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

Appendix G: Evaluation of Potential for Greater-Than-Class C Classification of Irradiated Hardware Generated by Utility-Operated Reactors

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

August 1991

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GREATER-THAN-CLASS C LOW-LEVEL WASTE CHARACTERIZATION

Appendix G: Evaluation of Potential for Greater-Than-Class C Classification of Irradiated Hardware Generated by Utility-Operated Reactors

James E. Cline

August 1991

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

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ABSTRACT

This study compiles and evaluates data from many sources to expand a base of data from which to estimate the activity concentrations and volumes of greater-than-Class C low-level waste that the Department of energy will receive from the commercial power industry. Sources of these data include measurements of irradiated hardware made by or for the utilities that was classified for disposal in commercial burial sites, measurements of neutron flux in the appropriate regions of the reactor pressure vessel, analyses of elemental constituents of the particular structural material used for the components, and the activation analysis calculations done for hardware. Evaluations include results and assumptions in the activation analyses. Sections of this report and the appendices present interpretation of data and the classification definitions and requirements.

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1. SUMMARY AND CONCLUSIONS

The U.S. Department of Energy (DOE) has responsibility for receiving and disposing of all low-level radioactive waste classified as greater-than-Class C (GTCC). A large source of waste, potentially GTCC, is irradiated metal components from nuclear reactors, of which the commercial nuclear power industry is the largest source. Nonfuel-bearing hardware internal to the pressure vessel in nuclear reactors absorbs neutrons during reactor operation and becomes highly radioactive. The bulk of the activity is ^{60}Co , and other radionuclides more important to waste classification usually have concentrations several orders of magnitude below its level. If the neutron flux seen by the components is sufficiently high, these other radioactive constituents could reach concentrations high enough to classify the material as GTCC LLW, requiring its disposal by DOE. Irradiated hardware that is potential GTCC LLW falls into two categories: that discharged in normal refueling and maintenance operations, and that discharged in decommissioning the reactor at the end of its operating lifetime. The present study evaluates activity levels and volumes of nonfuel-bearing components from commercial power generation that could classify as GTCC material.

The nuclides ^{59}Ni , ^{63}Ni , ^{94}Nb , and, to a lesser extent, transuranics (TRU), generally control the waste classification of activated hardware. Nickel and TRU nuclides emit only weak x-ray, beta, or alpha activities that are nearly impossible to measure in the presence of the dominant cobalt radiation, and the niobium isotope, although emitting energetic gamma rays, is present at such low levels that it, too, is overwhelmed by the cobalt activity. Thus, practical waste classification techniques for irradiated hardware (as for most other utility-generated waste) relate all other activities to the observable ^{60}Co concentration through scaling factors. Since cobalt is present in most reactor structural materials as a relatively uncontrolled trace material, as is niobium in all but inconel, current classification schemes give rise large uncertainties in estimating concentrations for the desired activities. A lack of data from measurements of radioactivity in many irradiated components adds to uncertainty in the estimates, particularly for those items discharged at decommissioning; many of the estimates rely solely on activation calculations that use estimates of the elemental compositions of the material, the average neutron flux seen by that component, and, in some cases, the neutron activation cross section for the reaction. Uncertainties in these values exacerbate total uncertainty in estimated activities of the waste and, therefore, estimated volumes of GTCC material.

The present study has compiled and evaluated data from many additional sources to expand a base of data from which to estimate the activity concentrations and volumes of GTCC LLW that DOE will receive from the commercial power industry. Sources of these data include measurements of irradiated hardware made by or for the utilities that was classified for disposal in commercial burial sites, measurements of neutron flux in the appropriate regions of the reactor pressure vessel, analyses of elemental constituents of the particular structural material used for the components, and the activation analysis calculations done for hardware. Evaluations include results and assumptions in the activation analyses. Sections of this report and the appendices present interpretation of data and the

classification definitions and requirements. This section of the report summarizes and interprets the results.

1.1 Radionuclide Concentrations

Table G-1 summarizes the evaluation of radioactive concentrations of five radionuclides important for waste classification of reactor-activated hardware removed in normal plant operation: ^{59}Ni , ^{63}Ni , ^{94}Nb , ^{99}Tc , and TRU (transuranic activity, TRU, is a component of the surface contamination acquired from contact with the reactor coolant). Section 3.1 presents and discusses data from which the summary is made. The table lists the seven boiling water reactor (BWR) and one pressurized water reactor (PWR) components identified as having the greatest potential for classification as GTCC LLW: the control rod blade (CRB), three incore instruments—local power range monitors (LPRM), intermediate range monitors (IRM), and source range monitors (SRM) poison curtains, fuel support piece, fuel flow channel from boiling water reactors, and the PWR fuel assembly skeleton. Section 2.3 describes these components.

Table G-1 expresses concentrations as ratios to the Class C activity concentration upper limits for that nuclide and highlights values that exceed 1.0 (GTCC). Section 2.2 describes the waste classification rules and practices. Data in the table are for typical discharge frequencies. Values for the ranges result from statistical analyses of measurement results. The table demonstrates the dominance of ^{63}Ni , a 10 CFR 61 Table 2 nuclide, in classifying waste. The other four activities are 10 CFR 61¹ Table 1 nuclides and contribute primarily through the "sum of the fractions" rule. The data in the table indicate that only the BWR incore instrumentation and PWR inconel-718 portions of the fuel assembly skeleton may have concentrations that exceed the Class C limits. Only skeletons manufactured by Westinghouse have significant portions made of inconel. Table G-1 has two entries for the incore instruments LPRM, IRM and SRM. These components are about 43 ft long, only 12 ft of which are in the reactor core (hot end). The portion labeled "hot end" represents this 12-ft section, and the other entry averages the concentration over the entire 43-ft long piece. Although the incore instrument tubes are GTCC LLW, they are low volume items, and the utilities usually dispose of them as Class C waste under the "factor of ten rule" by mixing them with the larger volume CRBs and averaging over the volume of SS-304 in the casks.

Table G-2 lists reactor internal components with greatest potential for GTCC classification and summarizes the nuclide concentrations expected at the end of the reactor lifetime. Section 2.2 discusses these components. The table here also expresses concentrations as ratios to the Class C upper concentration limits for that nuclide and highlights values that exceed 1.0. All values assume a 40-year reactor lifetime and a 75% duty cycle of full power operation. The table summarizes data presented and discussed in Section 3.2. The spread between the high and low values is very large and represents the uncertainties from a lack of data. The high values are principally from computational estimates of neutron fluxes and cross sections, and the low estimates rely on measured scaling factors and a measurement set of neutron flux in a region outside the core. The base values are a log-mean average of the two and represent the best estimate that can be made at this time. Base values indicate that none of the hardware may reach GTCC classification status, even with plant life extensions of 20 years.

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

	COMPONENT		RATIO OF ACTIVITIES TO CLASS "C" LIMITS RADIONUCLIDE				
			Ni-59	Ni-63	Nb-94	Tc-99	TRU
BWR	CRB	Min:	0.02	0.10	0.03	0.0003	0.0032
		Max:	0.29	0.46	0.14	0.0200	0.0200
		Avg:	0.09	0.24	0.07	0.0100	0.0100
	LPRM Hot End	Min:	0.11	0.54	0.16	0.0018	0.0180
		Max:	6.16	9.86	2.80	0.0300	0.3200
		Avg:	1.59	4.29	1.20	0.0100	0.1400
	with Cold End	Min:	0.03	0.15	0.05	0.0005	0.0050
		Max:	1.72	2.75	0.78	0.0084	0.0023
		Avg:	0.44	1.20	0.33	0.0028	0.0008
	IRM Hot End	Min:	0.85	4.00	1.15	0.0100	0.1300
		Max:	6.43	10.29	2.95	0.0300	0.3400
		Avg:	2.22	6.00	1.70	0.0200	0.2000
	with Cold End	Min:	0.24	1.12	0.32	0.0028	0.0360
		Max:	1.79	2.87	0.82	0.0080	0.0950
		Avg:	0.62	1.67	0.47	0.0056	0.0560
	SRM Hot End	Min:	1.13	5.29	1.55	0.0200	0.1700
		Max:	3.57	5.71	1.65	0.0200	0.1900
		Avg:	2.06	5.57	1.60	0.0200	0.1800
	with Cold End	Min:	0.32	1.50	0.43	0.0050	0.0470
		Max:	1.00	1.60	0.60	0.0060	0.0530
		Avg:	0.57	1.55	0.50	0.0056	0.0500
	POISON CURTAINS	Min:	0.08	0.36	0.10	0.0012	0.0120
		Max:	0.42	0.67	0.19	0.0023	0.0220
		Avg:	0.19	0.51	0.15	0.0016	0.0170
	FUEL SUPPORT PIECE	Avg:	0.004	0.020	0.005	0.0001	0.0050
	FUEL FLOW CHANNEL	Min:	1.1E-04	1.7E-04	0.002		0.0870
		Max:	1.1E-03	2.9E-03	0.040		0.1400
		Avg:	3.6E-04	9.7E-04	0.010		0.0490
PWR	FUEL						
	ASSEMBLY						
	SS-304		0.024	0.066	0.14		
	ZIRCALOY-4		0.013	0.035	0.34		
	INCONEL-718		0.93	2.50	12.30		

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

COMPONENT	RANGE	RADIONUCLIDE CONCENTRATION RELATIVE TO CLASS "C" LIMITS			
		Ni-59	Ni-63	Nb-94	Tc-99
<u>PWR</u>					
SHROUD	HIGH: (1)	8.5	13.3	10.3	0.22
	BASE: (2)	0.62	1.7	0.79	9.3E-03
	LOW: (3)	0.045	0.21	0.06	3.9E-04
CORE BARREL	HIGH: (1)	1.1	1.8	0.77	0.01
	BASE: (2)	0.085	0.23	0.06	1.3E-03
	LOW: (3)	6.4E-03	0.03	4.6E-03	1.8E-04
<u>BWR</u>					
CORE SHROUD	HIGH: (1)	2.7	4.3	1.7	2.7E-03
	BASE: (2)	0.13	0.55	0.13	3.6E-04
	LOW: (3)	5.9E-03	0.07	0.01	4.8E-05

1.2 Elemental Constituents in Reactor Hardware Alloys

Table G-3 summarizes the data and analyses presented in Appendix B for elemental constituents of alloys used for most reactor hardware. The relative abundances of these elements, many of which are only trace constituents, are very important for classifying the irradiated pieces. The table includes the nominal value (i.e., the manufacturer's specification, measured values previously reported for SS-304,² and the average values from two additional sources), material chemically analyzed for its components, and constituents deduced from measured activated radioactivity. The latter analyses include the activation cross sections and are a more pragmatic indicator of the expected radioactivities from activation. Where applicable, the values include root-mean-square (RMS) uncertainties resulting from the data analyses. Principal differences between the separate entries in Table G-3 are in the concentrations of niobium and cobalt. As the footnote in the table indicates, the low values of niobium deduced from the radioactivity data may result from inaccurate estimates of neutron capture cross section values. Cobalt is important not because it activates to a direct classification nuclide but because it gives rise to an activity that is measured directly and to which all other activities are indirectly related in practical classification techniques.³

1.3 Volume Estimates of Light Water Reactor GTCC LLW

Table G-4 summarizes the estimates of volumes of GTCC irradiated hardware wastes that result from normal operation and decommissioning of commercial nuclear power reactors. The estimates in the table are for typical 1000 MWe reactors operated for 40 years with a duty cycle of 75% full power operation. The values will not differ significantly if plant lifetime extensions add 10 or 20 years to the operation.

PWR

Data indicate that PWRs in normal operation generate only fuel assembly skeletons that have any real significance for irradiated hardware GTCC LLW. They also show that only the inconel-718 components of these skeletons will be GTCC, and only Westinghouse assemblies contain substantive quantities of inconel (in the grid support spacers). Table G-4 shows the volumes per Westinghouse assembly, since the present evaluation did not survey industry to ascertain the number of Westinghouse assemblies in use. The total volume of skeletons generated by industry is about 15 m³ per year, based on the data presented in Section 2.3.1.

An assumption that one-third are manufactured by Westinghouse and that the inconel portions are separated for disposal as GTCC yields an estimate of about 1 m³ per year of GTCC LLW. Such an analysis may be moot since the present status of thinking on fuel reconsolidation suggests that the assemblies will remain with the fuel and be a portion of the high-level waste. In such an event, the volume of GTCC LLW from normal operation of PWRs is estimated at zero. As the table indicates, for components from decommissioning, the high values of concentrations show a GTCC classification for the PWR shroud and core barrel and the BWR core shroud, whereas low values are well below this classification. Only the PWR shroud is in a GTCC category in the base case. Values in the table are the volumes per reactor. These estimates of decommissioning PWR GTCC LLW depend upon the actual neutron flux at the shroud and core barrels during reactor operation. The high values assume neutron fluxes at the levels predicted by the computations in Reference 1, whereas the low values use flux values scaled downward by the same ratio as found for BWRs (see the discussion in

Table G-3. Summary of data base for elemental constituents of reactor hardware alloys

MATERIAL	SOURCE	WEIGHT PERCENT					
		Ni	Co	Nb	Mo	Zr	Sn
<u>STAINLESS STEEL-304</u>	Nominal(2)	8.92	0.080	0.101	0.00	0.00	0.00
<u>BWR uses:</u>	NUREG/CR-						
CONTROL-ROD BLADES,	3474(1)	10.0+,-0.7	0.14+,-0.08	0.009+,-0.009	0.26+,-0.15	-	-
IN-CORE INSTRUMENTS,	Table B1	8.9+,-0.5	0.10+,-0.06	0.014+,-0.011	0.39+,-0.22	-	-
FUEL SUPPORT PIECES,	Table B2	8.9 (#1)	0.120	0.003 (#2)	0.15	-	-
FEED-WATER SPARGERS,							
POISON CURTAINS,							
CONTROL-ROD GUIDES,							
CORE SHROUD							
<u>PWR uses:</u>							
FUEL ASSEMBLY PIECES,							
SHROUD,							
CORE BARREL							
<u>ZIRCALOY-4</u>	Nominal(2)	0.002	0.001	0.012	0.00	98.0	1.6
<u>BWR uses:</u>	Table B1	<0.005	<0.0027	-	0.87 (#3)	94+,-11	2.7+,-0.6
FUEL FLOW CHANNELS	Table B2	0.021 (#4)	0.0014	8.4E-6 (#2)	-	-	-
<u>PWR uses:</u>							
FUEL ASSEMBLY PIECES							
<u>INCONEL-718</u>	Nominal(2)	52.0	0.47	5.55	3.0	0.0	0.0
<u>PWR uses:</u>	Table B1	49.7+,-2.5	0.087+,-0.036	4.4+,-1.1	2.79+,-0.11	-	-
FUEL ASSEMBLY PIECES	Table B2	49.7(#1)	0.16	1.1(#2)	-	-	-
#1 Value used to normalize scaling factor results.							
#2 Scaling factor results for Nb analyses show values about a factor of three lower than those from elemental analyses. The difference may arise from inaccurate neutron cross section values used in ORIGEN-2 computations (see discussion in text).							
#3 This is a "single-analysis" result.							
#4 Normalizing value results from analyses of concentration data in reference 3 and activation values from reference 4, discussed in the text.							

Table G-4. Estimates of volumes of LWR GTCC waste

TABLE 1.4 ESTIMATES OF VOLUMES OF LWR GTCC WASTE

COMPONENT REMOVED	YEARLY VOLUME OF GTCC WASTE	
	(M-3)	
PWR		
<u>FUEL ASSEMBLY</u>		
SS-304, ZIRCALOY-4	High:	0.0
	Base:	0.0
	Low:	0.0
INCONEL-718	High:	6.4E-4 (#1)
	Base:	6.4E-4
	Low:	6.4E-4
<u>SHROUD</u>	High:	1.6 (#2)
	Base:	1.6
	Low:	0.0
<u>CORE BARREL</u>	High:	3.4 (#2)
	Base:	0.0
	Low:	0.0
BWR		
<u>CONTROL-ROD BLADE</u>	High:	0.0
	Base:	0.0
	Low:	0.0
<u>IN-CORE INSTRUMENTS</u>		
LPRM	High:	0.12 (#3)
	Base:	0.09
	Low:	0.05
IRM	High:	1.6E-3 (#2)
	Base:	1.6E-03
	Low:	1.6E-03
SRM	High:	8.0E-4 (#2)
	Base:	8.0E-04
	Low:	8.0E-04
<u>POISON CURTAINS, FUEL-SUPPORT PIECE, FUEL FLOW CHANNEL</u>	High:	0.0
	Base:	0.0
	Low:	0.0
<u>CORE SHROUD</u>	High:	3.3 (#2)
	Base:	0.0
	Low:	0.0
#1. Inconel-718 is used for grid spacers only in Westinghouse assemblies. The value shown here is per Wesinghouse assembly.		
#2. These components are normally removed only at the end of reactor life. The values given are per reactor.		
#3. These values assume that packagers mix cold ends with hot ends and average over the volume of items in each package.		

Section 3.2). The base is a log-mean average of the two and presents the best estimate that can be made from the limited data base. The values have high uncertainties.

BWR

Analyses indicate that BWRs in normal operation produce GTCC material only in the form of incore instrumentation. LPRMs are low-volume SS-304 tubes that have generally been combined with other SS-304 components in burial casks for concentration averaging over the combined volume of material. IRM and SRM instrument tubes released at the end of plant life would be considerably more active and be GTCC material. The estimated volumes in the table are yearly total projections for all BWR plants except for the core shroud; the values are per plant. The high estimate assumes the high nuclide concentrations (using only the hot ends) in Table G-1 and that all of the instruments are shipped as GTCC LLW, whereas the low estimate assumes the low concentrations and that some averaging occurs. Base values are linear averages. Only the core shroud has potential for GTCC classification in wastes from BWR decommissioning. Estimates of GTCC volumes from this component have the high value from the calculated estimates in Reference 2 and the low value computed using the measured neutron fluxes at the shroud from Reference 21 (discussed in Section 3.2). Base values assume the base concentrations in Table G-2. The table indicates the BWR core shroud is probably not a candidate for GTCC LLW.

1.4 Recommendations for Efforts to Reduce Uncertainties in Estimated Volumes

The source of GTCC irradiated hardware waste with the largest potential volume is that arising from decommissioning of commercial nuclear power reactors. Estimates of projected volumes of these wastes have the greatest uncertainties, arising from a dearth of data on activities in these components that lie outside the reactor core. Principal causes of the large values are uncertainties in: (a) thermal and epithermal neutron flux values in these regions, (b) neutron capture cross sections--thermal and resonance--for important activations, especially for nickel and niobium, (c) the range of trace elemental constituents in the stainless steel-304 used in these components--cobalt and niobium--and (d) methods for practical waste classification of the large components resulting from decommissioning. Recommendations to decrease the spread in the estimated volumes include:

1. Perform underwater dose rate profiles of the core shroud in the three BWR plants retired from service--Humbolt Bay, Dresden Unit 1, and LaCross. These plants have the fuel removed from the core, and access to the shroud could be made easily by removing the pressure vessel top. The average dose rate of the shroud would be combined with dose to curie factors computed for this simple geometry and with decay- and irradiation-history-corrected scaling factors (from Appendix A) to estimate concentrations of the relevant nuclides. These values would be used to improve the neutron flux estimates for the larger reactors as well as for scaling the measured activities to the larger reactors and longer service life. Similar studies would occur at the retired Rancho Seco PWR on the shroud and core barrel.

2. Evaluate the available neutron cross section data--thermal and resonance--for nickel and niobium activation to ^{59}Ni , ^{63}Ni , and ^{94}Nb to obtain the best activation values. The evaluation would use reactor activation measurements using neutron energy-selective absorbers to separate thermal from epithermal activation and would utilize the DOE National Data Centers to access additional data and evaluations of known data on these cross sections.
3. Perform additional searches for industry data of measured or estimated neutron fluxes in the regions of core shrouds and barrels, particularly for PWRs. This would require additional contacts with the utility industry and with the three principal manufacturers of U.S. PWR plants.

1.5 Conclusions

1. PWRs generate little or no GTCC LLW in normal operational shutdowns; fuel assembly skeletons will probably remain with the spent fuel and go to the high-level waste repository.
2. BWRs, in normal refueling, generate only incore instruments that have GTCC LLW classification; they can be mixed with the higher volume CRBs, where concentration averaging will yield Class C waste.
3. It is uncertain whether PWR shrouds will be GTCC LLW at decommissioning because of uncertainties in neutron flux values at their location outside the core; core barrels will probably not classify as GTCC LLW.
4. BWR core shrouds will likely not be classified as GTCC material at decommissioning.
5. Data are needed from which to gain additional confirmation of activities in core shrouds and barrels.

2. BACKGROUND

2.1 Purpose and Scope of Evaluation

The U.S. Department of Energy is obligated to provide for disposal of high-level radioactive waste and low-level wastes that exceed the radioactivity limits for disposal in low-level surface repositories operated by the state compacts. Disposal of high-level waste, which consists primarily of spent nuclear fuel and concentrated fission product and transuranic activities that result from chemical processing of nuclear fuels, is separate from disposal of the low-level wastes and will probably be in deep repositories. The higher activity low-level wastes will require different facilities for their internment. These wastes come from many sources, and their activity levels and volumes are less known. Medical hospitals and research centers, industry (particularly the nondestructive testing segment), and electric utilities all generate significant volumes of GTCC LLW. To a large extent, levels of activity and volume for this material are unknown since disposal of the wastes has been minimal until the present. It is important to estimate these quantities in order to provide proper input for design of disposal facilities. EG&G Idaho, Inc., is developing a data base from which to project DOE needs for storage and disposal of this class of low-level wastes and desires assistance in providing additional data and assessments of current data in the base.

The purpose of the current study is to augment the EG&G data base by assessing current data and supplying new data on utility generated, low-level wastes that must be accepted by DOE. The emphasis is on nonfuel-bearing activated metal components used in operation of the utility reactors since studies have treated the other sources. Activated hardware results from normal refueling and maintenance operations and from decommissioning at the end of the plant life, currently presumed to occur after 40 years of operation. Current assumptions on the activity levels and volumes resulted from two studies by Battelle Pacific Northwest Laboratories (PNL).^{4,5} The present study will also evaluate assumptions in the current base against the new data and provide high (worst case) and low (best case) estimates of both the activity and volume levels for future GTCC LLW. Sources of new material are utility and industry data on classification and disposal of irradiated hardware as well as data on reactor operations. Only limited data are available from current or past decommissioning activities, and evaluations used data from other sources.

2.2 Definition and Classification of Greater-Than-Class C Wastes

In December 1982, the U.S. Nuclear Regulatory Commission (NRC) published an addition to the Code of Federal Regulations, 10 CFR Part 61, to take effect in December 1983 (Reference 1). The code defined three categories of low-level radioactive waste suitable for shallow-land disposal and some packaging regulations for the waste. Definition of the wastes was in terms of specific radionuclide concentrations, and the rules established Classes A, B, and C wastes for commercial disposal, Class C allowing the highest concentrations of these nuclides. Similarly, 10 CFR Part 60,⁶ defines high-level radioactive wastes. Wastes having activity concentrations higher than the limits in 10 CFR Part 61 for Class C material but falling outside the definition of high-level waste in 10 CFR Part 60 are designated as greater-than-Class C (GTCC) low-level wastes. DOE has been assigned the responsibility for disposal of this material.⁷ The NRC classification of wastes separates radionuclides into two categories, long-lived (Table 1) and short-lived (Table 2) nuclides. Nuclides

of relevance to irradiated hardware in Table 1 are ^{14}C , ^{59}Ni , ^{94}Nb , ^{99}Tc , and long-lived transuranics, where the Class C maximum permissible activity concentrations are, respectively, 80, 220, 3, and 0.2 Ci/m³ and 100 nCi/g. Similarly, the relevant nuclide in Table 2 is ^{63}Ni , where the upper limit for Class C is 7000 Ci/m³. In addition to individual concentration limits, the rules define a "sum of the fractions" limit for the nuclides in each table where the sum of the fractional limits for each nuclide (i.e., the concentration divided by the concentration limit) may not exceed unity (sum of the fractions rule). Each package or component of low-level radioactive waste must be classified as either Class A, B, C, or GTCC material before shipping for disposal. An additional rule states that the shipping manifest must list concentrations of specific nuclides even though their activities are too low to influence the classification of the material. These nuclides include most of those listed above as well as ^{129}I , a fission product.

The radionuclides listed above as relevant in irradiated metal do not emit radiation that can be detected in the presence of other activities in the components, principally ^{60}Co , whose allowed concentration is unlimited in Class C waste because of its short (five-year) half-life. Their concentrations can only be measured directly through destructive radiochemical analyses that are impractical for each item. Unless the component is analyzed as a whole: (a) representative sampling is nearly impossible, (b) radiation levels are too high for analyses of more than very small samples, and (c) the cost is prohibitive. The NRC has provided guidance⁸ suggesting two alternative methods for classification of activated hardware: use of nuclide scaling factors determined through measurements and through activation analysis. A scaling factor is a ratio of activity concentrations of the desired nuclide (scaled nuclide) to a scaling nuclide (^{60}Co for hardware). Both methods rely on determining experimentally the average radiation dose rate of each package and applying a dose to curie factor to convert average dose rate to average concentration of the dose contributors, nearly always ^{60}Co .

The two methods differ in the determination of scaling factors to ^{60}Co . The measurement technique radiochemically analyzes small samples of the activated metal to measure the scaling factors for ^{14}C , ^{59}Ni , ^{63}Ni , ^{94}Nb , ^{99}Tc and TRU (TRU comes from surface contamination). Small samples are justified in this case since scaling factors require only activity ratios. Activation analysis uses assumed elemental constituent abundances of the component alloy, a data base of thermal and epithermal neutron cross sections, an estimated thermal and epithermal neutron flux, and the radiation history of the component to compute the scaling factors. (Activation analysis calculations also yield estimates of activity concentrations that are sometimes used where dose rate measurements are not available.) Activation analysis calculations usually use the two computer codes ORIGEN2⁹ and ANISN.¹⁰

The NRC also allows the packaging several components in the same disposal cask and averaging the nuclide concentrations over the volume of similar components in the container (Reference 8). Rules for the averaging require that the components be similar (i.e., same material and same geometry) and that the average concentration in each component not differ from that in the others by more than a factor of ten (factor of ten rule).

2.3 Sources of Possible GTCC Hardware

There are two sources of irradiated components that could have activity concentrations that exceed the Class C limits, those that are changed in normal reactor refueling or maintenance activities, and those that are large structures internal to the reactor pressure vessel and will come from major reconstruction or decommissioning. Some of the components associated with refueling, such as PWR fuel skeletons, may remain with the fuel and thus be shipped with the high-level waste.

2.3.1 Components from Routine Refueling and Maintenance

Table G-5 lists the principal hardware items of irradiated hardware generated in refueling and maintenance operations that could result in GTCC LLW. As can be seen from the listing, the main source is the BWR.

PWR Components. The large potential source from PWRs is the fuel assembly skeleton, shown pictorially in Figure G-1. There are five manufacturers of the skeletons used by the utilities Westinghouse, Exxon, General Electric, Combustion Engineering, and Babcock & Wilcox. Four of the manufacturers produce skeletons that are fundamentally the same from a waste classification standpoint, whereas the Westinghouse assembly differs by having a substantial amount of inconel-718 as grid support (or grid spacers); the others use zircaloy-4 for that component. Other materials in the assembly are SS-304 and zircaloy-4 as shown in Table G-5. The table also lists the approximate mass and volume for each of the major components in a single assembly. A typical large PWR has about 300 fuel assemblies in the reactor core during operation. Approximately a third of these are discharged in a refueling operation that occurs every 12-18 months. Consideration of fuel reconsolidation (removal of the fuel rods from spent assemblies) has received study recently¹¹⁻¹⁶ in efforts to gain more available space in the utility spent fuel pools. If reconsolidation is done, the skeletons are potential GTCC LLW; if not, the skeletons will probably remain with the spent fuel to go to the high-level waste repositories.

Other PWR incore components are thimble plug assemblies and burnable poison assemblies that have relatively low volumes. These components spend most of their incore life in regions of relatively low flux and do not have serious potential for being GTCC LLW.

BWR Components. Table G-5 also lists six components of irradiated hardware that are removed periodically from BWRs during outages. These pieces are considered possible GTCC LLW.

Control Rod Blades (CRBs). The CRB is the largest volume source of irradiated hardware discharged in refueling operations. Table G-5 lists the typical volume and mass of each blade, and Figure G-2 shows a drawing of a typical CRB. The shape is that of a cruciform, and the materials are SS-304, except for boron carbide pellets contained inside thin-wall tubes in the blades and bearing rollers of stellite components). Because the cobalt activates to such a high activity level, the rollers are usually separated from the spent blades while they are in the spent fuel pool. The boron carbide adds little activity to the component except ^3H , but even this is at sufficiently low concentration that it doesn't influence the classification. BWR cores have 150-200 CRBs in place at any given time, and approximately 15-30 blades are discharged during each refueling cycle that occurs every 12-18 months.

Table G-5. Summary of relevant characteristics of activated hardware generated in normal reactor operations

COMPONENT	MANUFACTURER	MATERIAL	VOLUME (M-3)	MASS (Kg)
PWR				
FUEL ASSEMBLIES				
TOP AND BOTTOM NOZZELS (TIE PLATES)	W,E,G,C,B	SS-304	1.3E-03	11.0
GUIDE TUBES	W,E,G,C,B	ZIRCALOY-4	1.2E-03	9.0
INSTRUMENT TUBES				
HOLD-DOWN SPRINGS	W,E,G,C,B	INCONEL-718	2.2E-04	1.8
SPACER GRIDS	W E,G,C,B	INCONEL-718 ZIRCALOY-4	6.4E-04	5.3
BWR				
CONTROL-ROD BLADES (CRUCIFIX)		SS-304 (B-4C)	2.7E-02	210.0
IN-CORE INSTRUMENTS		SS-304	2.0E-04	
POISON CURTAINS		SS-304	2.1E-02	160.0
SPARGER NOZZLES		SS-304	7.3E-02	
FUEL-SUPPORT PIECE		SS-304	3.5E-03	27.0
FUEL FLOW CHANNELS		ZIRCALOY-4	5.3E-03	42.0
W: Westinghouse, E: Exxon Nuclear, G: General Electric Co. C: Combustion Enginnering, B: Babcock and Wilcox

Westinghouse 17 x 17 PWR fuel assembly

Figure G-1. Sketch of typical PWR fuel assembly skeleton.

Figure G-2. Sketch of typical BWR control rod blade.

Incore Instruments. A BWR uses three forms of incore instruments that are potential GTCC LLW when discharged: local power range monitors (LPRM), intermediate range monitors (IRM), and source range monitors (SRM). The construction is a thin-wall tube of SS-304 about 13 m long containing typically four fission chambers with mg quantities of different isotopes of uranium. The fission chambers are easily removed from used instruments when necessary, although the chambers do not influence classification. A typical BWR core contains about 100 LPRMs, 8 IRMs, and 4 SRMs. The geometries, weights, and masses of the three, listed in Table G-5, are about the same, although the fission chambers are different. About 10–30 LPRMs are discharged each refueling cycle (12–18 months). IRM and SRM tubes are intended to remain the life of the core but are occasionally discharged for various reasons. Each instrument consists of a 3.7-m hot end that is in the core during service and a 9.4-m cold end that sees few neutrons.

Poison Curtains. Poison curtains are flat plates of borated SS-304 that are located near the core during the reactor's first cycle to adjust the flux distribution. They are usually removed after that cycle since partially burned fuel can fulfill their function. Table G-5 lists the volume and mass of typical poison curtains. Although the curtains see a relatively high flux, they remain in place for only one cycle.

Sparger Nozzles, Fuel Support Pieces, Tie Plates. The feedwater sparger nozzle is a large piece of SS-304 that is changed in maintenance when it shows evidence of stress corrosion cracking or is removed at decommissioning. It sits in a relatively low neutron flux while in core. The fuel support pieces and tie plates are relatively low-volume items of SS-304 that are near the top of the core during use.

Fuel Flow Channels. These components are hollow, square cross section pieces of zircaloy-4 alloy that surround each fuel element. They contain small flow channels for cooling the fuel. There are approximately 500–840 channels in a BWR core, one for each fuel assembly. They remain with the assembly and are discharged with the fuel but are separated after discharge. Thus, they remain in core for approximately three refueling cycles, or 36–54 months.

2.3.2 Components from Decommissioning

Particularly important candidates for GTCC LLW remain in the reactor pressure vessel during the life of the plant. Table G-6 lists these components and gives their characteristics. Shrouds and core barrels are rather massive systems and could contribute significantly to the volume of GTCC LLW if they are so classified. They surround the core inside the pressure vessel. In a PWR, the shroud lies just outside the region of fuel, and the core barrel lies about 19 cm outside the shroud. The neutron flux falls off relatively fast in these regions. The BWR core shroud lies about 20 cm beyond the fuel, and, here also, the neutron flux falls off rapidly in this region.

Table G-6. Characteristics of relevant activated hardware from decommissioning

COMPONENT	MATERIAL	VOLUME (M-3)	MASS (Tonne)
<u>PWR</u>			
SHROUD	SS-304	1.6	12.4
CORE BARREL	SS-304	3.4	26.5
<u>BWR</u>			
CORE SHROUD	SS-304	3.5	27.0

2.4 Status of Knowledge Base for Irradiated Hardware

The current DOE knowledge base for utility-generated waste with potential for classification as GTCC is relatively sparse. A survey taken of utilities turned up only limited information, primarily because the plants had not classified many of the wastes and did not know the activity concentrations. No measurement data are identified for activity levels or radiation dose rates of core shrouds or barrels. The data base relies heavily on the data and computational results reported in two NUREG documents issued in 1980 (References 4 and 5) on activities and volumes of irradiated hardware projected to be GTCC LLW. These estimates, particularly for decommissioning waste, use the results of neutron flux projections for areas outside the core for input into the activation calculations. They also use the activation cross sections in ORIGEN2 (Reference 9) and the nominal elemental abundances in the alloys. Activation cross sections for three of the important activities, ^{59}Ni , ^{63}Ni , and ^{94}Nb , are not well known¹⁷, either in the thermal or epithermal regions. No data are available on cross sections for activation of the individual isotopes of nickel. Only data on activation of a naturally occurring nickel element are in the literature, and, since the element contains five stable isotopes, estimates of individual cross sections have considerable uncertainty. For niobium, even though it has only a single stable isotope, the thermal and resonance energy activation cross sections are not reported, and the scatter in the data for total cross section gives considerable uncertainties in the accuracy of the estimated values.

Data from power reactors that have retired from service (Humbolt Bay, Dresden Unit 1, Rancho Seco, LaCross, and Shippingport) are not available for the core shrouds and barrels. No decommissioning demolition or activity surveys have been made at the first four plants,^a and the Shippingport reactor pressure vessel was buried in toto after being filled with concrete; no radiation surveys were made of the reactor internals.¹⁸ Other uncertainty concerns in projecting GTCC needs for disposal of irradiated hardware components include future rulings for averaging concentrations over packaged components. If the current NRC interpretation (discussed in Section 2.2) is allowed by the regional compact sites, the volume of GTCC LLW will be considerably lower than that from a less favorable set of rules. The compacts have not seriously addressed this issue yet, and only speculation can be used for estimates. This report assumes the current regulations will continue.

a. Personal communication with Paul Roth, Commonwealth Edison Co., December 1990, and personal communication with Seymour Rafferty, Dairyland Power Cooperative, December 1990.

3. COMPILATION OF DATA OF HARDWARE RADIOACTIVITY AND VOLUMES

3.1 Operations-Generated Components

Utilities have removed and shipped for disposal a substantial amount of irradiated hardware items from their spent fuel pools. These items have primarily been discharged from the reactor during refueling operations. Characterization and classification of the components was necessary before shipment, and the data are important to estimates of future GTCC materials generated in these activities. Many of the methods for classification used measurements of radionuclide scaling factors, stable element abundances in the component alloys, dose to curie conversion factors for ^{60}Co , or average dose rates for the components. These data are very important for adding to the knowledge data base and for comparisons to the computational estimates that form the basis for many of the current projections. Appendices A, B, and C include the important data from these measurements.

3.1.1 Components from Boiling Water Reactors

Table G-7 summarizes the data on operations-generated irradiated hardware from BWRs. The values in the table are summary results of the data in the three appendices. The first column lists the eight components considered and the approximate volume of each. The second column labels the maximum, minimum, and average dose rate and radionuclide concentration values for the data included in the summary. The third column gives the values for the average contact dose rate, each averaged from dose rate profiles taken along the component while it was in the fuel pool. The average is a linear average of the many measurements for that type of component. Minimum and maximum values are those occurring in the reported measurements. Data labeled dose to curie factor show the average measured dose to curie factor $[(\text{Ci}/\text{m}^3)/(\text{R}/\text{hr})]$ for ^{60}Co (converting a dose rate to a nuclide concentration).

Reference 3 discusses the method for measuring this quantity that can also be computed.¹⁹ Computed and measured factors are in close agreement. The uncertainties in the factors are RMS values of the dispersion in the experimental results. The fifth column gives the concentration of ^{60}Co from converting the dose rates to curie concentrations through the dose to curie factor. The last four columns give the concentrations of ^{63}Ni , ^{94}Nb , ^{99}Tc , and TRU, respectively. These values result from multiplying the ^{60}Co concentration in column 5 by the appropriate nuclide scaling factor, whose values derive from Appendix A. TRU, expressed in the required units of nCi/g rather than Ci/m^3 , is not from bulk activation of the metal but from surface deposition of material on the component during its tenure in the reactor pressure vessel. Unless the tightly bonded surface crud is removed, it becomes a portion of the component for classification and disposal. TRU is the more restrictive of the surface contaminants. Data in Table G-7 show that the control rod blades, incore instruments and poison curtains contain the higher concentrations of all activities.

Table G-7. Average radionuclide concentrations for activated reactor components changed out in normal operations

COMPONENT		AVERAGE CONTACT DOSE RATE (kR/Hr)	DOSE-TO- CURIE FACTOR (Co-60)	AVERAGE CONCENTRATION				
				of Co-60 (Ci/M-3)	of Ni-63 (Ci/M-3)	of Nb-94 (Ci/M-3)	of Tc-99 (Ci/M-3)	of TRU (nCi/gm)
CRB VOLUME: .0266 M-3	MIN:	3.3	3.3	1.1E+04	6.9E+02	5.6E-03	9.6E-04	0.32
	MAX:	15.6	+1.3	5.1E+04	3.2E+03	2.7E-02	4.5E-03	1.52
	AVG:	8.2	-0.9	2.7E+04	1.7E+03	1.4E-02	2.4E-03	0.80
LPRM VOLUME 2E-4 M-3	MIN:	1.1	55.1	6.0E+04	3.8E+03	3.1E-02	5.3E-03	1.78
	MAX:	19.8	+11.1	1.1E+06	6.9E+04	5.6E-01	9.6E-02	32.30
	AVG:	8.5	-9.2	4.7E+05	3.0E+04	2.4E-01	4.1E-02	13.92
IRM VOLUME 2E-4 M-3	MIN:	8.1	55.1	4.4E+05	2.8E+04	2.3E-01	3.9E-02	13.15
	MAX:	20.6	+11.1	1.1E+06	7.2E+04	5.9E-01	1.0E-01	33.61
	AVG:	12.1	-9.2	6.6E+05	4.2E+04	3.4E-01	5.9E-02	19.66
SRM VOLUME 2E-4 M-3	MIN:	10.7	55.1	5.9E+05	3.7E+04	3.1E-01	5.2E-02	17.49
	MAX:	11.6	+11.1	6.4E+05	4.0E+04	3.3E-01	5.7E-02	18.97
	AVG:	11.2	-9.2	6.2E+05	3.9E+04	3.2E-01	5.4E-02	18.24
POISON CURTAINS VOLUME: 2.1E-2 M-3	MIN:	0.99	40.0	4.0E+04	2.5E+03	2.0E-02	3.5E-03	1.17
	MAX:	1.86	+14.9	7.4E+04	4.7E+03	3.8E-02	6.6E-03	2.20
	AVG:	1.43	-14.9	5.7E+04	3.6E+03	3.0E-02	5.0E-03	1.69
SPARGER NOZZLE VOLUME: 7.3E-2 M-3	MIN:	0.01	6.7	6.7E+01	4.2E+00	3.5E-05	5.9E-06	0.002
	MAX:	0.06		4.0E+02	2.5E+01	2.1E-04	3.5E-05	0.012
	AVG:	0.04		2.7E+02	1.7E+01	1.4E-04	2.4E-05	0.008
FUEL-SUPPOR PIECE VOLUME: 3.5E-3 M-3	MIN:	N.A.	6.7	N.A.	N.A.	N.A.	N.A.	N.A.
	MAX:	N.A.		N.A.	N.A.	N.A.	N.A.	N.A.
	AVG:	0.26		1.7E+03	1.1E+02	9.0E-04	1.5E-04	0.052
FUEL FLOW CHANNELS VOLUME: 5.3E-3 M-3	MIN:	0.023	4.3	9.9E+01	1.2E+00	4.3E-04	N.A.	0.866
	MAX:	0.378	+6.5	1.6E+03	2.0E+01	7.0E-03	N.A.	14.239
	AVG:	0.131	-2.9	5.6E+02	6.8E+00	2.4E-03	N.A.	4.942

3.1.2 Pressurized Water Reactors

Components removed in refueling and maintenance operations in PWRs relevant to GTCC consideration are limited to fuel skeletons. Table G-8 summarizes the measurement data from PWR fuel assembly skeletons given in the appendices. Industry uses five types of skeletons made by Westinghouse, Exxon, General Electric, Combustion Engineering, and Babcock & Wilcox. The table lists characteristics of assemblies made by four of these manufacturers and gives the approximate volumes and masses of the main components in each. Assemblies made by Westinghouse differ substantively from the others in having grid spacers of inconel-718. The similar Babcock & Wilcox assemblies, not listed, also have zircaloy-4 grid spacers. Volumes and masses for the components in each of the five types of skeletons are similar. There is not so much data for PWR skeletons as for the BWR hardware, and Table G-8 lists the reference for each set of measurements. Dose to curie values are given only for measurements of Westinghouse skeletons reported in Reference 11. The other measurements took small samples of material from each portion and assumed that the concentrations were representative of the entire item, and classification measurements did not include determining the average dose rate of components. The last three columns in the table give a summary of the reported concentrations of ^{60}Co , ^{63}Ni , and ^{94}Nb . In all cases, scaling of ^{63}Ni and ^{94}Nb concentrations from ^{60}Co used measured scaling factors from radiochemical analyses.

Results from the measurements by the two organizations are significantly different where a comparison can be made for the Westinghouse skeletons. Factors of over 100 difference between some reported concentrations are hard to explain. The data from Table G-8, reference 20, are an average of 36 assemblies, all using the same set of scaling factors; the other data in the table are from smaller numbers of samplings. The skeletons analyzed for the data in the table experienced similar total neutron fluences ($\approx 35,000 \text{ MW} \cdot \text{d/MTU}$) and should show comparable concentrations for the same alloy. Thus, the bottom of the table shows log-mean averages and dispersions computed for each material (calculations of log-mean average and dispersion values are described in Appendix A). Values of dispersions (average factors of deviation from the average) are large, especially for nickel and niobium activities.

3.2 Decommissioning Generated Components

Almost no data are available on activity concentrations of reactor internals for either operating plants or plants that have been shut down at the ends of their lifetimes. Because of the very short operating times, data from decommissioning of the Elk River and Pathfinder reactors have limited applicability to large scale light water reactors with 40-year operating experience, and, as was previously discussed, no measurements data can be found for four other closed reactors. In estimating the activity concentrations and volumes of the reactor internal components, reliance must be on available neutron flux, activation cross section, alloy elemental abundance data, and on computations that use these estimates.

3.2.1 Boiling Water Reactors

PNL has computed and reported (References 2 and 4) expected radial neutron fluxes--thermal, epithermal, and fast--in regions beyond the core in large (1000 MWe) BWRs. The reports also give the axial flux along the core center. Using these values, the two references predict the activity concentrations expected after a 40-year life of the BWR as far beyond the core as the biological

Table G-8. Measured radionuclide concentrations in fuel assembly skeleton

COMPONENT	MASS Kg	VOLUME M-3	REFER- ENCE	DOSE- RATE KR/Hr	CONCENTRATION Ci/M-3		
					Co-60	Ni-63	Nb-94
WESTINGHOUSE PWR FUEL ASSEMBLY (Average Burnup 33,000 MWD/MTU)							
Upper & Lower Tie Plate — SS-304	8.83	1.1E-03	20	1.4	9.4E+03	5.9E+02	4.9E-03
Spacer Grids Inconel-718	5.2	6.4E-04	20	2.3	9.4E+04	2.0E+04	2.0E-02
			27		1.6E+04	4.1E+03	2.5E+00
			27		7.2E+05	5.4E+05	6.6E+02
			23		4.0E+04	3.1E+04	1.2E+02
Guide Tubes Zircaloy-4	7.8	1.2E-03	20	0.19	9.9E+03	1.2E+02	4.3E-02
EXXON PWR FUEL ASSEMBLY (Average Burnup 33,000 MWD/MTU)							
Upper & Lower Tie Plate — SS-304	9.67	1.2E-03	20		1.1E+03	3.3E+02	5.0E-02
Hold-down Springs Inconel-78	1.0	1.2E-04	20				
Spacer Grid, Guide & Instrument Tubes Zircaloy-4	12.7	1.9E-03	20		2.5E+02	3.7E+01	1.3E-02
GENERAL ELECTRIC PWR FUEL ASSEMBLY (Average Burnup 27,500 MWD/MTU)							
Upper & Lower Tie Plate — SS-304	9	1.2E-03	27		3.9E+04	5.1E+03	1.8E-01
Spacer Grid, Guide & Instrument Tubes Zircaloy-4	12	1.8E-03	27		1.6E+03	2.0E+02	3.3E-02
COMBUSTION ENGINEERING PWR FUEL ASSEMBLY (Average Burnup 41,800 MWD/MTU)							
Upper & Lower Tie Plate — SS-304	9	1.2E-03	27		6.3E+04	1.3E+04	3.4E-02
Spacer Grid, Guide & Instrument Tubes Zircaloy-4	12	1.8E-03	27		1.1E+03	5.5E+01	1.9E-01
Hold-down Springs Inconel-78	0.1	1.2E-05	27		3.6E+05	3.7E+05	2.1E+02
SS-304				AVG:	1.9E+04	3.0E+03	3.1E-02
				DISP:	6.1	6.3	3.7
ZIRCALOY-4				AVG:	2.8E+03	2.7E+02	2.1E-01
				DISP:	7.2	15.2	39.4
INCONEL-718				AVG:	1.6E+05	9.3E+04	7.0E+01
				DISP:	7.7	15.2	19.2

shield. The computations show that only the BWR core shroud is important for consideration as GTCC LLW. In 1980 General Electric Co. at the Brown's Ferry Unit 3 plant instrumented the region between the core shroud and the pressure vessel with neutron flux monitors to determine thermal, epithermal, and fast neutron flux in this region.^{20,b} Oxides of ^{235}U , ^{238}U , ^{237}Np and ^{232}Th , salts of Fe, Ni, Cu, and Ti, and AlCo wires served as the neutron dosimeters. This combination (some capsules shielded with cadmium) gave redundant results for the entire neutron spectrum for verification purposes. Three radial locations were 4 and 30 cm from the shroud and 7 cm from the pressure vessel and at azimuthal angles of 3° and 45° around the core. The combination of locations showed radial axial and azimuthal flux variations and absolute values within this region of the vessel. Figure G-3 gives a cross-sectional view of the reactor pressure vessel and shows the locations of measurements relevant to flux values at the shroud and pressure vessel. Flux values at 45° should be slightly higher than those at 3° because the higher angle is at the corner of the reactor fuel array.

Measurements were taken at the core midplane and the core top. The values reported are substantially lower than those estimated in Reference 2. Table G-9 presents published values of neutron fluxes at the shrouds, calculated and measured, for both PWRs and BWRs. The table includes results for thermal, epithermal, and fast neutron fluxes from the BWR measurements taken at the location 4 cm from the shroud at both the midplane and core top. Computed estimates of flux in the table are from References 2 and 4, from ORIGEN2 and ANISN. The differences of nearly a factor of 100 between computed and measured values at the BWR core midplane and near 10 at the core top are very significant in estimating the activities and classification of the shrouds at decommissioning.

Other BWR reactor internal components close to the reactor core are the upper and lower fuel support pieces and feedwater sparger nozzles. Data presented in Section 3.1.1 included activities in feedwater spargers that had been removed after several years of use. The measured values indicated that the sparger activities, even at the end of the reactor life, would lie well below Class C limits. The limited data on the fuel support pieces and the neutron flux values estimated at their locations near the core indicate that these items also would not be candidates for GTCC LLW.

3.2.2 Pressurized Water Reactors

The two important components in PWRs that could qualify for GTCC classification are the shroud and the core barrel. Table G-9 includes the computed estimates of flux values at the location of these components in a large (1000 MWe) PWR. If the neutron fluxes are this high, the PWR shroud would constitute GTCC LLW. There are no identified measurement data similar to those made by GE for the BWR to compare to the calculations.

3.3.3 Summary

More measurement data are desirable from which to estimate the activities expected in the light water reactor (LWR) shrouds at the time of decommissioning, especially for PWRs where there are no available data. Radiation levels from these components in the shut down reactors at Humbolt Bay, Dresden, Rancho Seco and LaCross could be used to give valuable data.

b. Personal communication with Dr. Jerry Martin, General Electric Co., December 1990.

Reactor vessel cross section

Figure G-3. Cross section of BWR core showing positions of neutron flux measurements between the shroud and the pressure vessel.

Table G-9. Estimates of neutron flux at structural components

NEUTRON ENERGY RANGE		AVERAGE NEUTRON FLUX (n/cm-2/sec)	
<u>PWR</u>	<u>(#1)</u>	<u>SHROUD</u>	<u>CORE BARREL</u>
Thermal		1.3E+13	1.4E+12
Epithermal		3.5E+13	3.9E+12
Fast (>1 MeV)		7.0E+12	7.7E+11
Computed values from Reference 28; Computations used ORIGEN-2 and ANISN and are the flux at the core mid-plane			
<u>BWR CORE SHROUD</u>	<u>COMPUTED(#1)</u>	<u>MEASURED(#2)</u>	
		<u>@3 DEGREES</u>	<u>@45 DEGREES</u>
<u>CORE MID PLANE</u>			
Thermal	2.5E+13	3.0E+11	5.8E+11
Epithermal	3.0E+12	2.1E+11	3.8E+11
Fast (>1 MeV)	1.0E+12	7.6E+10	1.4E+11
<u>CORE TOP</u>			
Thermal	~6E+12	1.8E+11	
Epithermal	~5E+11	1.3E+11	
Fast (>1 MeV)	~1.5E+11	5.0E+10	
#1. Computed values from References 1 and 7 using ORIGEN2 and ANISN #2. Measured values taken from Reference 6.			

Table G-10 shows the results of estimates of activities in LWR internals at the time of decommissioning, 40-year lifetime with a full power factor of 75% (30 full power years). The table shows high, base, and low estimates of concentrations for the four radionuclides important for waste classification: ^{59}Ni , ^{63}Ni , ^{94}Nb , and ^{99}Tc . All values assume a 40-year lifetime for the reactor and a 75% duty cycle. The high values assume neutron fluxes at the level reported in Reference 2, and the low values assume values from Reference 21. The table applies the measured-to-computed flux ratios from the BWR data in Table G-9 to obtain the low values for PWRs. The ratio may not be valid unless the reasons for the difference between estimated and measured flux values outside the core in BWRs also apply to PWRs. Base values are a log-mean average of the high and low values.

The values listed for ^{59}Ni include an additional factor. ORIGEN2 computations and some measurements yielded a ratio for the two nickel activities, ^{63}Ni to ^{59}Ni , of about 150, whereas other measurements (References 3 and 14) reported the ratio as about 50. Two factors contribute to uncertainty in the actual ratio: (a) as previously discussed in Section 2.4, measured activation cross sections for ^{58}Ni and ^{62}Ni are not reported, and the computational codes assume a division of the total cross section for elemental nickel between its isotopes, and (b) no primary analytical standards are available for calibrating instrumentation in quantitative measurements of ^{59}Ni , and the laboratories rely on secondary standards. Recent measurements using a ^{59}Ni standard calibrated by the National Institute of Science and Technology (NIST)^c suggest that a ratio of about 60 may be appropriate. The high values for ^{59}Ni in Table G-10 use a ratio of 50 for computing the concentration, and the low values use a ratio of 150.

c. Private communication with F. Shima, National Institute of Science and Technology, August 1989.

Table G-10. Summary of high and low radionuclide estimates for potential GTCC-classified reactor internal components

COMPONENT	RANGE	RADIONUCLIDE CONCENTRATION (Ci/m-3)					
		Ni-59	Ni-63	Nb-94	Tc-99		
<u>PWR</u>	SHROUD	HIGH: (1)	1870	93000	2.1	0.66	
		BASE: (2)	140	12000	0.16	2.8E-02	
		LOW: (3)	10	1500	0.012	1.2E-03	
	CORE BARREL	HIGH: (1)	240	12600	0.15	0.03	
		BASE: (2)	19	1600	0.012	3.9E-03	
		LOW: (3)	1.4	200	9.2E-04	5.4E-04	
	<u>BWR</u>	CORE SHROUD	HIGH: (1)	600	30000	0.34	8.1E-03
			BASE: (2)	29	3900	0.026	1.1E-03
			LOW: (3)	1.3	500	0.002	1.4E-04

1. Flux and elemental constituent estimates from Reference 1, Ni-59 to Ni-63 ratio from references 5,16.

2. Log-mean average of high and low values.

3. Flux estimate from Reference 6; Elemental constituent estimates from analysis of all available data (TABLE 1.2); Ni-59 to Ni-63 ratio from References 1 and 25.

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APPENDIX A

COMPILATION OF MEASURED RADIONUCLIDE SCALING FACTORS FOR IRRADIATED HARDWARE

APPENDIX A: COMPILATION OF MEASURED RADIONUCLIDE SCALING FACTORS FOR IRRADIATED HARDWARE

For the past several years, utilities have shipped for burial in commercial low-level waste sites irradiated reactor components generated in routine plant maintenance or refueling operations. The classification of many of these components used radionuclide scaling factors (activity ratios of specific non-gamma ray-emitting nuclides to ^{60}Co) resulting from radiochemical analysis of small samples of the component. The measured scaling factors are important data in assessing the activity levels in future irradiated components, both those acquired in normal plant operation and those resulting from future decommissioning. Three metals are commonly used for reactor internals: stainless steel-304 (SS-304), zircaloy-4 and inconel-318 (or 350). Even though ^{60}Co is the preeminent activity in each of these materials, dominating the dose rate and gamma ray spectrum, cobalt is, for the most part, an uncontrolled trace element in the metals. Hence, the ratio of the other and more important activities to the easily observed ^{60}Co can vary widely in different specimens of the same alloy.

Scaling to this activity, pragmatically used for classifying the components, can lead to results with large uncertainties. Furthermore, levels of other important constituents in these same alloys, niobium in SS-304 and zirconium-4, and nickel in zirconium-4, are also present in trace amounts, mostly uncontrolled, compounding the uncertainties. Compilation and analysis of the available scaling factor data, used in conjunction with other analyses, assist both in understanding the limitations in the use of these factors and in improving the accuracy of the analyses of irradiated hardware. Analyses of the data, summarized in the following sections, show that measured scaling factors have demonstrated reproducibility better than a factor of 5, well within the NRC limits of a factor of 10,¹ and can be used effectively with other data to classify radioactive components. Samples on which the analyses were made were grindings or cuttings from the component. An underwater sampling tool sampled both the bulk material and the surface contamination of many of the components. In other cases, samples were cuttings obtained with hot cell manipulators and cutting tools. The uncertainties reported for the analytical results were computed primarily from radioactive counting statistics and chemical yield determinations and do not reflect any potential biases from the sampling method. A typical magnitude for reported uncertainties was 20%, much smaller than the spread in the results from different samplings of the same alloy, dominantly caused by variations in the constituent concentrations. Radioanalytical procedures were methods adapted or developed by industry for analyzing utility radioactive wastes, maintained under inspected and audited Quality Control and Quality Assurance programs. Robertson et al.² reports some typical radioanalytic procedures used in the analyses.

Components of Stainless Steel-304

The bulk of the available data on measured scaling factors for SS-304 comes from analyses of BWR components, although there is a pair of data points from PWR fuel assembly end fittings. Nonetheless, the data should be equally applicable to PWR components since the effects of the different neutron spectrum in PWRs present only minor differences (well within the experimental tolerances of the measurements). Table G-11 gives a list of all the measured scaling factors that could be found in the literature survey. The table organizes them by reactor component. The second column in the table cites the reference for the data, listed in Section 4.0. The next five columns give the measured activities of the given five nuclides relative to ^{60}Co . The first four are activation products within the metal itself, whereas the TRU represents the transuranic material contained in

Table G-11. Measured activity concentration ratios in activated components of stainless steel (SS-304)

COMPONENT	REF.	MEASURED ACTIVITY RATIOS (SCALING FACTORS)								
		ACTIVITIES TO Co-60					ACTIVITIES TO Ni-63			
		C-14	Ni-63	Nb-94	Tc-99	TRU	C-14	Nb-94	Tc-99	TRU
CRB	28	2.1E-07	1.0E-01	2.6E-06			2.1E-06	2.6E-05		
	29	1.2E-06	1.1E-01	5.8E-07		2.0E-06	1.1E-05	5.3E-06		1.8E-05
	29	5.3E-07	6.6E-02	3.2E-07		1.3E-06	8.0E-06	4.8E-06		2.0E-05
	30	3.8E-06	6.1E-02	<2E-7		5.6E-08	6.2E-05			9.2E-07
	31	2.2E-05	8.3E-02	1.4E-06		1.2E-08	2.7E-04	1.7E-05		1.4E-07
	31	8.4E-04	1.0E-01	8.1E-07		8.1E-08	8.4E-03	8.1E-06		8.1E-07
	32	2.5E-05	1.9E-01	<2E-7			1.3E-04			
	32	6.1E-06	1.2E-01	4.8E-06		2.4E-08	5.1E-05	4.0E-05		2.0E-07
	32	4.2E-06	1.0E-01	9.1E-07		1.9E-06	4.2E-05	9.0E-06		1.9E-05
	34	3.4E-07	4.1E-03	<1E-9		1.6E-08	8.3E-05			3.9E-06
	35	1.4E-06	6.0E-02	1.3E-05		8.3E-09	2.3E-05	2.2E-04		1.4E-07
	36	4.8E-06	1.0E-01	<1E-6		1.6E-07	4.8E-05			1.6E-06
	37	5.5E-05	4.3E-02	<5E-7		1.6E-06	1.3E-03			3.7E-05
	37	7.7E-08	7.6E-02	<2E-7		2.0E-06	1.0E-06			2.6E-05
	37	5.4E-06	1.8E-02	<2E-6		5.0E-07	3.0E-04			2.8E-05
	38	1.1E-05	9.6E-02	<3E-7			1.1E-04			
	33	5.6E-06	6.0E-02	<2E-7		1.3E-05	9.3E-05			2.2E-04
	33	1.0E-05	6.9E-02	<2E-7		2.4E-05	1.4E-04			3.5E-04
	33	9.1E-06	5.4E-02	<1E-7		5.1E-06	1.7E-04			9.5E-05
	33	8.6E-06	4.9E-02	<1E-7		1.7E-06	1.8E-04			3.5E-05
	39	1.2E-07	8.2E-02	1.5E-06		4.7E-10	1.5E-06	1.8E-05		5.7E-09
	40	8.8E-05	7.2E-02	2.0E-06	2.0E-07		1.2E-03	2.8E-05	2.8E-06	
	41	1.1E-05	9.6E-02			<3E-6	1.1E-04			
	41	9.8E-04	8.4E-02				1.2E-02			

Table G-11. (continued)

COMPONENT	REF.	MEASURED ACTIVITY RATIOS (SCALING FACTORS)					ACTIVITIES TO Ni-63			
		ACTIVITIES TO Co-60					C-14	Nb-94	Tc-99	TRU
LPRM	29	2.8E-05	3.8E-01			2.7E-06	7.4E-05			7.1E-06
	28	1.7E-06	6.6E-02	<1E-7			2.6E-05			
	29	8.9E-07	5.2E-02	<1E-7			1.7E-05			
	31	3.4E-07	9.9E-02	<2E-6		1.2E-07	3.4E-06			1.2E-06
	30	8.4E-06	3.3E-02	<9E-8		6.8E-08	2.5E-04			2.1E-06
	30	4.5E-06	5.0E-02	<2E-7		1.7E-07	9.0E-05			3.4E-06
	31	7.3E-05	4.8E-02	<2E-6		2.7E-07	1.5E-03			5.6E-06
	30	3.3E-06	3.7E-02	<2E-7		1.7E-07	8.9E-05			4.6E-06
	30	4.8E-06	4.9E-02	<4E-6		5.4E-08	9.8E-05			1.1E-06
	31	9.0E-05	9.2E-02	<3E-7		1.1E-07	9.8E-04			1.2E-06
	32	1.2E-05	5.7E-02	<4E-7		1.2E-07	2.1E-04			2.1E-06
	32	8.8E-06	3.7E-02			1.1E-08	2.4E-04			3.0E-07
	34	1.7E-06	7.0E-03	<6E-8		1.5E-07	2.4E-04			2.1E-05
	34	1.7E-06	6.7E-03				2.5E-04			
	35	1.7E-05	8.8E-02			3.3E-08	1.9E-04			3.8E-07
	36	2.3E-05	5.5E-02	<4E-7		6.8E-08	4.2E-04			1.2E-06
	37	4.6E-05	4.7E-02	<6E-6		1.6E-07	9.8E-04			3.4E-06
	37	4.7E-07	4.6E-03	<3E-7		1.9E-07	1.0E-04			4.1E-05
	38		4.7E-02	<6E-7						
	41	1.2E-05	6.8E-02	<2E-7		<2E-6	1.8E-04			
	33	6.6E-06	5.2E-02	<9E-8		5.7E-07	1.3E-04			1.1E-05
	33	7.9E-07	5.9E-02	<6E-8		6.5E-07	1.3E-05			1.1E-05
	33	6.0E-06	9.1E-02	3.3E-07		1.2E-06	6.6E-05	3.6E-06		1.3E-05
	42	3.4E-06	3.8E-02	<2E-7		<3E-9	8.9E-05			
	42	4.5E-06	3.0E-02	<2E-7		2.4E-08	1.5E-04			8.0E-07
	43	2.2E-08	5.4E-02	<2E-6	<5E-8	3.8E-08	4.1E-07			7.0E-07
	43	8.9E-08	4.2E-02	<9E-7	<1E-7	7.1E-08	2.1E-06			1.7E-06
	40	5.6E-04	4.2E-01	4.6E-06	1.2E-06		1.3E-03	1.1E-05	2.9E-06	
	40	5.6E-04	4.2E-01	4.6E-06	1.2E-06		1.3E-03	1.1E-05	2.9E-06	
	40	5.6E-04	4.2E-01	4.6E-06	1.2E-06		1.3E-03	1.1E-05	2.9E-06	

Table G-11. (continued)

COMPONENT	REF.	MEASURED ACTIVITY RATIOS (SCALING FACTORS)								
		ACTIVITIES TO Co-60					ACTIVITIES TO Ni-63			
		C-14	Ni-63	Nb-94	Tc-99	TRU	C-14	Nb-94	Tc-99	TRU
IRM	36	2.2E-05	2.4E-01	9.3E-06	1.3E-07	1.4E-06	9.1E-05	3.9E-05	5.4E-07	5.8E-06
	42	6.6E-07	4.7E-02	<6E-8		8.8E-08	1.4E-05			1.9E-06
	39	2.5E-08	1.3E-01	<3E-6	<3E-8	4.9E-08	1.9E-07			3.8E-07
	39	3.7E-08	1.1E-01	<5E-7	2.3E-08	6.0E-08	3.3E-07		2.1E-07	5.4E-07
FUEL SUPPORT PC.	37	2.4E-05	9.6E-02	<4E-6			2.5E-04			
	33	2.3E-06	7.3E-02	<1E-6			3.2E-05			
FW SPARGER	33	<3E-7	2.0E-02	<3E-6						
	28	1.1E-05	1.1E-02	<6E-6			1.0E-03			
POISON CURTAIN	41	2.8E-06	1.8E-01	5.8E-06		<2E-6	1.5E-05	3.2E-05		
	37	8.8E-06	5.1E-02	<8E-7		4.9E-07	1.7E-04			9.6E-06
	37	4.6E-06	2.8E-02	<5E-7		5.0E-07	1.6E-04			1.8E-05
CRB GUIDE	28	1.5E-07	9.3E-02	<3E-6			1.6E-06			
UPPER & LOWER TIE PLATE, (PWR FUEL ASSEMBLY)	44	2.7E-06	1.4E-01	6.2E-06			2.0E-05	4.6E-05		
	25		4.4E-02	1.0E-07				2.3E-06		
	25		5.8E-02	9.7E-08				1.7E-06		
	25		1.3E-01	4.3E-07				3.4E-06		
	25		6.5E-02	1.9E-07				2.9E-06		
	25		6.3E-02	6.3E-08				1.0E-06		
	25		5.0E-02	3.7E-08				7.4E-07		
	16		2.1E-01	2.0E-07				9.5E-07		
	16		2.5E-01	1.1E-06				4.4E-06		
	16		1.9E-02	2.0E-07				1.1E-05		
	16		2.8E-02	1.3E-07				4.6E-06		
	17			3.5E-05						
MINIMUM:			4.1E-03				1.9E-07	7.4E-07	2.1E-07	5.7E-09
MAXIMUM:			4.2E-01				1.2E-02	2.2E-04	2.9E-06	3.5E-04
LOG-MEAN AVERAGE:			6.3E-02				7.5E-05	8.3E-06	1.4E-06	3.5E-06
LOG-MEAN RMS DISPERSIO			2.5E+00				10.0	3.9	3.1	8.4
									In nCi/gm:	4.7E-04

the crud layer on the component, deposited presumably (because of its similarity to crud deposits on other incore components) while the component was in the reactor pressure vessel. The TRU data represent only typical contamination levels and do not influence the component classification since the levels are usually well below any classification limits. Nonetheless, listing of this activity is required by 10 CFR 61.^{1,3} The last four columns relate four critical activities to ⁶³Ni. Scaling factors to the nickel isotope might be expected to show a smaller variation since nickel is a controlled major constituent of SS-304. Values at the bottom of the table give, respectively, the minimum and maximum values in each column and the log-mean average and RMS dispersion values. Averaging used a log-mean value because of the wide variance in the values and because measured scaling factors in utility-generated waste have demonstrated^{4,5} a log-normal distribution. The dispersion value is the RMS value of the ratios of the individual values to the log-mean average (i.e., a value of 2 means the individual data have a one-sigma spread of a factor of 2). The large values of dispersion for ¹⁴C and TRU are expected because of the origins of the activities and the lack of a physical reason for a generic correlation to cobalt.

Components of Zircaloy-4

Data for scaling factors for zircaloy-4 come from both PWR fuel skeleton assemblies and BWR fuel flow channels. In this material, both cobalt and nickel are trace elements, and the results were expected to show wide variances. Table G-12 presents the available measured scaling factor data for zircaloy-4. The table format is the same as that for presented for SS-304, Table G-11 above. Dispersion values are comparable to those in Table G-11, contrary to expectations.

Components of Inconel-718

Data for measured scaling factors in inconel-718 come entirely from measurements of spacer grids in Westinghouse fuel assembly skeletons. Table G-13 shows the data for this alloy in the same format as the previous tables. Inconel has high and presumably controlled nickel and niobium contents, and the analyses should show a stable value for the ⁹⁴Nb to ⁶³Ni scaling factor. However, the dispersion shown in the table for this factor is still large for unknown reasons. Uncertainties in the difficult radiochemical analysis for ⁹⁴Nb can account for only a small portion of the large spread.

Table G-12. Measured activity concentration ratios in activated components of zircaloy-4

COMPONENT	REF.	MEASURED ACTIVITY RATIOS (SCALING FACTORS)								
		ACTIVITIES TO Co-60					ACTIVITIES TO Ni-63			
		C-14	Ni-63	Nb-94	Tc-99	TRU	C-14	Nb-94	Tc-99	TRU
FUEL FLOW CHANNEL	28	1.0E-04	3.5E-02	<1E-5		<2E-4	2.9E-03			
	29	1.9E-05	5.6E-02	<3E-6		8.7E-05	3.4E-04			1.6E-03
	30	1.6E-05	1.1E-01	<3E-5			1.5E-04			
	45	5.0E-07	7.0E-03	3.0E-06		3.1E-05	7.1E-05	4.3E-04		4.4E-03
	45	1.8E-06	2.0E-02	3.5E-06		2.1E-04	9.0E-05	1.7E-04		1.1E-02
	45	1.5E-06	2.1E-02	3.6E-06		3.3E-04	7.1E-05	1.7E-04		1.6E-02
	34	1.2E-06	1.3E-02	<6E-8		2.8E-05	9.2E-05			2.2E-03
	38	2.9E-04	6.0E-02	<2E-4			4.8E-03			
GUIDE TUBES SPACER GRIDS (PWR FUEL ASSEMBLY)	44	1.5E-06	2.2E-01	<5E-6			6.8E-06			
	25		1.4E-03							
	25		1.7E-02	4.8E-05				2.8E-03		
	25		1.1E-03							
	25		1.6E-02							
	25		8.5E-03							
	25		1.3E-02	4.6E-05				3.5E-03		
	25		6.5E-03							
	25		6.7E-03	6.3E-05				9.4E-03		
	25		2.0E-02	3.3E-06				1.6E-04		
	25		9.9E-03							
	25		1.4E-03							
	25		1.3E-02							
	25		4.0E-02	1.9E-06				4.8E-05		
	25		2.3E-02							
	25		1.8E-02	2.0E-07				1.1E-05		
	17			1.8E-04						
	17			1.0E-04						
MINIMUM:			1.1E-03				6.8E-06	1.1E-05	N.A	1.6E-03
MAXIMUM:			2.2E-01				4.8E-03	9.4E-03	N.A	1.6E-02
LOG-MEAN AVERAGE:			1.2E-02				1.8E-04	3.6E-04	N.A	4.8E-03
LOG-MEAN RMS DISPERSIO			3.7E+00				7.4	8.8		2.7
									In nCi/gm:	7.3E-01

Table G-13. Measured activity concentration ratios in activated components of Inconel-718

COMPONENT	REF.	MEASURED ACTIVITY RATIOS (SCALING FACTORS)								
		ACTIVITIES TO Co-60					ACTIVITIES TO Ni-63			
		C-14	Ni-63	Nb-94	Tc-99	TRU	C-14	Nb-94	Tc-99	TRU
SPACER GRID (PWR FUEL ASSEMBLY)	44	<3E-02	3.2E-01	8.1E-05				2.6E-04		
	25		7.2E-01	2.0E-04				2.8E-04		
	25		2.9E-01	1.0E-05				3.4E-05		
	25		3.1E-01							
	25		2.2E-01							
	25		2.0E-01							
	25		2.0E-01							
	25		2.4E-01							
	25		1.2E+00	8.4E-05						
	25		2.6E-01	1.3E-04				5.0E-04		
	25		7.8E-01	4.9E-05				6.3E-05		
	17		1.6E-01	3.1E-04				1.9E-03		
	17		1.0E-01	4.0E-04				4.0E-03		
	17		1.2E-01	5.5E-04				4.6E-03		
	17		8.2E-02	2.8E-04				3.4E-03		
	17			8.0E-04						
	17			7.1E-03						
	17			1.4E-03						
	17			1.3E-03						
MINIMUM:			8.2E-02				N.A.	3.4E-05	N.A	N.A
MAXIMUM:			1.2E+00				N.A.	8.8E+00	N.A	N.A
LOG-MEAN AVERAGE:			2.6E-01				N.A.	6.0E-04	N.A	N.A
LOG-MEAN RMS DISPERSIO			2.1E+00				N.A.	6.2	N.A	N.A
									In nCi/gm:	N.A

APPENDIX B

COMPILATION OF MEASURED ELEMENTAL ANALYSIS DATA

APPENDIX B: COMPILATION OF MEASURED ELEMENTAL ANALYSIS DATA

Computational analyses of radioactivity require the concentrations of all elemental constituents of the material being considered. An experimental study⁶ published in 1984 reported both major and minor constituents of common structural materials used in nuclear reactors. The results were extremely useful in reporting measured concentrations of the trace constituents of the materials, especially the cobalt and niobium important in radioactive waste management and classification. Activation analyses done since that time have widely used the results of that study. Although the study in Reference 6 analyzed several materials, the list included neither zirconium-4 nor inconel-718, used in some reactor internals. Some recent analyses added to the data base of analyses of SS-304 and include measurements of the other two structural alloys.

Table G-14 provides a summary of some later measurements of SS-304, zircaloy-4, and inconel-718. The table includes only the elements considered important for waste characterization and some major constituents of the alloys. Absent from the table is iron, whose activation does not influence GTCC LLW classification. Samples for the analyses were, for the most part, from unactivated reactor components obtained from utility plants. Analyses of principal and trace elemental constituents used combinations of precise instrumental analytical techniques of thermal neutron activation analysis, energy dispersive X-ray fluorescence, and inductively coupled argon plasma atomic emission spectrometry. These techniques are capable of measuring concentrations for the desired analyses to part-per-billion levels. Column 2 in the table lists the reference for the reported values. The elemental abundances are in terms of percent by weight. The table gives minimum, maximum, average and standard deviation (RMS) values for each element where there were sufficient data to warrant such evaluations. Averages are linear and not weighted. Table G-14 lists values for zirconium and tin only because they are the principal constituents of zircaloy and because tin activates to produce ¹²⁵Sb, a large contributor to the dose rate from activated zircaloy-4. For SS-304, the results of this compilation differ only slightly from those in Reference 6. The values for inconel-718 show a surprising and misunderstood variation in the reported niobium concentrations.

Analyses used measured relative radionuclide activities (scaling factors) to determine elemental constituent abundance in alloys. Computing elemental abundances from scaling factors also requires assumptions for the relative neutron cross sections of the activation. Thus, comparisons of abundances computed from scaling factors to those from directly measured stable abundances will show the uncertainties in these cross sections as well. Nonetheless, compilation of elemental abundances from scaling factor data provides additional data for the knowledge base. The results are directly applicable to hardware activation since they result from actual activation of the material in power reactor cores. Table G-15 gives a summary of elemental constituents of SS-304, zirconium-4 and inconel-718 from an analysis of scaling factor data presented in Appendix A. Scaling factor values yield only relative values, and normalization must be used to obtain absolute abundances. Analyses used an assumed abundance for nickel for normalization, and the second column in the table lists these values. For SS-304 and inconel, the normalization value is the average abundance of nickel from Table G-14. Since no positive analyses for nickel in zirconium-4 were found, the value in the table resulted from analysis of data reported by McCarten and Luksic. McCarten⁷ gives results from the analysis of 36 inconel components of Prairie Island fuel skeletons in units of Ci/gm. Luksic⁸ presents activation activities for fuel skeletons in units of Ci/gm of the element irradiated. These latter values were from computations using ORIGEN2 for fuel burnup comparable to that for the Prairie Island fuel ($\approx 32,000$ MW·d/MTU).

Table G-14. Measured elemental constituents of reactor components

MATERIAL	REF	ABUNDANCE (WT %)					
		Ni	Co	Nb	Mo	Zr	Sn
SS-304	4,25	8.3	0.149	0.0033	0.23		
	4,25	9.1	0.150	0.0024	0.25		
	4,25	9.4	0.051	0.0300	0.37		
	4,25	8.0	0.128	0.0156	0.87		
	4,25	9.7	0.115		0.41		
	4,25	8.5	0.147		0.23		
	4,25	9.6	0.092				
	4,25	9.8	0.148				
	4,25	8.4	0.026	0.0010			
	4,25	8.3	0.025	0.0007			
	4,25	8.7	0.210	0.0370			
	4,25	8.8	0.210	0.0180			
	16	8.9	0.025	0.0250			
	16	9.0	0.028	0.0085			
	16	9.1	0.030	0.0110			
	16	8.8	0.028	0.0240			
	16	8.5	0.071	0.0091			
	16	8.9		0.0100			
	MAX:	9.8	0.210	0.0370	0.87		
	MIN:	8.0	0.025	0.0007	0.23		
	AVG:	8.9	0.096	0.0140	0.39		
	STD:	0.5	0.064	0.0110	0.22		
ZIRCALOY-4	4,25	<0.005	<0.0027		0.87	91.8	2.2
	4,25					89.0	2.3
	4,25					91.7	2.4
	4,25					92.0	2.6
	4,25					92.5	3.0
	4,25					115.0	3.9
	4,25					106.0	3.4
	4,25					73.5	2.0
	MAX:					115.0	3.9
	MIN:					73.5	2.0
	AVG:					93.9	2.7
	STD:					11.4	0.6

Table G-14. (continued)

MATERIAL	REF	ABUNDANCE (WT %)					
		Ni	Co	Nb	Mo	Zr	Sn
INCONEL-718	4,25	50.8	0.071	4.5	2.86		
	4,25	53.0	0.148	4.4	2.79		
	4,25	46.5	0.089	4.6	2.66		
	4,25	48.0	0.115	2.3	2.83		
	4,25	50.8	0.131	2.3	2.98		
	4,25	47.0	0.119		2.65		
	4,25	49.2	0.130		2.87		
	4,25	49.3	0.111		2.67		
	16	51.9	0.051	4.6			
	16	52.0	0.055	4.9			
	16	44.0	0.043	4.8			
	16	48.9	0.074	4.8			
	16	52.5	0.035	5.1			
	16	49.9	0.052	4.9			
	16	52.0		5.6			
	MAX:	53.0	0.148	5.6	2.98		
	MIN:	44.0	0.035	2.3	2.65		
	AVG:	49.7	0.087	4.4	2.79		
	STD:	2.5	0.036	1.0	0.11		

Table G-15. Elemental composition analysis from measured nuclide scaling-factor data

<u>MATERIAL</u>	<u>WEIGHT % NICKEL</u>	<u>SCALING FACTOR</u>	<u>a (Gm/Gm)</u>	<u>WEIGHT % COBALT</u>	<u>SCALING FACTOR</u>	<u>(Gm/Gm)</u>	<u>WEIGHT % NIOBIUM</u>	<u>SCALING FACTOR</u>	<u>(Gm/Gm)</u>	<u>WEIGHT % TC-MO</u>
STAINLESS STEEL-304	8.9	0.063 b	1.3×10^{-2}	0.12	8.2×10^{-6}	3.1×10^{-4}	0.003	1.4×10^{-6}	1.7×10^{-2}	0.15
ZIRCALOY -4	2.1×10^{-4} c	0.019	6.9×10^{-2}	1.4×10^{-3}	1.1×10^{-5}	4.1×10^{-4}	8.4×10^{-6}			
INCONEL -718	49.7	0.26	3.2×10^{-3}	0.16	6.0×10^{-4}	2.2×10^{-2}	1.11			

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a. Values of metal activation are from ORIGEN2 computations reported in reference 4 in units of Ci produced per gram of the element: ^{63}Ni : 7.42×10^{-2} , ^{60}Co : 89.6, ^{94}Nb : 1.99×10^{-3} , ^{99}Tc : 5.99×10^{-6} . These values will change with neutron fluent (i.e., reactor power level and position in vessel); however, since only ratios of the values are used, their accuracies should be adequate.

b. The equation for calculating the gram per gram ratios is:

$$\frac{\left(\frac{Ci}{CC}\right)_A}{\left(\frac{Ci}{CC}\right)_B} \times \frac{\left(\frac{Ci}{Gm}\right)_B}{\left(\frac{Ci}{Gm}\right)_A} = \left(\frac{Gm_A}{Gm_B}\right)$$

where the first term has the scaling factors and the second term has the values from #1.

c. The normalizing value results from analyses of concentration data in Reference 7 and activation values from Reference 8 as discussed in the text.

The resulting value for nickel concentration is a factor of 10 below the detection limit reported for the single reported attempt to measure elemental nickel, shown in Table G-14. Columns 3-5 in the table give values for cobalt. Column 3 shows the log-mean average of the ^{60}Co scaling factor (relative to ^{63}Ni) from Appendix A. Column 5 gives the value for grams of cobalt per gram of nickel, computed using the equation in footnote b. The first ratio in the equation is the scaling factor, and the second ratio contains the activation factors listed in the footnote and extracted from Reference 8. Column 5 gives the abundance of cobalt obtained from multiplying the value in column 4 by the normalizing abundance of nickel in column 2. These analyses are repeated for niobium and molybdenum in the last six columns. The results are comparable to those in Table G-14 except for niobium, where the values are about a factor of four lower in Table G-15, not as surprising for SS-304, where the abundance is low, as for inconel, where analytical precision for both elemental and scaling factor analyses should be relatively high. No explanation is known except for possible errors in the assumed activation cross sections for niobium, both thermal and resonance, or systematic errors in measured niobium scaling factors.

APPENDIX C

MEASURED ^{60}Co ACTIVITIES IN COMPONENTS-- DOSE RATE DATA

APPENDIX C: MEASURED ^{60}Co ACTIVITIES IN COMPONENTS-- DOSE RATE DATA

For most bulk radioactive waste, and particularly for irradiated hardware, it is impractical from both radiation exposure and economic considerations to analyze completely an entire component for radioactive content. Furthermore, most wastes and irradiated hardware have radioactivity concentrations and sometimes materials that are inhomogeneous over the entire component. Thus, extrapolating and scaling the analyses of small samples to the entire volume is usually inaccurate and impractical. A more practical and most often used method of classifying bulk radioactive waste is a three-step process: (a) measure the average gamma ray dose rate from the component or package, (b) apply a predetermined dose to curie factor to convert the average dose rate to average concentration of ^{60}Co , and (c) apply a set of scaling factors to estimate the concentration of the other activities from the ^{60}Co activity. Scaling factors are discussed in Appendix A. Dose to curie conversion factors are both measured⁹ and computed,¹⁰ using known or estimated relative concentrations of gamma ray-emitting nuclides. Determination of average dose rate results from averaging values from a series of measurements at uniform intervals over the volume of the component.

Control Rod Blades

Table G-16 lists typical dose rate profile data along the lengths of BWR control rod blades. The table shows data from nine typical blades measured while they were in a utility fuel pool using an Eberline model R07 dose rate meter contained and calibrated in an immersable housing. The distance along the blade from the top end is as shown in the first column (the top is defined as the hot end containing the stellite rollers). In many of the rods, the dose rate at zero meters is highly influenced by the stellite (a cobalt alloy used by GE for bearings), and its effect must be removed before averaging. The reference for each set of data is shown at the top of the columns, and the maximum, minimum, and linear averages are shown at the bottoms of the columns. The more active rods were in the reactor core for a fuel burnup equivalency of about 35,000 MW·d/MTU, whereas the lower activity rods were removed prematurely for various reasons. This spread is typical for blades in utility fuel pools, and the low concentration rods are mixed with high activity rods in packages for burial to achieve lower averages over the contents.

Table G-17 lists the average dose rates for 45 scanned CRBs, together with the reference for the data and the utility rod identification where available. The bottom of the dose rate column gives the minimum, maximum, and linear average values for the compilation. The dose to curie factor for ^{60}Co (Reference 9), resulting from an average of measured values, is given in column 5. The last column lists the average concentration of ^{60}Co for each entry, resulting from multiplying the average dose rate by the dose to curie factor.

Table G-16. Typical dose profile data, control rod blades

DISTANCE (Meters)	DATA REFERENCE REPORT								
	29	29	32	32	32	31	31	37	37
	DOSE RATE (kR/Hr)								
0.0	14	15.8	20	22	9.8	22	21	13.8	16.5
0.3	7.4	14.9	7.3	20	0.5	21	20	6.6	12.0
0.6	7.4	12.4	5.3	18.8	0.35	19.7	18.1	5.7	12.1
0.9	5.9	11.3	0.7	14.6	0.25	20	20	5.5	12.0
1.2	5.4	10.9	0.7	12	0.13	20	19.8	5.6	11.4
1.5	5.1	9.9	0.3	10.9	0.06	20	19.5	0.2	10.3
1.8	0.9	9.8		9.3	0.01	20	18.7	3.3	9.5
2.1	0.8	9.6	3.1	8.6	0.005	20	17.8	1.9	8.6
2.4	3.9	10	2.3	7.5		20	16.2	0.7	7.5
2.7	3.3	5.5	1.5	5.9		18.6	13	0.2	5.7
3.0	1.1	3.1	0.9	0.1		15.7	8.1	0.06	2.9
3.3	0.15	1.05	0.4	2.7		10.8	3.9	0.03	1.3
3.6	0.01	0.06	0.15	1.5		4.9	1.6	0.01	0.27
3.9		0.01	0.04	0.5		1.8	0.31		
4.2						0.13	0.04		
MAX:	14.0	15.8	20.0	22.0	9.8	22.0	21.0	13.8	16.5
MIN:	0.01	0.01	0.04	0.1	0.005	0.13	0.04	0.01	0.27
AVG:	4.3	8.2	3.3	9.6	1.4	15.6	13.2	3.4	8.5

Table G-17. Average dose rate and Co-60 concentration data

COMP.	DATA REFERENCE	IDENT.	AVERAGE CONTACT DOSE RATE (kR/Hr)	DOSE-TO CURIE FACTOR (Co-60)	AVERAGE CONCENTRATION OF Co-60 (Ci/M-3)
VOLUME: .0266 M-3	46	6	7.43	3.3	24.5
	46	12	9.05	+1.3	29.9
	46	21	10.31	-0.9	34.0
	46	26	7.12		23.5
	46	32	7.43		24.5
	46	40	7.01		23.1
	46	44	6.50		21.5
	46	61	8.26		27.3
	46	71	8.66		28.6
	46	75	13.84		45.7
	46	77	7.45		24.6
	46	84	5.91		19.5
	46	87	5.63		18.6
	46	98	10.29		34.0
	46	109	9.92		32.7
	46	C3	8.25		27.2
	46	G6	11.91		39.3
	46	H9	6.64		21.9
	46	63	5.55		18.3
	46	91	4.75		15.7
	46	122	4.78		15.8
	46	124	3.69		12.2
	46	128	5.38		17.8
	46	52	6.39		21.1
	46	17	4.43		14.6
	46	106	8.60		28.4
	46	130	4.75		15.7
	46	504	12.63		41.7
	46	81	10.81		35.7
	46	316	12.80		42.2
	46	152	12.02		39.7
	46	491	12.48		41.2
	46	101	5.91		19.5
	46	107	5.77		19.0
	46	41	11.30		37.3
	46	126	8.68		28.6
	46	16	11.86		39.1
	31		15.60		51.5
	31		13.20		43.6
	37		3.40		11.2
	37		8.50		28.0
	29		4.30		14.2
	29		8.20		27.1
	32		3.30		10.9
	32		9.60		31.7
		MIN:	3.30		10.9
		MAX:	15.60		51.5
		AVG:	8.23		27.2

Incore Instrumentation

Table G-18 shows typical dose profile data for six local power range monitors (LPRM) and three intermediate range monitors (IRM). For these components, also, the higher activity LPRMs received a total neutron fluence equivalent to a fuel burnup of about 35,000 MW·d/MTU, whereas the others were in for shorter periods for various reasons. The range of activities is typical. The IRMs are designed to remain in the core for the reactor life and receive proportionally larger doses. The three listed in the table were removed prematurely for some reason but, nonetheless, show considerably more activity than the LPRMs. This, too, is typical. Both LPRMs and IRMs contain fission chambers having milligram quantities of fast and thermal fissioning isotopes of uranium; however, their fission product and fissile material contents, averaged over the volume of metal in the instrument, are well below Class C limits and do not influence the classification of the component.

Table G-19 summarizes average dose rate data from 17 LPRMs, 9 IRMs, and 2 SRMs, representing typical activities of these components currently in fuel pools. The references for the data and the utility identifications are also listed. Three of the LPRMs were identified by manufacturer (Reuter Stokes, General Electric and Westinghouse). No significance should be given to their different average dose rates other than the fact that they each received a different total neutron fluence. The table also includes the measured dose to curie (DTC) factor for ^{60}Co (Reference 9) for these three components (used for all three since the geometries are the same for each type of instrument). The last column gives the average concentration of ^{60}Co from application of the DTC factor to the average dose rate. The dose rate and ^{60}Co columns are averaged at the bottom of the columns.

Fuel Flow Channels

Table G-20 shows dose profile data from six typical BWR fuel flow channels. The channels typically are used with one fuel assembly during its life in the core ($\approx 35,000$ MW·d/MTU). Because of the low cobalt content in the zircaloy channel, the average dose rate is low. Further, the high tin content of zircaloy-4 causes over 50% of the dose rate to result from its activation to ^{125}Sb in fresh channels. This ratio that changes with time because of the different half lives of ^{60}Co and ^{125}Sb must be taken into account when using dose rate to classify components of zircaloy-4.

Table G-21 summarizes the available average dose rate data from fuel flow channels and from three other BWR components, poison curtains, feedwater sparger nozzles, and fuel support pieces. The table includes dose to curie factors (Reference 9) for ^{60}Co , average ^{60}Co concentrations for each component, and averages for each type of component.

Table G-18. Typical dose profile data, incore instruments

DISTANCE (Meters)	DATA REFERENCE REPORT								
	28	29	29	32	31	30	34 (IRM)	42 (IRM)	37 (IRM)
	DOSE RATE (kR/Hr)								
0.0	0.7	0.4	0.0	2.5	1.7	2.2	21.0	2.6	3.0
0.3	1.5	1.7	1.2	4.0	2.5	15.2	21.0	16.2	8.3
0.6	1.7	3.1	3.2	4.9	3.0	17.0	21.0	23.5	12.3
0.9	1.8	3.0	4.0	6.6	9.8	19.0	21.0	24.0	13.2
1.2	1.8	3.1	4.5	6.7	11.8	19.5	21.0	25.0	12.9
1.5	1.8	2.8	4.4	6.8	13.3	20.0	21.0	22.3	14.0
1.8	1.4	2.8	4.0	6.8	13.4	19.0	21.0	23.6	13.3
2.1	1.8	2.6	3.8	6.7	13.5	19.0	21.0	23.0	13.9
2.4	1.4	2.5	3.4	7.9	14.0	18.3	21.0	22.5	13.0
2.7	1.3	2.1	2.8	7.1	13.0	17.0	21.0	21.7	14.5
3.0	1.2	2.0	2.0	6.7	12.0	16.6	16.5	18.1	14.8
3.3	1.5	1.8	0.8	6.7	11.8	15.8	3.8	18.1	15.2
3.6	0.4	0.9	0.3	6.8	11.5	11.6	1.0	5.0	15.2
3.9				4.3	10.0	6.0		1.0	15.8
4.2				3.5	7.0	1.0			13.3
4.5				2.7	3.0				8.6
4.8				1.1	1.0				1.6
5.1				0.07					0.1
MAX:	1.8	3.1	4.5	7.9	14.0	20.0	21.0	25.0	15.8
MIN:	0.4	0.4	0.0	0.1	1.0	1.0	1.0	1.0	0.1
AVG:	1.4	2.2	2.6	5.1	9.0	14.5	17.8	17.6	11.3

Table G-19. Average dose rate and Co-60 concentration for incore instruments,
measured dose rates and scaling factors

COMP.	DATA REFERENCE	IDENT.	AVERAGE CONTACT DOSE RATE (kR/Hr)	DOSE-TO CURIE FACTOR (Co-60)	AVERAGE CONCENTRATION OF Co-60 (Ci/M-3)
LPRM VOLUME 2E-4 M-3	42		14.00	55.1	771.4
	34		19.80	+11.1	1091.0
	29		2.20	-9.2	121.2
	29		3.00		165.3
	28		2.30		126.7
	28		1.09		60.1
	32		10.30		567.5
	32		6.00		330.6
	33		15.20		837.5
	30		7.40		407.7
	31		10.40		573.0
	31	REUTER-ST	6.80		374.7
	31	GE	15.20		837.5
	31	WEST	4.50		248.0
	31	40-25	6.15		338.9
	31		11.80		650.2
	31		8.90		490.4
MIN:			1.09		60.1
MAX:			19.80		1091.0
AVG:			8.53		470.1
IRM	36	36-13	12.92		711.9
	36	20-29	11.26		620.4
	36	12-13	9.29		511.9
	36	20-21	11.79		649.6
	36	28-29	9.89		544.9
	36	28-21	12.01		661.8
	36	36-45	12.61		694.8
	36	12-45	8.06		444.1
	42		20.60		1135.1
MIN:			8.06		444.1
MAX:			20.6		1135.1
AVG:			12.05		663.8
SRM	45	12-29	11.63		640.8
	45	20-13	10.72		590.7
AVG:			11.18		615.7

Table G-20. Typical dose profile data, fuel flow channels

DISTANCE (Meters)	DATA REFERENCE REPORT					
	28	29	45	45	45	34
	DOSE RATE (kR/Hr)					
0.0	0.010	0.020	0.056	0.660	0.470	0.020
0.3	0.023	0.021	0.087	0.280	0.260	0.150
0.6	0.020	0.063	0.130	0.340	0.270	0.210
0.9	0.025	0.090	0.176	0.360	0.420	0.170
1.2	0.025	0.106	0.173	0.430	0.440	0.140
1.5	0.025	0.109	0.170	0.440	0.430	0.120
1.8	0.025	0.114	0.178	0.440	0.400	0.110
2.1	0.025	0.112	0.175	0.430	0.430	0.100
2.4	0.025	0.113	0.168	0.430	0.440	0.100
2.7	0.025	0.104	0.165	0.430	0.420	0.070
3.0	0.025	0.107	0.156	0.450	0.390	0.040
3.3	0.025	0.098	0.144	0.420	0.370	0.040
3.6	0.025	0.085	0.109	0.360	0.350	
3.9	0.019	0.053	0.035	0.300	0.300	
4.2		0.021	0.028	0.230	0.220	
MAX:	0.025	0.114	0.178	0.660	0.470	0.210
MIN:	0.010	0.020	0.028	0.230	0.220	0.020
AVG:	0.023	0.081	0.130	0.400	0.374	0.106

Table G-21. Average dose rate and Co-60 concentration for assorted components, measured dose rates and measured scaling factors

	DATA REFERENCE	IDENT.	AVERAGE CONTACT DOSE RATE (kR/Hr)	DOSE-TO CURIE FACTOR (Co-60)	AVERAGE CONCENTRATION OF Co-60 (Ci/M-3)
POISON	41		1.44	40.0	57.6
CURTAINS	36		1.86	+14.9	74.4
VOL:2.1E-2	36		0.99	-14.9	39.6
SPARGER	33		0.01	6.70	0.07
NOZZLE					
VOL: 7.25E-2	28		0.06		0.40
FUEL-SPT PC	33		0.26	6.70	1.76
VOLUME:					
3.5E-3					
FUEL FLOW	28		0.023	4.3	0.10
CHANNELS	29		0.092	+6.5	0.39
INCONEL	32	DN-814	0.136	-2.9	0.59
DENS:6.56	32	DN-764	0.097		0.42
VOLUME:	32	DN-767	0.079		0.34
5.3E-3	34	DN-769	0.789		3.39
	1		0.113		0.49
MIN:			0.023		0.10
MAX:			0.789		3.39
AVG:			0.190		0.82

PWR Fuel Assemblies

Figure G-4 shows pictorially a dose rate profile of a typical Westinghouse fuel assembly skeleton. The Westinghouse skeletons have inconel grid spacers and zircaloy tubes, and the dose rate profile reflects these different materials, with the spacers averaging nearly 4000 R/hr and the guide tubes averaging less than 300 R/hr. The figure shows data from a typical full burnup Westinghouse fuel assembly skeleton. Skeletons from the other manufacturers do not show the high dose rates at the grid spacers.

SKELETON DOSE-RATE PROFILE

Assembly I.D.: C-73

Location: S. S.-11

Date: 3/16/88

Days Out of Core
As of 3/16/88: 3409

Recorder:

Burnup: 38,654 MWD/MTU

Figure G-4. Typical skeleton dose rate profile.

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