. It was assumed there that no fuel reprocessing takes place after year 2001, and that if reprocessing continues after year 2001, each additional year of fuel reprocessing generates an equivalent of 50 canisters per year and an equivalent borosilicate glass activity of 7.3E+6 curies per year after CY 2010. It was also assumed that the neutralized current acid waste would be canistered first (930 canisters), then the complexant concentrate (580 canisters), and finally the plutonium finishing plant waste (350 canisters). Note that this table does not show the maximum radioactivity per canister, only the average.

Table 3.4.6. Hanford Operations. Estimated cumulative radioactivity and thermal power per canister of HLW glass[a]

End of calendar year	Cumulative number of canisters produced	Cumulative radioactivity and thermal power of HLW glass			
		Cumulative radioactivity		Cumulative thermal power[b]	
		Total (10^6 Ci)	per canister (Ci)	Total (10^3 W)	per canister (W)
1995	0	0	0		
1996	145	47	324,000	130	900
1997	290	76	262,000	212	730
1998	435	100	230,000	278	640
1999	508	110	217,000	305	600
2000	653	130	200,000	359	550
2001	798	160	200,000	439	550
2002	870	170	195,000	470	540
2003	1,015	170	167,000	470	460
2004	1,160	170	146,000	470	405
2005	1,305	160	123,000	450	345
2006	1,378	160	116,000	450	327
2007	1,523	150	98,000	420	276
2008	1,668	150	90,000	420	252
2009	1,740	140	80,000	390	224
2010	1,860	140	75,000	390	210
2011	1,860	140	75,000	390	210
2012	1,860	130	70,000	360	194
2013	1,860	130	70,000	360	194
2014	1,860	130	70,000	360	194
2015	1,860	130	70,000	360	194
2016	1,860	120	65,000	330	177
2017	1,860	120	65,000	330	177
2018	1,860	120	65,000	330	177
2019	1,860	110	60,000	310	167
2020	1,860	110	60,000	310	167

[a]Calculated from Table 4 of Coony 1987. It was assumed there that no fuel reprocessing takes place after year 2001, and that if reprocessing continues after year 2001, each additional year of fuel reprocessing generates an equivalent of 50 canisters per year and an equivalent borosilicate glass activity of 7.3E+6 curies per year after CY 2010. It was also assumed that the neutralized current acid waste would be canistered first (930 canisters), then the complexant concentrate (580 canisters), and finally the plutonium finishing plant waste (350 canisters). Note that this table does not show the maximum radioactivity per canister, only the average.

[b]Thermal power was estimated by ratio from radioactivity.

Table 3.4.7. Hanford Operations. Chemical compositions of HWVP reference HLW (NCAW), substituted NCAW, frit, and borosilicate glass[a]

Component	Reference NCAW waste composition wt %	Substituted NCAW waste composition[b] wt %	Frit composition wt %	Glass composition wt %
SiO_2	2.9	3.0	67.25	51.3
B_2O_3	0.0	0.0	12.75	9.6
Na_2O	10.5	10.7	10.25	10.4
Li_2O	0.0	0.0	5.0	3.8
CaO	0.3	0.3	3.75	2.9
MgO	0.2	0.3	1.0	0.8
Fe_2O_3	44.0	44.4	--	11.1
Al_2O_3	17.0	17.2	--	4.3
Cr_2O_3	5.3	5.3	--	1.3
ZrO_2	2.3	2.4	--	0.6
NiO	2.3	2.4	--	0.6
La_2O_3	2.2	2.2	--	0.5
SO_4	1.8	1.8	--	0.4
Nd_2O_3	1.7	2.1	--	0.5
MoO_3	1.2	1.2	--	0.3
F	1.2	1.2	--	0.3
CuO	0.6	0.6	--	0.1
TOC	0.6	0.6	--	--
MnO_2	0.6	0.7	--	0.2
CeO_2	0.6	0.7	--	0.2
RuO_2	0.6	0.6	--	0.1
U_3O_8	0.6	Sub Nd	--	--
Cs_2O	0.6	1.0	--	0.2
BaO	0.4	0.4	--	0.1
SrO	0.4	0.4	--	0.1
Pr_6O_{11}	0.4	0.4	--	0.1
Tc_2O_7	0.4	Sub Mn	--	--
Rb_2O	0.2	Sub Cs	--	--
Y_2O_3	0.2	0.2	--	0.04
Sm_2O_3	0.2	0.2	--	0.04
PdO	0.2	Del	--	--
Rh_2O_3	0.2	Del	--	--
NpO_2	0.1	Sub Ce	--	--
TeO_2	0.1	Del	--	--
Pm_2O_3	0.1	Sub Nd	--	--
BeO	0.1	Sub Mg	--	--
SeO_2	0.03	Del	--	--
SnO_2	0.02	Del	--	--
CdO	0.02	Del	--	--
Eu_2O_3	0.02	Sub Nd	--	--
PuO_2	0.02	Sub Ce	--	--
Am_2O_3	0.02	Sub Nd	--	--
P_2O_5	0.02	Del	--	--
Ag_2O	0.01	Del	--	--
Nb_2O_5	0.01	Sub Mo	--	--
Gd_2O_3	0.01	0.01	--	0.003
Ta_2O	0.01	Del	--	--
TiO_2	0.01	Del	--	--
Total	100	100	100	100

[a]Source: Mitchell 1986. Reference glass is HW-39. Data given are for a waste oxide loading of 25 wt% and are based on approximately 4-year old waste.

[b]Components marked sub were substituted as indicated. Components marked Del were deleted.

3.5 IDAHO NATIONAL ENGINEERING LABORATORY (INEL)

3.5.1 Introduction

The Idaho Chemical Processing Plant (ICPP), which is located at INEL, has as its primary purpose the reprocessing of DOE fuels for the recovery of uranium and other elements. Fuels routinely processed include aluminum-, stainless steel-, and zirconium-based fuels, the latter comprising the majority of fuel. The acidic high-level liquid waste resulting from dissolution and organic solvent extraction of these fuels is temporarily stored in stainless steel tanks and is subsequently solidified by a fluidized-bed calcining process. The granular oxide calcine resulting from this process is stored retrievably on-site in stainless steel bins located in below-ground concrete vaults. Thus far, about 5.6 million gallons of liquid HLW have been solidified by calcining, resulting in an average volume reduction of about 7:1.

3.5.2 Types of HLW Produced

Various alternatives for the immobilization of HLW are being studied at INEL; both glass and hot-isostatic-pressed glass-ceramic (also referred to as "ceramic-based") compositions are being considered for possible use as final waste forms. A final decision on the waste form has not yet been made. Volumetric considerations favor the glass-ceramic form, which has only about 40% of the volume of the glass form (Staples, Knecht, and Berreth 1986). The terminology "glass-ceramic" is used here rather than "ceramic" because the solid is a mixture of an amorphous glass phase and a crystalline ceramic phase.

3.5.3 Physical Description

Regardless of whether glass-ceramic or vitrified HLW is produced, it appears likely that the waste will be contained in canisters similar in dimensions to those planned for use at WVDP, SRP, and HANF; that is,

61 cm diameter by 300 cm high. If glass-ceramic blocks are to be placed in such canisters, the canister would be designed with a wide-mouth opening and several blocks could be placed in each canister. Table 3.5.1 gives estimated physical characeristics of the canister and its contents based on the assumption that the glass-ceramic form of HLW is used. This physical description should be considered preliminary at this time.

3.5.4 Inventory and Production Schedule

Under the Defense Waste Management Plan, construction of a HLW immobilization facility will be started at INEL in 2002; operation of the facility is scheduled to begin by 2011 (Berreth 1987). If the glass-ceramic form is chosen, the maximum rate of immobilized HLW production would be approximately 1000 canisters per year, as shown in Table 3.5.2. This is based on an estimated 650 canisters per year required to handle the waste from anticipated annual fuel reprocessing operations, plus an additional 350 canisters per year to work off the backlog of stored calcine from past operations (Knecht 1986a). Other assumptions are that (1) there is no pretreatment of the calcine to remove inerts prior to immobilization, (2) the usable waste volume per canister is 0.57 m^3, (3) the waste loading (calcine in glass-ceramic) is 70 wt%, (4) the density of the glass-ceramic is 3200 kg/m^3, and (5) during the first three years of operation, the immobilization plant runs at a reduced rate (500 to 700 canisters/year) sufficient to keep up with current production (Berreth 1987). Final decisions on processing options for INEL have not yet been made, so the schedule and canistered waste characteristics presented here should be considered as preliminary (Berreth and Knecht 1986, Berreth 1987a).

3.5.5 Radionuclide Content of Canister

Table 3.5.3 shows the estimated radionuclide composition of a canister based on the assumptions that the calcined HLW is converted to ceramic form and that each canister contains 1825 kg of ceramic, which

is the equivalent of 1277 kg of calcine (Berreth 1986c). The radionuclide composition of the calcine for these calculations represents 3-year-old calcine and was taken from an INEL report (IDO-10105, 1982). In practice, the feed to the immobilization plant could include calcine with an age greater than three years, and the activity per canister would accordingly be lower. The composition given is intended to represent the maximum activity per canister. Because of security restrictions, no radionuclide composition data have been officially releaaed by INEL; therefore, the estimates presented in Table 3.5.3 should be considered preliminary.

3.5.6 Radioactivity and Thermal Power

Table 3.5.4 shows the calculated radioactivity and thermal power per canister as functions of decay time ranging from 0 to 10^6 years. These calculations were made by the ORIGEN2 program using the radionuclide composition shown in Table 3.5.3 and hence carry the same caveats as those mentioned for Table 3.5.3; however, they are intended to represent the maximum radioactivity per canister that could be encountered.

Appendix 3A presents detailed decay tables showing the contributions of individual radionuclides to total curies and watts per canister for decay times ranging from 0 to 10^6 years; these are for the maximum activity canister only.

Table 3.5.5 shows estimated year-by-year projections of cumulative average radioactivity and thermal power per canister. These were calculated from projected estimates of total curies and watts for calcined waste from INEL's FY 1987 Integrated Data Base submittal (Berreth 1987). The cumulative averages shown in the table were calculated from the IDB submittal based on the assumption that two-thirds of the canisters produced in a given year would be made from "fresh" calcine (actually aged 3 or more years), and the other one-third would be made from old calcine. Obviously, this may not correspond to the actual scheduling of

feeds to the immobilization plant; however, the average values shown should be more useful than maximum values for estimation of total repository radioactivity and thermal loads.

3.5.7 Chemical Composition

Table 3.5.6 shows the compositions of typical calcines produced at INEL by the calcination of high-level liquid wastes. These calcines can be densified and immobilized by hot isostatic pressing with added components that convert sodium and boron in the waste to an interstitial glass phase and stabilize the ceramic-based product. The chemical composition of the final ceramic-based product has not been completely decided and will depend on the type of calcine fed to the plant. Table 3.5.7 gives approximate chemical compositions of five ceramic-based products that have been produced during process development studies. These studies are continuing, and it should not be assumed that the compositions of the actual immobilized high-level wastes produced at INEL are typified by the developmental results shown here (Baker 1986; Staples, Knecht, and Berreth 1986).

3.5.8 Assessment of Data

Because the strategy and processing for disposal of INEL high-level waste will not be decided prior to the 1990s, estimates of canister production and radioactivity given here are preliminary. These estimates also are based on incomplete information on immobilized waste radionuclide compositions. The data contained in the most recent Integrated Data Base submittal (Berreth 1987) give projections of total curies and watts for liquid waste and calcined waste inventories from 1987 to 2020. However, these data cannot be used to estimate the maximum radioactivity per canister, since the cumulative average radioactivity gives no indication of the maximum radioactivity in a given year of production. Because of security limitations, no data were furnished by INEL on the radionuclide compositions of the interim waste forms, nor on the compositions of glass or ceramic immobilized wastes made from the interim

wastes. Our estimates were based on an assumed radionuclide composition of 3-yr aged calcine from a 1982 report. Repository calculations will require information on the maximum expected radioactivity and thermal power per canister and on the decay of these quantities as a function of time.

3.5.9 References for Section 3.5

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Berreth 1987a. Letter from J. R. Berreth, INEL, to Royes Salmon, ORNL, August 6, 1987.

Berreth and Knecht 1986. J. R. Berreth and D. A. Knecht, Potential Process Options to Minimize Immobilized HLW Volume at the Idaho Chemical Processing Plant, WINCO-M-10079, presented at the ANS International Symposium on Waste Management, Niagara Falls, September 14-18, 1986.

Berreth 1986a. Letter from J. R. Berreth, INEL, to J. E. Solecki, DOE, BERR-21-86, dated March 14, 1986.

Berreth 1986b. Telephone conversation, J. R. Berreth (INEL) and R. Salmon (ORNL), June 17, 1986.

Berreth 1986c. Letter from J. R. Berreth, INEL, to K. J. Notz, ORNL, December 1, 1986.

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Staples, Knecht, and Berreth 1986. B. A. Staples, D. A. Knecht, and J. R. Berreth, Technology for the Long-Term Management of Defense HLW at the ICPP, Westinghouse Idaho Report WINCO-1038, June 1986.

Table 3.5.1. Idaho National Engineering Laboratory. High level waste form and canister characteristics.[a]

Waste form	glass-ceramic blocks in closed canister
Canister material	stainless steel type 304L
Glass-ceramic density, g/cm^3	3.2
Weights per canister:	
Empty canister, kg	500
Glass-ceramic, kg	1825
Total loaded weight, kg	2325
Waste loading in glass-ceramic, wt%	70[b]
Glass-ceramic volume per canister, m^3	0.57[b]
Canister dimensions:	
Outside diameter, cm.	61
Height overall, cm.	300
Wall thickness, cm.	0.95
Radionuclide content, curies/canister	108,900[c]
Heat generation rate, watts/canister	339[c]

[a]Based on the following assumptions:

1. Glass-ceramic form is chosen for HLW immobilization. The term "glass-ceramic" denotes an immobilized waste form consisting of a glass phase dispersed in a ceramic phase.
2. Canister load is equivalent to 1277 kg calcine.
3. Calcine is 3 years old at time of immobilization.
4. Canister is similar in dimensions to DWPF canister.
5. Radionuclide content of calcine is as shown in IDO-10105 (see Table 3.5.3).

[b]Reference: Berreth 1987.

[c]At time of immobilization. Quantities shown are estimated maximum values; average values are expected to be considerably less.

Table 3.5.2. Idaho National Engineering Laboratory. Estimated production schedule of canisters of HLW glass-ceramic.[a]

Calendar year	Number of canisters produced during year	Cumulative number of canisters produced
2010	0	0
2011	500	500
2012	600	1,100
2013	700	1,800
2014	1,000	2,800
2015	1,000	3,800
2016	1,000	4,800
2017	1,000	5,800
2018	1,000	6,800
2019	1,000	7,800
2020	1,000	8,800

[a]This assumes that a glass-ceramic form (density 3.2 g/cm^3) is selected for HLW disposal and that each canister contains 1277 kg of calcine (1825 kg of glass-ceramic). Waste loading is 70 wt%. Canister production will continue after 2020 but is not shown. Source: Berreth 1987.

Table 3.5.3. Idaho National Engineering Laboratory. Radioisotope content per HLW Canister.[a]

	Isotope	Curies/canister	Grams/canister
1	Se-79	0.8173E-01	0.1173E+01
2	Rb-87	0.4597E-05	0.5252E+02
3	Sr-90	0.1660E+05	0.1217E+03
4	Y-90	0.1660E+05	0.3051E-01
5	Zr-93	0.3959E+00	0.1575E+03
6	Nb-93M	0.9577E-01	0.3387E-03
7	Tc-99	0.2682E+01	0.1582E+03
8	Ru-106	0.1239E+04	0.3701E+00
9	Rh-106	0.1239E+04	0.3479E-06
10	Pd-107	0.2554E-02	0.4965E+01
11	Sn-126	0.4086E-01	0.1440E+01
12	Sb-126M	0.4086E-01	0.5201E-09
13	Sb-126	0.4086E-01	0.4887E-06
14	Cs-134	0.4214E+04	0.3256E+01
15	Cs-135	0.9577E-01	0.8316E+02
16	Cs-137	0.1660E+05	0.1908E+03
17	Ba-137M	0.1532E+05	0.2848E-04
18	Ce-144	0.1047E+05	0.3282E+01
19	Pr-144	0.1047E+05	0.1386E-03
20	Pm-147	0.1532E+05	0.1653E+02
21	Sm-151	0.2171E+03	0.8250E+01
22	Eu-154	0.2299E+03	0.8513E+00
23	U-233	0.1532E-08	0.1583E-06
24	U-234	0.5491E-06	0.8785E-04
25	U-235	0.2299E-05	0.1063E+01
26	U-236	0.1277E-04	0.1973E+00
27	U-237	0.6130E-08	0.7507E-13
28	U-238	0.1277E-10	0.3797E-04
29	Np-237	0.6130E-04	0.8693E-01
30	Pu-238	0.8939E+02	0.5221E+01
31	Pu-239	0.8939E+00	0.1437E+02
32	Pu-240	0.8300E+00	0.3642E+01
33	Pu-241	0.2043E+03	0.1983E+01
34	Pu-242	0.2299E-02	0.6018E+00
35	Am-241	0.1162E+01	0.3385E+00
36	Am-243	0.1060E-01	0.5315E-01
37	Cm-242	0.8300E+00	0.2510E-03
38	Cm-244	0.6640E+00	0.8201E-02
	Total	0.1088E+06	0.8315E+03

[a]Quantities are at time of filling canister and are based on 3-yr old calcine immobilized in glass-ceramic with a load of 1277 kg of calcine per canister (1825 kg of glass-ceramic per canister). Based on IDO-10105 (1982) and Berreth 1986c.

Table 3.5.4. Idaho National Engineering Laboratory. Calculated radioactivity and thermal power per HLW canister.[a]

Decay time after immobilization, years	Total radioactivity per canister (Ci)	Total thermal power per canister (W)
0	108,900	339
1	89,400	267
2	78,600	230
5	64,100	185
10	53,600	157
15	46,900	138
20	41,500	123
30	32,800	97
50	20,500	61
100	6,430	20
200	680	2.6
300	98	0.67
350	48	0.45
500	16	0.24
1,000	7.2	0.11
1,050	7.0	0.10
2,000	5.6	0.06
5,000	5.0	0.04
10,000	4.6	0.033
20,000	4.2	0.023
50,000	3.6	0.012
100,000	3.1	0.008
500,000	1.4	0.003
1,000,000	0.71	0.001

[a]Results of ORIGEN2 calculations based on glass-ceramic form, assuming 1277 kg of calcine per canister (1825 kg of glass-ceramic per canister), with the initial radionuclide composition shown in Table 3.5.3.

Table 3.5.5. Idaho National Engineering Laboratory. Estimated cumulative average radioactivity and thermal power per canister of HLW glass-ceramic[a]

End of calendar year	Cumulative number of canisters produced	Cumulative radioactivity and thermal power of HLW glass-ceramic			
		Cumulative radioactivity		Cumulative thermal power	
		Total (10^6 Ci)	Per canister (Ci)	Total (kW)	Per canister (W)
2010	0	0	0	0	0
2011	500	20	40,300	56	112
2012	1100	44	40,000	124	112
2013	1800	70	38,900	202	112
2014	2800	107	38,200	313	112
2015	3800	143	37,600	421	111
2016	4800	177	36,900	526	110
2017	5800	210	36,200	624	108
2018	6800	242	35,600	726	107
2019	7800	272	34,900	819	105
2020	8800	301	34,200	908	103

[a]Calculated from estimates given in Berreth 1987, using the assumptions that two-thirds of the glass-ceramic produced in a given year is made from fresh calcine, while the other one-third is made from old calcine, and that each canister contains 1277 kg of calcine, which is equivalent to 0.91 m^3 of calcine in bulk form. The term "glass-ceramic" denotes a ceramic-based immobilized waste. Cumulative radioactivity per canister means cumulative immobilized radioactivity divided by cumulative number of canisters produced.

Table 3.5.6. Composition of typical HLW calcines produced at INEL[a]

Component	Type of calcine and composition (wt %)			
	Alumina	Zirconia	Fluorinel	Zirconia-sodium
Al_2O_3	82-95	13-17	6	12-14
Na_2O	1-3	---	---	0-5
ZrO_2	---	21-27	23	20-26
CaF_2	---	50-56	56	48-53
Ca	---	2-4	4	2-4
NO_3	5-9	0.5-2	0.5-2	0.5-4
B_2O_3	0.5-2	3-4	4	3-4
CdO	---	---	6	---
Fission products and actinides	≤ 1	≤ 1	≤ 1	≤ 1

[a]Source: Staples, Knecht, and Berreth, 1986.

Table 3.5.7. Compositions of typical ceramic-based waste forms developed for immobilization of INEL calcined HLW

Formulation number	SiO_2 (wt %)	Na_2O (wt %)	Li_2O (wt %)	B_2O_3 (wt %)	Waste (wt %)
12	8.6	1.1	0.5	2.6	87.2
11	16.0	0.0	0.0	1.4	82.6
17	30.3	0.0	0.0	2.3	67.5
6	28.6	2.1	0.9	3.5	64.9
1	14.2	2.6	1.2	1.7	80.3

[a]Source: Baker 1986.