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7. Abstract <i>RMB 7/13/94</i> This document compiles technical data on irradiated fuel stored at the Hanford Site in support of the Hanford SNF Management Environmental Impact Statment. Fuel included is from the Defense Production Reactors (N Reactor and the single-pass reactors; B, C, D, DR, F, H, KE and KW), the Hanford Fast Flux Test Facility Reactor, the Shippingport Pressurized Water Reactor, and small amounts of miscellaneous fuel from several commercial, research, and experimental reactors.		10. RELEASE STAMP <div style="border: 1px solid black; padding: 5px; text-align: center;">OFFICIAL RELEASE DATE JUL 13 1994 <i>STA 4</i></div>
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HANFORD SPENT FUEL INVENTORY BASELINE

K. H. BERGSMAN

SPENT FUEL MANAGEMENT PROJECT, 2C200

JUNE 1994

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1.0 INTRODUCTION

Various types of irradiated fuel are currently stored on the U.S. Department of Energy's (DOE) Hanford Site (Hanford), which is shown in Figure 1.1. Most fuel was irradiated at Hanford to produce plutonium for national defense, although some was irradiated for research purposes. A small amount of fuel was irradiated off-site and shipped to Hanford for storage and/or processing.

The DOE has determined that irradiated fuel is not needed for defense purposes at the present time and that all viable alternatives should be evaluated for interim storage of the irradiated fuel at the Hanford Site. A new Environmental Impact Statement (EIS) will be prepared to evaluate the storage alternatives. The purpose of this document is to provide technical data on the irradiated fuel that is within the scope of the EIS. These data will facilitate development and analysis of the EIS storage alternatives.

The fuel included in the scope of the Hanford SNF Management EIS is from the Defense Production Reactors (N Reactor and the single pass production reactors; B, C, D, DR, F, H, KE, and KW), the Hanford Fast Flux Test Facility (FFTF) Reactor, the Division of Naval Operations Shippingport Pressurized Water Reactor (PWR), and small amounts of fuel from several commercial and experimental reactors. Fuel from the Washington Public Power Supply System Reactor located at Hanford is not included in the EIS scope. The approximate quantities of the irradiated fuel included are:

<u>Fuel Type</u>	<u>Quantity</u> MTHM	<u>Storage Location</u>
N Reactor	2100 ¹	105 K-Basins and PUREX ²
Shippingport PWR Core II	15.8	200W Area T-Plant
FFTF Reactor	11.0	400 Area
Single Pass Reactors	3.4	PUREX and 105-K Basins ³
Miscellaneous	2.6	300 Area and 200W Area

1. MTHM stands for metric tons of heavy metal (uranium plus plutonium).
2. There are 0.3 MTHM of N Reactor fuel in PUREX canyons cells.
3. There are 0.5 MTHM of single pass reactor fuel in the 105-K Basins.

In addition to the fuel identified in this document, there are legacy fuel materials at, or suspected to be at, various locations on the Hanford Site, including the 100, 200, and 600 Areas. Some or all of these materials may meet the definition of low level or Transuranic (TRU) waste as defined in DOE Order 5820.2A. As the legacy fuel materials become better identified and characterized, materials that are identified as spent nuclear fuel (SNF) will be included in revisions to this document.

The characteristics of the fuel described in this document (quantity, radioactivity, etc.), reflect the actual inventories at Hanford in its post-production period. Previously released documents with different purposes,

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e.g. support for extended irradiated fuel reprocessing missions, may present different fuel characteristics. When differences exist, the characteristics described herein should be used for the evaluation of alternatives for the interim storage (up to 50 years) of irradiated fuel.

2.0 105-N REACTOR FUEL

2.1 BACKGROUND (WHC 1984a, PNL 1991a)

The 105-N Reactor (N Reactor) is a graphite-moderated, pressurized-water cooled reactor located in the 100-N Area of the Hanford Site. It was initially designed for plutonium production for national defense. Initial operation occurred in 1963. Two years later, N Reactor was modified to produce steam to be used by the Washington Public Power Supply System to generate electricity. N Reactor was the only dual-purpose reactor in the United States.

The core of N Reactor was an 1800-ton graphite block, 33 feet (10 meters) high by 33 feet (10 meters) wide by 39 feet (12 meters) long. A total of 1003 horizontal Zircaloy-2 process tubes held the fuel and contained the cooling water. The cooling water transferred the reaction heat from the 366 metric tons of uranium fuel to the secondary coolant water in steam generators. Perpendicular to the process tubes were 84 horizontal water-cooled, boron containing control rods (WHC 1988b). These rods entered the reactor from both sides and provided operating reactivity control, neutron flux shaping, and emergency shutdown control. A completely independent backup emergency shutdown control was provided by 107 vertical channels penetrating the core that could be gravity filled with special neutron absorbing balls.

When operating, N Reactor produced up to 4000 megawatts of heat and up to 13 million pounds per hour of low pressure steam, which produced 860 megawatts of electricity. The production of tritium and various non-defense target elements was also demonstrated.

N Reactor ceased operation in 1987, but fuel remained in the core pending a possible restart. The final core was discharged in April 1989.

2.2 FUEL DESCRIPTION (WHC 1984a, WHC 1989)

The N Reactor fuel elements consist of two concentric tubes made of uranium metal co-extruded into Zircaloy-2 cladding. There are two basic types of fuel elements differentiated by their uranium enrichment. Mark IV fuel elements have a pre-irradiation enrichment of 0.947% ^{235}U in both tubes and an average uranium weight of 50 pounds (22.7 kilograms). They have an outside diameter of 2.42 inches (6.1 centimeters) and a length of 17.4, 23.2, 24.6, or 26.1 inches (44, 59, 62, or 66 centimeters). Mark IA fuel elements have a pre-irradiation enrichment of 1.25% ^{235}U in the outer tube and 0.947% ^{235}U in the inner tube. They have an average uranium weight of 35.9 pounds (16.3 kilograms). Mark IA fuel elements have an outside diameter of 2.40 inches (6.1 centimeters) and a length of 14.9, 19.6 or 20.9 inches (38, 50, or 53 centimeters). There is also a small amount of fuel with 0.71%

^{235}U content; which was designated as Mark IVB fuel. The Mark IVB fuel has the same dimensions and weights as the 26.1 inch (66 centimeters) Mark IV fuel. A Mark IV fuel element is shown in Figure 2.1. Mark IA fuel elements are similar.

Information on the physical characteristics of the fuel is given in Table 2.1 and on drawing H-1-39775. The chemical compositions of fuel element components are shown in Table 2.2 (UNC 1979).

2.3 FUEL FABRICATION (WHC 1984a, WHC 1989)

The choice of materials and fabrication processes used for N Reactor fuel were affected by the requirement to produce steam for efficient generation of electrical power as well as the need to efficiently produce plutonium. Electricity generation required reactor operation at high temperature and pressure. However at high temperatures, metallic uranium can undergo physical changes that are detrimental to reactor operation. Consequently, materials and fabrication processes were selected that met the needs of plutonium production without significantly reducing steam production.

In the co-extrusion fabrication process, each uranium tube was encased in an inner and outer sleeve of Zircaloy-2 with an outer sheath of copper. This assembly was evacuated and sealed to prevent oxidation during preheating and extrusion. As the tube assembly was extruded under high pressure and elevated temperature, a solid state diffusion bond formed between the uranium core and the Zircaloy-2 cladding.

The extruded fuel was cut to the desired lengths and a recess was machined into the uranium at each end of the fuel section. This was followed by an acid stripping of the copper sheath and an acid etching process to remove residual uranium from the cladding. The tube ends were closed by placing a braze ring, made of Zircaloy-2 and 5% beryllium, and a Zircaloy-2 end cap in the recess in each end of the fuel and induction heating the assembly to brazing temperature (approximately 1050°C) in a vacuum. The junction of the end cap, braze, and cladding was fusion welded to alloy the braze material with the cladding to improve corrosion resistance and provide a hermetic seal. The fuel was then heat treated and cleaned with an abrasive grit blast and acid baths.

Six spacers were welded to the outside of the inner tube to ensure proper alignment and locking with the outer tube when the two tubes were assembled together. Eight Zircaloy-2 support clips were welded to the outside of the outer tube to ensure proper alignment of the fuel when inserted into the N Reactor core. Low-carbon steel "shoes" were crimped onto these Zircaloy-2 clips to reduce the effect of their rubbing against the reactor tubes during fuel element insertion and removal. The shoes vary in thickness from 10 mils to 18 mils and contain 0.42 to 0.74 grams of iron. Finishing operations on each tube included nondestructive testing, heating in an autoclave, inspecting, and inserting an inner tube into an outer tube to produce the finished fuel element.

Table 2.1: 105-N Reactor Fuel Element Description

	Mark IV				Mark IA		
Pre-irradiation enrichment of ^{235}U	0.947% Enriched				1.25-0.947% Enriched		
Type-Length code ¹	E	S	A	C	M	T	F
Length (in)	26.1	24.6	23.2	17.4	20.9	19.6	14.9
Element diameter (in)							
1. Outer of outer	2.42				2.40		
2. Inner of outer	1.70				1.77		
3. Outer of inner	1.28				1.25		
4. Inner of inner	0.48				0.44		
Cladding weight (lb)							
1. Outer element	2.41	2.29	2.18	1.74	1.94	1.83	1.45
2. Inner element	1.21	1.15	1.10	0.88	1.18	1.12	0.89
Weight of uranium in outer (lb)							
1. (0.947% ^{235}U)	35.2	33.1	31.2	23.1			
2. (1.25% ^{235}U)					24.4	22.9	17.3
Weight of uranium in inner (lb) 0.947% ^{235}U	16.5	15.5	14.6	10.9	12.1	11.3	8.6
Weighted average of uranium in element (lb)	50.0				35.9		
Ratio of Zircaloy-2 to uranium (lb/ton)	140	141.6	143.2	154.1	171.0	172.5	180.7
Weighted ave. (lb/ton)	140.6				171.4		
% of total elements	63				37		
% of length type of each fuel	78	10	7	5	87	10	3
Displacement volume (gal/ton uranium)	16				16		

1. Letter code differentiates the different lengths of the Mark IV or Mark IA fuel elements, i.e. a type "E" element is 26.1 inches long.

Table 2.2: Chemical Composition of 105-N Reactor Fuel Elements¹

Element	Uranium Alloy 601	Zircaloy-2	Braze Filler
Aluminum	700-900	75	145
Beryllium	10	-	4.75-5.25 wt%
Boron	0.25	0.5	0.5
Cadmium	0.25	0.5	0.5
Carbon	365-735	275	500
Chromium	65	0.05-0.15 wt%	0.05-0.15 wt%
Cobalt	-	10	20
Copper	75	50	60
Hafnium	-	200	200
Hydrogen	2.0	25	50
Iron	300-400	0.07-0.20 wt%	0.06-0.21 wt%
Lead	-	100	130
Magnesium	25	20	60
Manganese	25	50	60
Molybdenum	-	50	50
Nickel	100	0.03-0.08 wt%	0.03-0.08 wt%
Nitrogen	75	80	200
Oxygen	-	-	2300
Silicon	124	100	250
Sodium	-	20	20
Tin	-	1.20-1.70 wt%	1.14-1.70 wt%
Titanium	-	50	50
Tungsten	-	50	100
Uranium	Balance	3.5	4
Vanadium	-	50	50
Zirconium	65	Balance	Balance

1. Concentrations given in parts per million (ppm) maximum or ppm range, unless indicated otherwise.

2.4 FUEL INVENTORY

Most N Reactor fuel was irradiated to produce plutonium with an average ^{240}Pu isotope content of either 6% or 12%, although small amounts of plutonium with a nominal 4%, 9%, or 16% ^{240}Pu isotope content were also produced. Fuel with plutonium having approximately 6% ^{240}Pu isotope content, or less, is considered weapon grade material; fuel with plutonium having higher ^{240}Pu isotope content is considered fuel grade.

The total inventory of irradiated N Reactor fuel at Hanford is approximately 2100 MTHM, based on nuclear material safeguards records. The inventory contains 1768.1 MTHM of fuel grade fuel and 331.4 MTHM of weapon grade fuel (WHC 1992a, WHC 1992b, WHC 1994a). There is also 0.3 MTHM of fuel with uncertain ^{240}Pu isotope content, which is discussed further in Section 2.8. The fuel grade fuel was discharged from N Reactor between 1970 and 1980; the weapon grade fuel was discharged between 1986 and 1989, although reactor operation ceased in 1987. The fuel grade material includes approximately 45 MTHM with plutonium having a nominal 9% ^{240}Pu isotope content and approximately 18 MTHM with plutonium having a nominal 16% ^{240}Pu content. The weapon grade fuel includes approximately 200 MTHM of fuel from the final N Reactor core discharge, having plutonium with an average ^{240}Pu content of about 4% (WHC 1988a). A breakdown of the irradiated N Reactor fuel inventory is shown in Table 2.3, based on nuclear material safeguards records (WHC 1992a, WHC 1992b, WHC 1994a).

2.5 FUEL EXPOSURE AND RADIONUCLIDE CONTENT

The average fuel exposure is shown in Table 2.4 for Mark IV and Mark IA fuel having 6% and 12% ^{240}Pu in its plutonium (RHO 1985a). Also shown is the range of time since irradiation for Mark IV and Mark IA weapon grade and fuel grade fuel.

Table 2.3: 105-N Reactor Fuel Inventory

Fuel Location and Type	Uranium	Plutonium	Total
	MTHM	MTHM	MTHM
KE Basin			
Weapon grade, Mark 1A	0	0	0.00
Mark IV	291.3	0.20	291.50
Fuel Grade, Mark 1A	0	0.01	0.01
Mark IV	852.4	1.94	854.34
Total KE Basin	1,143.70	2.15	1,145.85
KW Basin			0.00
Weapon grade, Mark 1A	39.3	0.02	39.32
Mark IV	0.6	0	0.60
Fuel grade, Mark 1A	588.0	1.21	589.21
Mark IV	323.9	0.64	324.54
Total KW Basin	951.80	1.87	953.67
Total N Reactor Fuel			
Weapon grade, Mark 1A	39.30	0.02	39.32
Mark IV	291.90	0.20	292.10
Fuel grade, Mark 1A	588.00	1.22	589.22
Mark IV	1,176.30	2.58	1,178.88
PUREX	.3	0	0.30
All N Reactor Fuel	2,095.80	4.02	2,099.80

Table 2.4: Exposure and Time Since Irradiation of 105-N Reactor Fuel

	<u>Fuel</u>	<u>Mark IV</u>	<u>Mark IA</u>
Exposure:	12% ^{240}Pu 6% ^{240}Pu	2268 MWD/MTU ¹ 907 MWD/MTU	2720 MWD/MTU 1089 MWD/MTU
Decay Time ² :	fuel grade ³ weapon grade ³	13-23 yrs 7-8 yrs	13-23 yrs 7-8 yrs

1. MWD/MTU stands for megawatt days per metric ton of initial uranium.
2. Time since last irradiation as of 12/93.
3. Weapon-grade fuel contains plutonium having approximately 6% ^{240}Pu isotope content, or less; fuel-grade fuel contains plutonium having higher ^{240}Pu isotope content.

The concentration of radionuclides present in the N Reactor fuel is a function of the original fuel composition, irradiation history, and decay time. Short lived fission products have decayed to insignificant concentrations during the time since the fuel was irradiated. The radionuclide activity and decay heat present in N Reactor fuel were determined by ORIGEN2 analysis (WHC 1991a, WHC 1992c). The activity of radionuclides in Mark IV and Mark IA fuel with 12% ^{240}Pu content in the plutonium, which exceeds five curies per metric ton of initial uranium, is shown in Table 2.5 at 10, 20, and 30 years since irradiation. A more complete listing of activity, including longer decay periods is presented in (WHC 1991a). The activity of radionuclides in Mark IV and Mark IA fuel with 6% ^{240}Pu content in the plutonium, which exceeds five curies per metric ton of initial uranium, is shown in Table 2.6 at 5 and 10 years since irradiation. A more complete listing is presented in (WHC 1992c). Activity for longer, practical decay periods has not been determined. However, the activity of the fuel with 6% ^{240}Pu content in the plutonium is not bounding, as it can be seen to be less than that of the fuel with 12% ^{240}Pu content in the plutonium on any desired date, i.e. the 10 and 20 year decayed fuel with 12% ^{240}Pu content in the plutonium have greater activity than fuel with 6% ^{240}Pu content in the plutonium decayed for 5 and 10 years, respectively.

The activity and decay heat of the high exposure Mark IA fuel with 16% ^{240}Pu content in the plutonium is also shown, in Table 2.6, for Mark IA fuel irradiated 10 years ago (WHC 1992c). Again, Table 2.6 is restricted to radionuclides having activity above five curies per metric ton of initial uranium. Activity for longer, practical decay periods has not been determined. Although there is a limited quantity of fuel with 16% ^{240}Pu content in the plutonium, it has the highest total activity per metric ton among N Reactor fuel.

The mass of the actinides present in this N Reactor fuel has been estimated by the same ORIGEN2 analyses referenced above. Actinides with mass above ten grams per metric ton of initial uranium are listed in Tables 2.7 and 2.8.

Table 2.5: Activity of Selected Radionuclides in Mark IA and Mark IV Fuel with 12% ^{240}Pu in the Plutonium¹

Nuclide	Time Since Reactor Discharge					
	10 Years		20 Years		30 Years	
	Mark IA	Mark IV	Mark IA	Mark IV	Mark IA	Mark IV
^3H	3.08E+01	2.64E+01	1.76E+01	1.51E+01	1.00E+01	8.59E+00
^{55}Fe	8.27E+00	7.64E+00	5.75E-01	5.31E-01	4.00E-02	3.69E-02
^{60}Co	6.14E+00	4.92E+00	1.65E+00	1.32E+00	4.42E-01	3.54E-01
^{85}Kr	5.89E+02	4.81E+02	3.09E+02	2.52E+02	1.62E+02	1.32E+02
^{90}Sr	6.80E+03	5.53E+03	5.36E+03	4.35E+03	4.23E+03	3.43E+03
^{90}Y	6.80E+03	5.53E+03	5.36E+03	4.35E+03	4.23E+03	3.43E+03
^{106}Ru	5.56E+01	5.15E+01	5.74E-02	5.31E-02	5.92E-05	5.48E-05
^{106}Rh	5.56E+01	5.15E+01	5.74E-02	5.31E-02	5.92E-05	5.48E-05
^{125}Sb	1.39E+02	1.27E+02	1.13E+01	1.04E+01	9.30E-01	8.50E-01
$^{125\text{m}}\text{Te}$	3.38E+01	3.09E+01	2.77E+00	2.53E+00	2.27E-01	2.08E-01
^{134}Cs	1.49E+02	1.22E+02	5.17E+00	4.22E+00	1.79E-01	1.46E-01
^{137}Cs	8.39E+03	7.01E+03	6.66E+03	5.57E+03	5.29E+03	4.42E+03
$^{137\text{m}}\text{Ba}$	7.94E+03	6.64E+03	6.30E+03	5.27E+03	5.00E+03	4.18E+03
^{144}Ce	3.24E+01	2.67E+01	4.40E-03	3.62E-03	5.96E-07	4.91E-07
^{144}Pr	3.24E+01	2.67E+01	4.40E-03	3.62E-03	5.96E-07	4.91E-07
^{147}Pm	2.24E+03	1.88E+03	1.59E+02	1.34E+02	1.14E+01	9.55E+00
^{151}Sm	9.77E+01	8.74E+01	9.05E+01	8.09E+01	8.38E+01	7.49E+01
^{154}Eu	1.00E+02	8.49E+01	4.47E+01	3.79E+01	2.00E+01	1.69E+01
^{155}Eu	3.73E+01	3.42E+01	9.21E+00	8.45E+00	2.28E+00	2.09E+00
^{238}Pu	5.05E+01	4.82E+01	4.66E+01	4.46E+01	4.31E+01	4.12E+01
^{239}Pu	1.10E+02	1.10E+02	1.10E+02	1.10E+02	1.10E+02	1.10E+02
^{240}Pu	5.97E+01	5.77E+01	5.97E+01	5.77E+01	5.96E+01	5.76E+01
^{241}Pu	4.47E+03	4.43E+03	2.76E+03	2.74E+03	1.71E+03	1.69E+03
^{241}Am	9.33E+01	9.26E+01	1.48E+02	1.47E+02	1.81E+02	1.79E+02
^{244}Cm	4.64E+00	4.54E+00	3.17E+00	3.10E+00	2.16E+00	2.11E+00
Total ²	3.83E+04	3.25E+04	2.75E+04	2.32E+04	2.11E+04	1.78E+04
Decay Heat ²	1.02E+02	8.57E+01	8.11E+01	6.87E+01	6.71E+01	5.73E+01

1. Units are curies per MTU, except decay heat is watts per MTU. Nuclides with activity below 5 curies per assembly are not listed.
2. Totals include effect of all radionuclides present in one MTU.

Table 2.6: Activity of Selected Radionuclides in Mark IV Fuel with 6% ^{240}Pu in the Plutonium and Mark IA Fuel with 6% and 16% ^{240}Pu in the Plutonium

Nuclide	5 Years Since Discharge		10 Years Since Discharge		
	6% ^{240}Pu		6% ^{240}Pu		16% ^{240}Pu
	Mark IA	Mark IV	Mark IA	Mark IV	Mark IA
^3H	1.58E+01	1.34E+01	1.19E+01	1.01E+01	4.59E+01
^{55}Fe	1.38E+01	1.27E+01	3.63E+00	3.36E+00	1.22E+01
^{60}Co	4.86E+00	3.89E+00	2.52E+00	2.02E+00	8.78E+00
^{85}Kr	3.50E+02	2.87E+02	2.53E+02	2.08E+02	8.07E+02
^{90}Sr	3.30E+03	2.70E+03	2.93E+03	2.39E+03	9.32E+03
^{90}Y	3.30E+03	2.70E+03	2.93E+03	2.39E+03	9.32E+03
^{106}Ru	5.56E+02	5.12E+02	1.79E+01	1.64E+01	8.52E+01
^{106}Rh	5.56E+02	5.12E+02	1.79E+01	1.64E+01	8.52E+01
^{125}Sb	1.83E+02	1.67E+02	5.25E+01	4.76E+01	2.02E+02
$^{125\text{m}}\text{Te}$	4.48E+01	4.06E+01	1.28E+01	1.16E+01	4.94E+01
^{134}Cs	1.19E+02	9.63E+01	2.22E+01	1.79E+01	3.01E+02
^{137}Cs	3.76E+03	3.13E+03	3.35E+03	2.79E+03	1.20E+04
$^{137\text{m}}\text{Ba}$	3.56E+03	2.96E+03	3.17E+03	2.64E+03	1.14E+04
^{144}Ce	1.40E+03	1.16E+03	1.63E+01	1.34E+01	3.97E+01
^{144}Pr	1.40E+03	1.16E+03	1.63E+01	1.35E+01	3.97E+01
$^{144\text{m}}\text{Pr}$	1.68E+01	1.39E+01	1.96E-01	1.61E-01	4.77E-01
^{147}Pm	4.26E+03	3.55E+03	1.14E+03	9.47E+02	2.72E+03
^{151}Sm	6.52E+01	5.70E+01	6.28E+01	5.49E+01	1.10E+02
^{154}Eu	2.18E+01	1.82E+01	1.46E+01	1.22E+01	2.17E+02
^{155}Eu	4.35E+01	4.02E+01	2.16E+01	2.00E+01	5.14E+01
^{238}Pu	7.49E+00	7.41E+00	7.20E+00	7.12E+00	1.22E+02
^{239}Pu	5.60E+01	5.58E+01	5.60E+01	5.58E+01	1.37E+02
^{240}Pu	1.42E+01	1.42E+01	1.42E+01	1.46E+01	9.99E+01
^{241}Pu	6.54E+02	6.64E+02	5.14E+02	5.22E+02	8.72E+03
^{241}Am	6.00E+00	6.10E+00	1.06E+01	1.08E+01	1.84E+02
^{244}Cm	4.28E-02	4.37E-02	3.54E-02	3.61E-02	2.62E+01
Total ²	2.37E+04	1.99E+04	1.47E+04	1.22E+04	5.61E+04
Decay Heat ²	6.37E+01	5.35E+01	4.03E+01	3.37E+01	1.48E+02

1. Units are curies per MTU, except decay heat is watts per MTU. Nuclides with activity below 5 curies per assembly are not listed.
2. Totals include effect of all radionuclides present in one MTU.

Table 2.7: Mass of Selected Actinides in Mark IA and Mark IV Fuel with 12% ^{240}Pu in the Plutonium¹

Nuclide	Time Since Reactor Discharge					
	10 Years		20 Years		30 Years	
	Mark IA	Mark IV	Mark IA	Mark IV	Mark IA	Mark IV
^{234}U	7.46E+01	7.45E+01	7.48E+01	7.47E+01	7.50E+01	7.49E+01
^{235}U	8.41E+03	7.04E+03	8.41E+03	7.04E+03	8.41E+03	7.04E+03
^{236}U	9.22E+02	8.14E+02	9.22E+02	8.14E+02	9.22E+02	8.15E+02
^{238}U	9.85E+05	9.87E+05	9.85E+05	9.87E+05	9.85E+05	9.87E+05
^{237}Np	4.11E+01	3.94E+01	4.17E+01	4.00E+01	4.25E+01	4.07E+01
^{239}Pu	1.76E+03	1.77E+03	1.76E+03	1.77E+03	1.76E+03	1.77E+03
^{240}Pu	2.62E+02	2.53E+02	2.62E+02	2.53E+02	2.62E+02	2.53E+02
^{241}Pu	4.33E+01	4.30E+01	2.68E+01	2.66E+01	1.66E+01	1.64E+01
^{241}Am	2.72E+01	2.70E+01	4.32E+01	4.28E+01	5.26E+01	5.22E+01
Total ²	9.97E+05	9.97E+05	9.97E+05	9.97E+05	9.97E+05	9.97E+05

Table 2.8: Mass of Selected Actinides in Mark IV Fuel with 6% ^{240}Pu in the Plutonium and Mark IA Fuel with 6% and 16% ^{240}Pu in the Plutonium¹

Nuclide	5 Years Since Discharge		10 Years Since Discharge		
	6% ^{240}Pu		6% ^{240}Pu		16% ^{240}Pu
	Mark IA	Mark IV	Mark IA	Mark IV	Mark IA
^{234}U	7.45E+01	6.80E+01	7.45E+01	6.80E+01	6.94E+01
^{235}U	1.01E+04	8.35E+03	1.01E+04	8.35E+03	7.39E+03
^{236}U	7.36E+02	6.90E+02	7.36E+02	6.90E+02	1.18E+03
^{238}U	9.87E+05	9.89E+05	9.87E+05	9.89E+05	9.84E+05
^{237}Np	1.59E+01	1.56E+01	1.59E+01	1.56E+01	6.74E+01
^{239}Pu	9.01E+02	8.97E+02	9.01E+02	8.97E+02	2.20E+03
^{240}Pu	6.23E+01	6.25E+01	6.23E+01	6.25E+01	4.38E+02
^{241}Pu	6.34E+00	6.45E+00	4.99E+00	5.07E+00	8.46E+01
^{242}Pu	2.46E-01	2.50E-01	2.46E-01	2.50E-01	1.64E+01
^{241}Am	1.75E+00	1.78E+00	3.09E+00	3.14E+00	5.36E+01
Total ²	9.99E+05	9.99E+05	9.99E+05	9.99E+05	9.95E+05

1. Units are grams per metric ton of unirradiated uranium. Actinides with mass below 10 grams per assembly are not listed.

2. Totals include all actinides from one metric ton of unirradiated uranium.

2.6 FUEL STORAGE

The N Reactor fuel is stored in the 105-KE and 105-KW Basins in the 100 Area of the Hanford Site (DOE-RL 1991). The basins are constructed of reinforced concrete walls and floors. Their dimensions are 125 feet long (38 meters), 67 feet wide (20 meters), and 21 feet deep (6 meters). Figure 2.2 shows a schematic of the basins. Water levels are maintained in each basin at a minimum of 10 feet (3 meters) above the irradiated fuel to cool the fuel and provide radiological shielding for personnel working in the facility. The water in each basin is recirculated through a closed water-cooling system using mechanical chillers. Filters and ion exchange systems maintain basin water clarity and remove radionuclides.

The 105-KW Basin presently contains approximately 954 MTHM of N Reactor fuel in 3815 closed canisters and the basin water is essentially clean from radionuclide contamination (WHC 1992a, WHC 1994a, WHC 1994b). The 105-KE Basin contains approximately 1146 MTHM of N Reactor fuel stored in 3667 open canisters and its water is contaminated with radionuclides. It is planned to place the 105-KE Basin fuel in containers, which will prevent the water in contact with the fuel from mixing with the basin water, to reduce radioactivity in the basin. (Each basin also contains a small amount of single pass reactor fuel; 105-KW Basin contains 0.1 MTHM and 105-KE Basin contains 0.4 MTHM).

There are three different types of fuel canisters which hold the irradiated fuel elements; Mark 0, Mark I, and Mark II. Each has two cylindrical barrels that hold seven elements vertically. Mark 0 canisters have screen bottoms and open tops, and were fabricated of aluminum. A typical open canister is shown in Figure 2.3. Closed canisters are of two designs. Mark I and Mark II canisters have solid bottoms and sealable tops, although some canisters do not have the tops installed. Each sealed canister barrel is vented through an external gas trap to prevent any buildup of pressure within the canister. The Mark I canisters were made of either aluminum (Al) or stainless steel (SS). All Mark II canisters were made of stainless steel. The Mark I and Mark II canisters are also shown in Figure 2.3. The reference drawings for these storage canisters are listed in Table 2.9.

Table 2.9: Reference Drawings for 105-N Reactor Fuel Storage Canisters

Mark 0 Canister (Al)	H-1-36935, Sheets 1 & 2
Mark 0 Canister (SS)	H-1-42594, Sheets 1 & 2
Mark I Canister	H-1-42793, Sheets 1 & 2
Mark I gas trap	H-1-42817
Mark II canister	H-1-46215, Sheets 1-7

Canisters contain either Mark IA or Mark IV fuel. Some canisters contain fuel elements that have different lengths or have missing inner or outer tubes. Some canisters contain a mixture of odd pieces (UNC 1982).

Figure 2.2: 105 N-REACTOR FUEL STORAGE AT 105-KE 105 KW-BASINS

Figure 2.3

105-N REACTOR FUEL STORAGE CANISTERS

(FBI0009A)

All three types of canisters were filled under water. The fuel, water, and gas volumes in the canisters vary with the type and condition of the fuel. When the Mark I and Mark II canisters were originally filled with fuel and closed, the top 2.5 inches of water in each barrel were displaced with inert nitrogen gas. As a result, they contained 2.8-3.9 gallons of water and 82-128 in³ of nitrogen (UNC 1982).

2.7 FUEL CONDITION

2.7.1 Fuel Oxidation

Uranium and zirconium metals are chemically reactive in water and air (AEC 1960, UNC 1979). Their reactivity is increased by greater metal surface area and higher temperatures, as well as other factors. Small pieces of the metals are more reactive than large pieces, and under the right conditions are pyrophoric. Powders are pyrophoric. Precautions are required to avoid fires when handling or processing uranium or zirconium.

Uranium is oxidized in water and air to form uranium dioxide. Hydrogen is also formed in the reaction with water. Where there is a limited oxygen supply, uranium hydride may also form. The oxide and hydride have significantly lower densities than the metal. The initial uranium dioxide formed creates a protective coating on the uranium metal which greatly retards additional oxidation. After an indeterminate time, the protective layer breaks due to swelling caused by the lower oxide density and is converted to the much more distinctive blue-black form of uranium dioxide. In this form, uranium dioxide does not hinder further oxidation. In addition, the uranium oxide layer cracks and crumbles as it builds. Zirconium oxidation occurs in a similar manner, although the initial protective layer of oxide does not give way to a less protective form.

2.7.2 Breached Cladding

The fuel inventory in the storage basins contains many elements whose cladding was breached during reactor discharge, subsequent handling, or storage. The cladding failures range from cracks to severed fuel elements. The exact number of damaged elements is unknown.

In the storage basins, the uranium metal in the fuel elements that have breached cladding will oxidize after contact with the basin water. This is true for both the open and closed storage canisters. The lower densities of the uranium oxide and uranium hydride products will cause swelling of the metal within the cladding and can cause additional cladding damage, including complete severing of the fuel element. Any additional uranium metal exposed will oxidize as well. New damage to the fuel elements can result in additional fission products release and further contamination of water in the canister or storage basin. In the extreme, the uranium completely oxidizes and forms a mud-like mixture with the water. Fuel in this condition is either contained within Mark I and Mark II canisters or falls through the screen

bottom of the Mark 0 canisters and accumulates on the bottom of the storage basin.

The hydrogen created by these oxidizing reactions is released to the storage basin water. In the closed Mark I and Mark II canisters, nitrogen and water vapor are swept out of the canister along with the hydrogen through the gas vent. The hydrogen generation may eventually displace all the nitrogen from the canister gas space, leaving only hydrogen and water vapor. The conditions present in the closed canisters may result in increased fuel hydriding.

2.7.3 Estimate of Fuel Element Damage

Two surveys have been conducted which provide an indication of the extent of fuel damage within the N Reactor fuel inventory. Both used only visual observations of the fuel within the open topped canisters in the KE Basin. The fuel was not removed from the canisters, so only the top end of elements could be seen; the bottom end and the middle section were not observable. Additionally, there was no documented criteria used for deciding whether a fuel element was damaged; the decision was based on the qualitative observance of the blue-black uranium dioxide corrosion product and the assumption that the observed corrosion product originated from the element on which it lay. The two surveys are described below.

In 1981, a survey of 4993 open-topped canisters stored in 105-KE Basin was made to estimate the number of damaged elements, but not the degree of damage (UNC 1981). From the qualitative observance of the uranium dioxide corrosion product, it was estimated that at minimum, 6% of the fuel elements had cracked cladding and that 50% of the canisters contained an element with cracked cladding. The cladding on 1% of the elements was cracked sufficiently to be considered broken.

In 1991, a similar inspection was made on a much smaller scale (only 100 canisters) (WHC 1992d). In this inspection, only the top ends of fuel elements in one of the two barrels of the canisters could be seen. From these observations, it could be inferred that at least 12% of the fuel elements in this sample are damaged and that 90% of these fuel canisters could contain at least one damaged fuel element. It is not known whether the inferred frequency of damage in this small sample is representative of the conditions in the complete fuel inventory.

The actual extent of damage in the fuel inventory may differ from these estimates due to:

- the uncertainty inherent in the visual observations used in the surveys
- the unknown validity of the assumption that the conditions seen in the top ends of the fuel elements exist in the same frequency in the bottom ends

- the uncertainty of whether the observations could identify the presence of any damage covered by the less noticeable, protective form of the uranium dioxide corrosion product. (This form can occur in the cool, air saturated water in the basins. It was observed during the 1981 investigation that the fuel stored longest in the basin had the most indications of leakage.)

2.7.4 Sludge in the Storage Basins

Fuel handling operations and fuel oxidation resulted in irradiated fuel accumulating on the 105-KE and 105-N Basin floors. Zirconium oxide, iron oxide, concrete grit, and other materials have also accumulated and mixed with the fuel to form a sludge on the basin floors.

Some fuel elements broke into pieces during fuel discharges from N Reactor. While N Reactor operated, 105-N Basin was used to store N Reactor fuel prior to shipment for chemical processing. Some of these pieces remained on the 105-N Basin floor after the bulk of the fuel was loaded into storage canisters. In both 105-N and 105-KE Basin, fuel which had breached cladding oxidized to smaller pieces and uranium oxide "mud," whether it was on the floor or in canisters. Some of the oxidized fuel in Mark 0 canisters fell through the screen bottoms to the basin floor.

In 105-KE Basin, some of the fuel in open topped canisters were repackaged into closed canisters and sent to 105-KW Basin. Some of the oxidized fuel from the open topped canisters remained on the 105-KE Basin floor after the repackaging program.

The 105-KW Basin floor contains no sludge accumulation, because only closed canisters are stored there.

The estimated sludge volumes are 2300 cubic feet in 105-KE Basin and 332 cubic feet in 105-N Basin (WHC 1992e). Neither sludge is adequately characterized, but four samples from each basin were analyzed for activity (WHC 1990a). The averaged results from each basin are shown in Table 2.10. Additional characterization is required to determine the amount of irradiated fuel and other constituents within this sludge.

2.8 FUEL ON THE PUREX CANYON FLOOR

PUREX dissolver cells A, B, and C contain some intact Mark IV fuel elements and fuel element pieces which could not be retrieved after they spilled to the floor during dissolver charging operations over many years. An estimate of the amount and location of fuel on the floor of the PUREX dissolver cells is given in Table 2.11, based on reports and records of the fuel spills (ARH 1969, ARH 1971a, ARH 1971b, ARH 1971c, RHO 1983, RHO 1985b, RHO 1987, WHC 1987, WHC 1990b).

Table 2.10: Activity of Selected Radionuclides in the 105-KE and 105-N Storage Basin Sludge^{1,2}

Nuclide	105-KE Basin	105-N Basin
³ H	4.56E-02	7.49E-02
¹⁰ Be	N.A. ³	1.65E-04
¹⁴ C	1.03E-03	1.59E-03
⁵¹ Cr	1.33E+00	N.A.
⁵⁴ Mn	N.A.	5.36E-01
⁵⁷ Co	N.A.	6.41E-02
⁵⁹ Ni	1.23E-03	3.65E-03
⁶⁰ Co	7.65E-01	7.28E+00
⁶³ Ni	5.35E-02	1.72E-01
⁶⁵ Zn	N.A.	4.91E-02
⁸⁹ Sr	5.61E+00	2.73E-01
⁹⁰ Sr	1.94E+01	6.15E-01
¹²⁵ Sb	1.95E-01	3.09E-02
¹³⁴ Cs	1.77E-01	N.A.
¹³⁷ Cs	2.87E+01	2.70E+00
¹⁴⁰ Ba/ ¹⁴⁰ La	1.39E-01	N.A.
¹⁴⁴ Ce/ ¹⁴⁰ Pr	7.71E-02	3.64E-01
²³⁴ U	2.85E-03	1.27E-03
²³⁵ U	4.94E-04	2.18E-04
²³⁸ U	2.27E-03	1.03E-03
²³⁸ Pu	2.89E-01	2.88E-02
²³⁹ Pu/ ²⁴⁰ Pu	1.64E+00	1.69E-01
²⁴¹ Pu	3.97E+01	6.87E+00
²⁴¹ Am	3.80E+00	9.19E-02
²⁴² Cm	7.55E-03	1.31E-04
²⁴³ Cm/ ²⁴⁴ Cm	2.13E-02	1.19E-03
Total ⁴	1.02E+02	1.93E+01

1. Values shown are averages from the analysis of up to four samples taken at different locations in the basins to support the 1990 Characterization Report.
2. Units are curies per metric ton of sludge.
3. N.A. indicates that no activity values for the nuclide are available.
4. Totals only reflect the sum of the activities listed; other radionuclides may be present.

Table 2.11: Estimate of 105-N Reactor Fuel Elements on the PUREX Canyon Floor

<u>PUREX CELL</u>	<u>Uranium Kg</u>	<u>Number of Fuel Elements</u>	<u>Mark IV Fuel Length Type (Estimated)</u>
A	26	3.5 inner	E
B	230	22.5 inner, 11.5 outer	C
C	4-8	1.0 inner	E

More than 90% of this 260 - 264 kilogram estimate is uranium with a 0.947% pre-irradiation enrichment, with the remainder being 0.71% pre-irradiation content. The exact amount, length, and ^{240}Pu content in the plutonium of the fuel cannot be determined until the fuel is recovered from the canyon floor.

3.0 SHIPPINGPORT PRESSURIZED WATER REACTOR CORE II FUEL

3.1 PURPOSE OF SHIPPINGPORT PRESSURIZED WATER REACTOR CORE II FUEL

The Shippingport Pressurized Water Reactor (PWR) Core II fuel was a developmental fuel in the U. S. Energy Research and Development Administration (now DOE) Light Water Breeder Reactor research program (BAPL 1968). It was the second of three cores irradiated in the Shippingport Atomic Power Station at Shippingport, Pennsylvania. The purpose of the program was to advance reactor technology in the design, fabrication, and operation of central station nuclear power plant cores. A principal objective of the second Shippingport core was the development of a high performance seed and blanket core, in terms of both power density and lifetime. (The electricity produced from the Shippingport Reactor with PWR Core II went to the Duquesne Light Company.)

A high power density, high total energy core required the development of a fuel element of high loading which would withstand long irradiation exposure at high heat fluxes. The experience of the first Shippingport core, showed that bulk uranium oxide had great potential for stability during irradiation. Placing the uranium oxide into compartmented fuel plates to lower fuel temperatures by increasing heat transfer area within the assembly was a logical step in the development of the second core assemblies. Zircaloy-4 was chosen for the skeleton of the compartmented fuel plates and for the assembly cladding, because of its low neutron and hydrogen absorption rates.

3.2 SHIPPINGPORT REACTOR CORE AND OPERATING HISTORY

The reactor core contained 97 assemblies, including 72 standard blanket assemblies (natural uranium above 96% theoretical bulk density), 20 seed assemblies (93.2% enriched ^{235}U driver fuel), four special blanket assemblies (two containing natural uranium at 90% theoretical bulk density and two containing equal amounts of natural uranium with theoretical bulk densities of 90% and at least 96%), and one multipurpose extended life blanket assembly (MELBA), which contained irradiated fuel from the first core (BAPL 1968).

The reactor operated from April 1965 through March 1969 with one set of seed assemblies and from July 1969 through February 1974 with a second set of seed assemblies (BAPL 1983). The reactor operated at an average of 46% of full power for 1250 days with the first seed and at an average of 30% of full power for 1410 days with the second seed. The 72 standard blanket assemblies, the four special blanket assemblies and the MELBA remained in the core during the entire irradiation period. Their exposure during the PWR Core II irradiation was 16,200 MWD/MTU.

In 1974, the second reactor core was shutdown. The 72 standard blanket assemblies were shipped to the Hanford Site during 1978 and 1979 and have been stored underwater in the T-Plant Canyon Building since then (DLC 1979). None

of the other irradiated assemblies were brought to Hanford. A single, unirradiated standard blanket assembly was brought to Hanford in July 1989 to serve as a cold test unit for demonstrating mechanical handling, disassembly, and processing concepts. Testing was initiated, but later canceled due to budget constraints. The unirradiated assembly is currently stored in the T-Plant canyon.

3.3 STANDARD BLANKET ASSEMBLY CONSTRUCTION

The 72 standard blanket assemblies are 142.3 inches (3.6 meters) long and have a 7.5 inch (19 centimeter) square cross-section (BAPL 1968). An assembly is shown schematically in Figure 3.1. They contain three basic sections which are bolted together: an extended fuel cluster and top and bottom extension brackets. The fuel cluster is clad with Zircaloy-4. The extension brackets are made of 304 stainless steel. Each assembly weighs approximately 1180 pounds (536 kg).

The fuel cluster consists of two identical oxide fuel plate subassemblies welded together to form a square structure and two Zircaloy-4 cluster extensions welded to the ends of the subassemblies. Each subassembly consists of 30 compartmented fuel plates and two Zircaloy-4 end plates welded together to form parallel coolant channels. The fuel plate design includes many small fuel wafers surrounded by a Zircaloy-4 grid to provide adequate structural strength. The wafers have a pyrolytic carbon coating which prevented the zirconium from reacting chemically with the uranium oxide (BAPL 1962). The upper and lower halves of the extended fuel cluster are mirror images.

The minor differences in the top and bottom extension brackets between blanket assemblies are due to the two-pass flow arrangement used to optimize heat transfer in the reactor core. The top extension brackets are identical, except for the location of the cooling water outlet passages. The bottom extension brackets are basically the same, except for key slots to prevent inserting an assembly into the wrong cooling water pass location.

The assembly dimensions and materials of construction are summarized in Table 3.1. Drawings showing the details of assembly construction are included in Appendix 1.

3.4 UNIRRADIATED BLANKET ASSEMBLY COMPOSITION

The pre-irradiation bulk composition of the standard blanket fuel assemblies is shown in Table 3.2 (BAPL 1962, BAPL 1968, RHO 1984). The calculational details are shown in Appendix 2. The impurities in the specific materials used for the standard blanket assemblies are not available. However, the limits found in documents describing part of the development of the PWR Core II assembly materials and in the American Iron and Steel Institute standards, can be used for guidance when impurity levels are important. These limits are listed in Table 3.3 (AISI, BAPL 1962, ASTM, BAPL 1968).

Figure 3.1

Table 3.1: Shippingport Core II Standard Blanket Assembly Components Dimensions and Materials

<u>Assembly Component</u>	<u>Dimensions</u>		<u>Materials¹</u>
	<u>Length</u> inches	<u>Cross-section or diameter</u> inches	
Blanket assembly	142 5/16	7.5 x 7.5	Zircaloy-4, 304 SS, UO ₂ , carbon
Blanket Subassemblies			
top extension	18.5	7.5 x 7.5	304 SS
bottom extension	19 5/16	7.375	304 SS
extended fuel cluster	104.5	7.5 x 7.5	Zircaloy-4, UO ₂ , carbon
Extended Fuel Cluster			
extensions	3.398	7.395 x 7.376	Zircaloy-4
fuel cluster	97.705	7.395 x 7.376	Zircaloy-4, UO ₂ , carbon
Fuel Cluster			
fuel plates	97.705	7.0 x 0.14	Zircaloy-4, UO ₂ , carbon
square-end fuel wafers	1.529	0.248 x 0.10	UO ₂ , carbon
radial-end fuel wafers	1.480	0.248 x 0.10	UO ₂ , carbon

1. "304 SS" is American Iron and Steel Institute Type 304 stainless steel.

Table 3.2: Shippingport Core II Standard Blanket Assembly Bulk Composition

<u>Material</u>	<u>Amount</u>	
	<u>Kg/assembly</u>	<u>Kg/MTU</u>
Natural Uranium (in UO ₂)	225.5	1,000
Oxygen (in UO ₂)	30.3	134
Zircaloy-4 cladding	196.1	870
304 Stainless Steel	83.3	369
Carbon coating on fuel	0.08	0.3

Table 3.3: Impurity Limits Within Shippingport Core II Materials				
Impurity	Materials			
	Uranium	Zircaloy-4	Pyrolytic Carbon	Type 304 Stainless
	ppm	ppm	ppm	ppm
Aluminum	150	-	17	500
Arsenic	-	-	-	300
Boron	1	-	-	10
Cadmium	.5	-	-	-
Carbon	-	500	-	-
Chromium	-	-	13	174,700
Chromium + Iron	-	4,000	-	-
Cobalt	10	-	-	500
Copper	-	-	-	400
Hafnium	-	45,000	-	-
Hydrogen	-	50	40	-
Iron	150	-	40	Balance
Manganese	-	-	-	17,500
Molybdenum	100	-	-	-
Nickel	-	-	6	113,00
Niobium	-	-	-	500
Nitrogen	200	250	-	100
Oxygen	-	1,800	-	-
Silicon	-	-	17	6,200
Sulfur	-	-	-	100
Tantalum	-	-	-	200
Tin	-	20,000	-	-
Vanadium	-	-	-	2,000

3.5 FUEL INVENTORY

The irradiated Shippingport Core II fuel contains 15.66 metric tons of uranium and 0.12 metric tons of plutonium, based on nuclear material safeguards reports (WHC 1993a). The uranium and plutonium content of the 72 irradiated blanket assemblies is very similar, although small differences exist due to their different locations within the reactor core. The average uranium content of an irradiated assembly is 217.5 kilograms, while the average plutonium content is 1.67 kilograms (DLC 1979, WHC 1993a). The ^{240}Pu isotopic content of the plutonium is estimated at 23% by ORIGEN2 analysis (WHC 1991b).

3.6 RADIONUCLIDE CONTENT

The concentration of radionuclides present in the Shippingport Core II fuel is a function of the original fuel composition, irradiation history, and decay time. Short lived fission products have decayed to insignificant concentrations during the time since the fuel was irradiated. The radionuclide activity and decay heat of the Shippingport Core II assemblies were determined by ORIGEN2 analysis, based on the bulk composition and general operating history described above (WHC 1991b). The activity and decay heat results are shown in Table 3.4 at 10, 20, 25, and 30 years since irradiation. Similar information for the removable stainless steel extensions of the assemblies is shown in Table 3.5. The radionuclides that have decayed to less than ten curies per assembly are not listed in these tables. The mass of the actinides in the assemblies has also been determined by the same ORIGEN2 analysis. Actinides with mass above 100 grams per assembly are presented in Table 3.6. More complete radionuclide inventories, including longer decay periods, is presented in reference (WHC 1991b).

At discharge from the Shippingport Reactor, some radioactive corrosion product adhered to the assemblies. The source was the reactor coolant and the radioactivity was believed to be primarily from cobalt-60 and iron-55 (BAPL 1977).

3.7 FUEL STORAGE (RHO 1978, RHO 1979)

The Shippingport Core II blanket assemblies are stored at T Plant in the 221-T Building canyon pool cell (Cell 4). The water depth in the pool cell is 19 feet (5.8 meters). Each assembly is stored vertically in a separate compartment of a rectangular metal rack. The assembly storage is shown schematically in Figure 3.2. Support systems include ion exchange purification to remove any radionuclides entering the water, filtration to remove particulates falling into the pool water and water cooling to remove the decay heat. Additional information on these systems is shown in the instrument and electrical flow diagram, H-2-91285.

Limited contamination in the routine samples of the pool water and a visual inspection of the stored assemblies indicate that the assembly cladding is not breached.

Table 3.4: Activity of Selected Radionuclides in Shippingport Core II Blanket Assemblies¹

Nuclide	Time Since Reactor Discharge			
	10 Years	20 Years	25 Years	30 Years
³ H	3.88E+01	2.22E+01	1.67E+01	1.26E+01
⁵⁵ Fe	1.51E+03	1.05E+02	2.78E+01	7.32E+00
⁶⁰ Co	1.05E+03	2.81E+02	1.46E+02	7.55E+01
⁶³ Ni	6.69E+02	6.21E+02	5.98E+02	5.76E+02
⁸⁵ Kr ²	3.99E+02	2.09E+02	1.51E+02	1.10E+02
⁹⁰ Sr ²	4.47E+03	3.53E+03	3.13E+03	2.78E+03
⁹⁰ Y ²	4.47E+03	3.53E+03	3.13E+03	2.78E+03
¹⁰⁶ Ru	7.19E+01	7.42E-02	2.38E-03	7.65E-05
¹⁰⁶ Rh	7.19E+01	7.42E-02	2.38E-03	7.65E-05
¹²⁵ Sb	2.67E+02	2.19E+01	6.26E+00	1.79E+00
^{125m} Te	6.52E+01	5.34E+00	1.53E+00	4.37E-01
¹³⁴ Cs	4.04E+02	1.40E+01	2.61E+00	4.86E-01
¹³⁷ Cs	8.98E+03	7.13E+03	6.35E+03	5.66E+03
^{137m} Ba	8.50E+03	6.74E+03	6.01E+03	5.35E+03
¹⁴⁴ Ce	1.10E+01	1.50E-03	1.74E-05	2.03E-07
¹⁴⁴ Pr	1.10E+01	1.50E-03	1.74E-05	2.03E-07
¹⁴⁷ Pm	1.12E+03	8.00E+01	2.13E+01	5.69E+00
¹⁵¹ Sm	4.65E+01	4.30E+01	4.14E+01	3.98E+01
¹⁵⁴ Eu	5.17E+02	2.31E+02	1.54E+02	1.03E+02
¹⁵⁵ Eu	1.91E+02	4.72E+01	2.35E+01	1.17E+01
²³⁸ Pu	2.74E+02	2.54E+02	2.44E+02	2.34E+02
²³⁹ Pu	6.88E+01	6.88E+01	6.88E+01	6.88E+01
²⁴⁰ Pu ²	9.32E+01	9.33E+01	9.33E+01	9.33E+01
²⁴¹ Pu	1.71E+04	1.06E+04	8.30E+03	6.53E+03
²⁴¹ Am	3.98E+02	6.07E+02	6.77E+02	7.30E+02
²⁴⁴ Cm	1.97E+02	1.35E+02	1.11E+02	9.18E+01
Total ³	5.10E+04	3.44E+04	2.93E+04	2.53E+04
Decay Heat ³	1.36E+02	1.04E+02	9.58E+01	8.92E+01

1. Units are curies per assembly, except decay heat is watts per assembly. Nuclides with activity below 10 curies per assembly are not listed.
2. Higher activities for these nuclides are listed in (RHO 1978), but the basis is unavailable. These higher activities should be considered, when significant to the application: ⁸⁵Kr may be 40% higher, ⁹⁰Sr and ⁹⁰Y may be 50% higher, and ²⁴⁰Pu may be 10% higher.
3. Totals include effect of all radionuclides present in one assembly.

Table 3.5: Activity of Selected Radionuclides in Stainless Steel Extensions of the Shippingport Core II Blanket Assemblies¹

Nuclide	Time Since Reactor Discharge			
	10 Years	20 Years	25 Years	30 Years
⁵⁵ Fe	1.49E+03	1.04E+02	2.74E+01	7.22E+00
⁶⁰ Co	9.96E+02	2.67E+02	1.39E+02	7.18E+01
⁶³ Ni	6.69E+02	6.21E+02	5.98E+02	5.76E+02
Total ²	3.17E+03	9.98E+02	7.69E+02	6.60E+02
Decay Heat ²	1.57E+01	4.37E+00	2.38E+00	1.34E+00

1. Units are curies per assembly, except decay heat is watts per assembly. Nuclides with activity below 10 curies per assembly are not listed.
2. Totals include all activation products present in one assembly.

Table 3.6: Mass of Selected Actinides in Shippingport Core II Blanket Assemblies¹

Nuclide	Time Since Discharge			
	10 Years	20 Years	25 Years	30 Years
²³⁵ U	4.84E+02	4.84E+02	4.84E+02	4.84E+02
²³⁸ U	2.19E+05	2.19E+05	2.19E+05	2.19E+05
²³⁹ Pu	1.11E+03	1.11E+03	1.11E+03	1.11E+03
²⁴⁰ Pu	4.09E+02	4.09E+02	4.09E+02	4.10E+02
²⁴¹ Pu	1.66E+02	1.03E+02	8.06E+01	6.33E+01
²⁴¹ Am	1.16E+02	1.77E+02	1.97E+02	2.13E+02
Total ²	2.22E+05	2.22E+05	2.22E+05	2.22E+05

1. Units are grams per assembly. Actinides with mass below 100 grams per assembly are not listed.
2. Totals include all actinides in one assembly.

4.0 FAST FLUX TEST FACILITY FUEL

4.1 BACKGROUND (ANS 1990)

The purpose of the Fast Flux Test Facility (FFTF) was to provide testing capability to satisfy the diverse technology development needs of the United States advanced reactor programs. The mission included irradiation and evaluation of different types of fuel assemblies and different materials for fuel assembly construction. Also included was direct production of useful materials such as medical isotopes. In December 1993, the DOE directed that the FFTF be transitioned to shutdown status. In April 1994, removal of fuel from the reactor began.

The FFTF includes a liquid-sodium cooled reactor capable of simulating the irradiation conditions of a full scale commercial liquid-metal reactor. To meet the goals of the test programs, the assembly configuration in the reactor core can be routinely redesigned. Assemblies can be relocated within the core or replaced with different assemblies (irradiated or unirradiated) from outside the core. There are three storage locations for irradiated assemblies, not including the core: the in-vessel storage (IVS), the interim decay storage (IDS), and the fuel storage facility (FSF). All are liquid sodium cooled storage. Assemblies stored in the IVS or IDS have been routinely reintroduced into the reactor core, however fueled assemblies stored in the FSF have not been reinserted into the core.

4.2 FUEL ASSEMBLIES

The FFTF assemblies are divided into two general categories; driver fuel assemblies (DFA) and test assemblies. The primary purpose of the driver fuel assemblies is to hold the fuel which creates the neutron flux environment to carry out the various FFTF tests. The primary purpose of the test assemblies is to hold the fuel and non-fuel materials being tested. The fueled test assemblies also contribute to the neutron flux within the reactor core.

Test assemblies have been further categorized for the purpose of this document. The test assemblies which are identical to or very similar to the DFAs are categorized as test driver fuel assemblies (TDFA). The test assemblies which contain experimental fuel are categorized as test fuel assemblies (TFA), even if they also contain some standard driver fuel pins. The remaining test assemblies do not contain fuel.

As of October 1992, there were 210 DFAs, 65 TDFAs, and 54 TFAs irradiated at the FFTF, although 1 DFA, 14 TDFAs and 13 TFAs have been disassembled. All the irradiated fuel is included in the scope of the Hanford SNF Management EIS. The non-fueled test assemblies are not.

As of October 1992, there were also 55 DFAs, one TDFA, and two TFAs at the Hanford Site that had been fabricated, but not irradiated. Unirradiated

assemblies are not included in the scope of the Hanford SNF Management EIS.

4.2.1 Driver Fuel Assemblies

4.2.1.1 Standard Driver Fuel Assembly Construction (ANS 1990) - The DFAs are hexagonally shaped components, which are composed of a bundle of 217 fuel pins, a surrounding duct, a shield orifice assembly, an inlet nozzle, load pads, and a handling socket. The assembly is 12 feet (3.6 meters) long, 4.575 inches (11.6 centimeters) wide across the hexagon flats, 5.16 inches (13.1 centimeters) wide across the hexagon points, and weighs 381 pounds (173 kilograms). The assembly is shown schematically in Figure 4.1. The assembly cross-section is shown in Figure 4.2.

The principal structural component is the 316 stainless steel (SS), hexagonal duct which extends from a handling socket at the top to a shield/orifice region located below the fuel pins. The duct wall thickness is nominally 0.120 inches (0.30 centimeters), but varies from 0.065 inches (0.17 centimeters) through 0.190 inches (0.48 centimeters) along the length to facilitate lateral positioning of the assemblies within the core and the distribution of force among the assemblies within the core.

The fuel pins are 0.23 inches (0.58 centimeters) in diameter, approximately 93.5 inches (2.37 meters) long and have a 36 inch (91 centimeter) fuel bearing region, which is centered 65.5 inches (1.66 meters) from the bottom end of the fuel assembly. Each fuel pin is helically wrapped with a 0.056 inch (0.14 centimeter) diameter steel wire to provide lateral spacing along its length. The fuel pins are arranged in a triangular pitch within the hexagonal duct. The fuel pin is also shown in Figure 4.1.

The fuel region contains approximately 150 pressed and sintered, mixed uranium-plutonium oxide pellets and has two 0.4 inch (1 centimeter) uranium oxide pellets on each end for temperature insulation. The mixed uranium-plutonium oxide is at a nominal theoretical bulk density of 90.4% and the uranium oxide in the insulator pellets is at a nominal theoretical bulk density of 95%.

Within each fuel pin, there is a 37 inch (94 centimeter) fission gas plenum above the fuel region to limit internal gas pressure buildup caused by gaseous fission product generation. The fuel pin cladding and end caps seal

FAST FLUX TEST FACILITY
FUEL PIN BUNDLE CROSS SECTION

(FBI0007A)

Figure 4.2

these gases within the fuel pin. The plenum also contains a unique krypton/xenon tag gas which facilitates identifying a leaking assembly.

The shield/orifice assembly is in the lower end of the fuel assembly. The orifices controlling the sodium coolant flow are multiple holes in a series of steel plates. The volume of steel in the orifice assembly together with the sodium coolant provide the required neutron attenuation.

A summary of the materials used in the DFAs, including fuel pins is shown in Table 4.1. Reference drawings for the major assembly components are listed in Table 4.2.

4.2.1.2 Assembly Enrichment - The center area of the FFTF Reactor core has a higher neutron flux than the outer core area. To flatten radial power distribution, the assemblies that are used in the center region of the core, called the inner enrichment zone, have lower plutonium enrichments than those that are used in the outer enrichment zone area. The FFTF Reactor vessel and Reactor core are shown in Figures 4.3 and 4.4. A second set of inner and outer area assemblies was built with higher enrichments to permit a larger number of non-fuel test assemblies to be in the core at the same time. The first set of assemblies are designated as Series I and includes assembly types 3.1 and 3.2, while the second set is designated as Series II and includes assembly types 4.1 and 4.2. In both series, the ".1" identifies an outer zone assembly and the ".2" identifies an inner zone assembly.

The plutonium enrichment of the DFAs are expressed as weight percent plutonium of the total plutonium and uranium present ($\%Pu/[Pu+U]$). The pre-irradiation enrichments are shown in Table 4.3, along with the plutonium and uranium contents of the assemblies and fuel pins. The isotopic breakdown is also shown.

4.2.1.3 Assembly Exposure and Radionuclide Content - In operating the FFTF to create the proper reactor environment for the test assemblies, the individual DFAs are frequently moved into and out of the FFTF Reactor. As a result, they have widely varying exposures and resultant radionuclide inventories. Figure 4.5 shows the distribution of average exposures for the DFAs and the TDFAs in megawatt days per kilogram of total uranium and plutonium (WHC 1992f). The maximum exposure of any DFA is 100 MWD/kg and only seven have been exposed above 90 MWD/kg. The maximum exposure of any TDFA is below 150 MWD/kg and only four have been exposed above 90 MWD/kg. The maximum exposure of any TFA is just below 200 MWD/kg and only four have been exposed above 150 MWD/kg. The exposure of individual DFAs is shown in Appendix 3. The exposure of the individual TDFAs is shown in Section 4.2.2, while the exposure of the individual TFAs is shown in Section 4.2.3.

Table 4.1: Materials Used in the Fast Flux Test Facility Standard Driver Fuel Assemblies

<u>Item</u>	<u>Material</u>
Handling socket	316-SS ¹
Duct	316-SS, 20% cold worked
Shield orifice assembly	316-SS
Inlet nozzle	316-SS, 20% cold worked
End caps	316-SS, 20% cold worked
Plenum spacer assembly	316-SS, 20% cold worked
Wire wrap	316-SS, 17% cold worked
Pin cladding	316-SS, 20% cold worked
Axial reflector	high nickel alloy
Insulator pellets	natural or depleted uranium, at 95% T.D. ²
Fuel pellets	mixed plutonium-uranium oxide, at 90.4% T.D.
Pin attachment	316-SS
Fill gas	helium
Tag gas capsule	various stainless steels
Tag gas	isotopes of stable xenon and/or krypton

1. "316-SS" is American Iron and Steel Institute Type 316 stainless steel
2. "T.D." is theoretical density

Table 4.2: Reference Drawings for the Fast Flux Test Facility Standard Driver Fuel Assemblies

<u>Description of Drawing</u>	<u>Drawing Number</u>	
	<u>Series I</u> <u>(Types 3.1 & 3.2)</u>	<u>Series II</u> <u>(Types 4.1 & 4.2)</u>
Assembly	H4-21500 Rev 9	H4-54101 Rev 3
Duct tube	H4-21476 Rev 7	H4-54114 Rev 4
Fuel pin assembly	H4-20581 Rev 8	H4-54122 Rev 5
Handling socket	H4-21510 Rev 8	H4-54111 Rev 3
Nozzle	H4-23064 Rev 5	H4-54106 Rev 1
Fuel Pellet	H4-20579 Rev 6	H4-54126 Rev 2

Figure 4.3: FAST FLUX TEST FACILITY REACTOR VESSEL

FAST FLUX TEST FACILITY
REACTOR CORE CROSS SECTION

(FB10006A)

Figure 4.4

Table 4.3: Plutonium and Uranium Content of Unirradiated Fast Flux Test Facility Driver Fuel Assemblies

	Driver Fuel Assembly Type			
	<u>3.1</u>	<u>3.2</u>	<u>4.1</u>	<u>4.2</u>
Plutonium				
Enrichment, %Pu/(Pu+U)	27.37	22.43	29.28	25.14
Assembly content, kg	9.071	7.421	9.722	8.333
Fuel pin content, gm	41.8	34.2	44.8	38.4
Isotopic Fraction				
²³⁹ Pu	0.8696	0.8696	0.8711	0.8711
²⁴⁰ Pu	0.1173	0.1173	0.1163	0.1163
²⁴¹ Pu	0.0104	0.0104	0.0102	0.0102
Uranium ¹				
Enrichment, %U/(Pu+U)	72.63	77.57	70.72	74.86
Assembly content, kg	24.070	25.666	23.481	24.813
Fuel pin content, gm	110.9	118.3	108.2	114.3
Isotopic Fraction				
²³⁵ U	0.0070	0.0070	0.0020	0.0020
²³⁸ U	0.9930	0.9930	0.9980	0.9980

Note: 1. Excludes the depleted uranium in the two insulator pellets within each fuel pin, at the top and bottom of the fuel column. Each assembly hold nominally 1.5 kilograms of uranium in the insulator pellets.

AVERAGE EXPOSURE OF STANDARD FUELED FFTF ASSEMBLIES
(THROUGH FFTF OPERATING CYCLE 12B2)

* Assemblies have been grouped in 5MWD/Kg increments. Exposure values are maximum of 5MWD/Kg range.
Figure 4.5

The concentration of radionuclides present in the FFTF fuel is a function of the original fuel composition, irradiation history, and decay time. Many fission products will have decayed to insignificant concentrations shortly after the fuel is removed from the reactor. The radionuclide activity of Type 4.1 and Type 4.2 DFAs at the 100 MWD/kg and 150 MWD/kg exposures has been determined using the ORIGEN2 computer code (WHC 1991c, WHC 1991d). These results are summarized in Tables 4.4 and 4.5, for activities above ten curies per assembly. Total assembly decay heat is also shown. The mass of the actinides in Type 4.1 and Type 4.2 assemblies was also estimated using ORIGEN2 analysis. Actinides with mass above ten grams per assembly are shown in Tables 4.6 and 4.7.

4.2.1.4 Disassembled Driver Fuel Assembly - Only one irradiated DFA (serial number 16392) has been dismantled (WHC 1992g). Its fuel pins have been placed into three assembly component containers, called Ident 69 containers: ID69001 holding 34 of the pins, ID69009 holding 118 of the pins, and ID69012 holding 64 of the pins. These three Ident 69 containers are stored at the FSF.

4.2.2 Test Driver Fuel Assemblies

The TDFAs are either identical or very similar to the standard DFAs. Each contains fuel with only one of the standard four plutonium enrichments used in the DFAs, except the C-1 assembly which holds two different types of standard fuel. Some TDFAs have components made of D9 or HT9 stainless steels in lieu of 316 stainless steel.

In general, the purpose of irradiating the TDFAs was to determine the operating characteristics and the design limits of the standard DFAs. Tables 4.8 through 4.11 summarize information on the TDFAs. Table 4.8 lists the average exposure of each assembly and the FFTF operating cycle when each assembly was last irradiated (WHC 1992f). Table 4.9 lists the duct and cladding material for those TDFAs not having the standard 20% cold-worked 316 stainless steel (ANS 1990).

Fuel pins were removed from some TDFAs and shipped off site for examination. Some of these fuel pins were either returned whole in Ident 69 containers or as pieces in storage casks described in Section 6.2 of this report. Most of the remaining fuel pins will eventually be returned to Hanford. The other fuel pins from these TDFAs, which were not shipped off site, are either stored in the original TDFA or in Ident 69 containers. Table 4.10 lists the TDFAs from which some fuel pins have been removed and the number of fuel pins remaining at FFTF. Table 4.11 lists similar information for the TDFAs from which all fuel pins have been removed and the distribution of fuel pins among the Ident 69 containers (WHC 1992h, WHC 1992i).

The TDFAs and their specific purposes are briefly discussed below. More information is available in the Detailed Irradiation Plans and the test assembly records of the Manager of Irradiation Test Management, stored in the 400 Area of the Hanford Site.

Table 4.4: Activity of Selected Radionuclides in Fast Flux Test Facility Types 4.1 and 4.2 Driver Assemblies with 100 MWD/kg Exposure

Nuclide	Time Since Reactor Discharge					
	5 Years		10 Years		20 Years	
	Type 4.1	Type 4.2	Type 4.1	Type 4.2	Type 4.1	Type 4.2
³ H	5.17E+01	5.25E+01	3.90E+01	3.96E+01	2.23E+01	2.26E+01
⁵⁴ Mn	1.36E+02	1.76E+02	2.38E+00	3.06E+00	7.20E-04	9.26E-04
⁵⁵ Fe	1.44E+02	1.72E+02	3.81E+01	4.54E+01	2.65E+00	3.16E+00
⁶⁰ Co	1.53E+02	1.81E+02	7.93E+01	9.39E+01	2.13E+01	2.52E+01
⁶³ Ni	1.31E+01	1.49E+01	1.26E+01	1.44E+01	1.17E+01	1.33E+01
⁸⁵ Kr	4.39E+02	4.48E+02	3.18E+02	3.24E+02	1.67E+02	1.70E+02
⁹⁰ Sr	3.25E+03	3.30E+03	2.89E+03	2.93E+03	2.28E+03	2.31E+03
⁴ ⁹⁰ Y	3.26E+03	3.30E+03	2.89E+03	2.93E+03	2.28E+03	2.31E+03
¹⁰⁶ Ru	3.94E+03	4.38E+03	1.27E+02	1.41E+02	1.31E-01	1.45E-01
¹⁰⁶ Rh	3.94E+03	4.38E+03	1.27E+02	1.41E+02	1.31E-01	1.45E-01
¹²⁵ Sb	8.65E+02	9.28E+02	2.48E+02	2.66E+02	2.03E+01	2.17E+01
^{125m} Te	2.11E+02	2.26E+02	6.04E+01	6.48E+01	4.94E+00	5.31E+00
¹³⁴ Cs	1.11E+03	1.30E+03	2.07E+02	2.41E+02	7.18E+00	8.36E+00
¹³⁷ Cs	8.93E+03	8.94E+03	7.96E+03	7.97E+03	6.32E+03	6.32E+03
^{137m} Ba	8.45E+03	8.46E+03	7.53E+03	7.54E+03	5.98E+03	5.98E+03
¹⁴⁴ Ce	1.37E+03	1.58E+03	1.60E+01	1.84E+01	2.17E-03	2.50E-03
¹⁴⁴ Pr	1.37E+03	1.58E+03	1.60E+01	1.84E+01	2.17E-03	2.50E-03
^{144m} Pr	1.65E+01	1.90E+01	1.92E-01	2.21E-01	2.60E-05	3.00E-05
¹⁴⁷ Pm	6.89E+03	7.05E+03	1.84E+03	1.88E+03	1.31E+02	1.34E+02
¹⁵¹ Sm	3.68E+02	3.62E+02	3.54E+02	3.48E+02	3.28E+02	3.22E+02
¹⁵⁴ Eu	2.36E+02	2.61E+02	1.58E+02	1.75E+02	7.04E+01	7.80E+01
¹⁵⁵ Eu	8.01E+02	8.08E+02	3.98E+02	4.02E+02	9.84E+01	9.93E+01
²³⁸ Pu	9.32E+01	1.01E+02	8.98E+01	9.69E+01	8.35E+01	9.00E+01
²³⁹ Pu	3.97E+02	3.45E+02	3.97E+02	3.45E+02	3.97E+02	3.44E+02
²⁴⁰ Pu	3.44E+02	3.02E+02	3.44E+02	3.02E+02	3.43E+02	3.02E+02
²⁴¹ Pu	1.28E+04	1.15E+04	1.01E+04	9.05E+03	6.23E+03	5.59E+03
²⁴¹ Am	1.89E+02	1.65E+02	2.79E+02	2.45E+02	4.02E+02	3.55E+02
^{242m} Am	8.00E+00	7.71E+00	7.82E+00	7.54E+00	7.47E+00	7.20E+00
²⁴² Am	7.96E+00	7.68E+00	7.78E+00	7.50E+00	7.43E+00	7.17E+00
²⁴² Cm	8.66E+00	8.81E+00	6.44E+00	6.21E+00	6.15E+00	5.93E+00
Total ²	5.99E+04	6.04E+04	3.65E+04	3.57E+04	2.52E+04	2.45E+04
Decay Heat ²	1.71E+02	1.76E+02	1.02E+02	9.97E+01	8.69E+01	8.31E+01

1. Units are curies per assembly, except decay heat which is watts per assembly. Nuclides with activity below 5 curies per assembly are not listed.
2. Totals include effect of all radionuclides present in one assembly.

Table 4.5: Activity of Selected Radionuclides in Fast Flux Test Facility Types 4.1 and 4.2 Driver Assemblies with 150 MWD/kg Exposure¹

Nuclide	Time Since Reactor Discharge					
	5 Years		10 Years		20 Years	
	Type 4.1	Type 4.2	Type 4.1	Type 4.2	Type 4.1	Type 4.2
³ H	7.55E+01	7.72E+01	5.70E+01	5.83E+01	3.25E+01	3.33E+01
⁵⁴ Mn	1.69E+02	2.26E+02	2.93E+00	3.93E+00	8.89E-04	1.19E-03
⁵⁵ Fe	2.08E+02	2.52E+02	5.48E+01	6.65E+01	3.81E+00	4.63E+00
⁶⁰ Co	2.43E+02	2.99E+02	1.26E+02	1.55E+02	3.37E+01	4.15E+01
⁶³ Ni	2.09E+01	2.37E+01	2.01E+01	2.28E+01	1.87E+01	2.12E+01
⁸⁵ Kr	6.45E+02	6.64E+02	4.67E+02	4.81E+02	2.44E+02	2.52E+02
⁹⁰ Sr	4.82E+03	4.90E+03	4.28E+03	4.35E+03	3.37E+03	3.43E+03
⁹⁰ Y	4.82E+03	4.90E+03	4.28E+03	4.35E+03	3.37E+03	3.43E+03
¹⁰⁶ Ru	4.55E+03	5.31E+03	1.46E+02	1.70E+02	1.51E-01	1.76E-01
¹⁰⁶ Rh	4.55E+03	5.31E+03	1.46E+02	1.70E+02	1.51E-01	1.76E-01
¹²⁵ Sb	1.16E+03	1.27E+03	3.31E+02	3.63E+02	2.71E+01	2.97E+01
^{125m} Te	2.82E+02	3.10E+02	8.07E+01	8.86E+01	6.61E+00	7.25E+00
¹³⁴ Cs	2.41E+03	2.84E+03	4.49E+02	5.29E+02	1.56E+01	1.84E+01
¹³⁷ Cs	1.32E+04	1.33E+04	1.18E+04	1.19E+04	9.36E+03	9.40E+03
^{137m} Ba	1.25E+04	1.26E+04	1.12E+04	1.12E+04	8.85E+03	8.89E+03
¹⁴⁴ Ce	1.51E+03	1.84E+03	1.76E+01	2.14E+01	2.38E-03	2.90E-03
¹⁴⁴ Pr	1.51E+03	1.84E+03	1.76E+01	2.14E+01	2.38E-03	2.90E-03
^{144m} Pr	1.81E+01	2.20E+01	2.11E-01	2.56E-01	2.86E-05	3.47E-05
¹⁴⁷ Pm	8.27E+03	8.55E+03	2.21E+03	2.28E+03	1.57E+02	1.62E+02
¹⁵¹ Sm	5.06E+02	4.95E+02	4.87E+02	4.76E+02	4.51E+02	4.41E+02
¹⁵⁴ Eu	4.98E+02	5.45E+02	3.33E+02	3.64E+02	1.49E+02	1.63E+02
¹⁵⁵ Eu	1.11E+03	1.14E+03	5.50E+02	5.66E+02	1.36E+02	1.40E+02
²³⁸ Pu	1.55E+02	1.64E+02	1.49E+02	1.58E+02	1.38E+02	1.46E+02
²³⁹ Pu	3.43E+02	3.01E+02	3.43E+02	3.01E+02	3.43E+02	3.01E+02
²⁴⁰ Pu	3.69E+02	3.27E+02	3.69E+02	3.27E+02	3.69E+02	3.26E+02
²⁴¹ Pu	1.49E+04	1.36E+04	1.17E+04	1.07E+04	7.23E+03	6.61E+03
²⁴¹ Am	2.15E+02	1.85E+02	3.19E+02	2.80E+02	4.62E+02	4.11E+02
^{242m} Am	1.10E+01	9.80E+00	1.08E+01	9.58E+00	1.03E+01	9.15E+00
²⁴² Am	1.10E+01	9.75E+00	1.07E+01	9.53E+00	1.03E+01	9.11E+00
²⁴² Cm	1.16E+01	1.09E+01	8.88E+00	7.89E+00	8.48E+00	7.53E+00
²⁴⁴ Cm	2.16E+01	2.19E+01	1.78E+01	1.80E+01	1.22E+01	1.23E+01
Total ²	7.92E+04	8.13E+04	4.99E+04	4.94E+04	3.48E+04	3.43E+04
Decay Heat ²	2.31E+02	2.45E+02	1.39E+02	1.38E+02	1.14E+02	1.10E+02

1. Units are curies per assembly, except decay heat which is watts per assembly. Nuclides with activity below 5 curies per assembly are not listed.

2. Totals include effect of all radionuclides present in one assembly.

Table 4.6: Mass of Selected Actinides in Fast Flux Test Facility Types 4.1 and 4.2 Driver Assemblies with 100 MWD/kg Exposure¹

Nuclide	Time Since Reactor Discharge					
	5 Years		10 Years		20 Years	
	Type 4.1	Type 4.2	Type 4.1	Type 4.2	Type 4.1	Type 4.2
235 U	3.21E+01	3.26E+01	3.30E+01	3.34E+01	3.48E+01	3.49E+01
238 U	2.14E+04	2.24E+04	2.14E+04	2.24E+04	2.14E+04	2.24E+04
237 Np	1.16E+01	1.35E+01	1.22E+01	1.39E+01	1.37E+01	1.53E+01
239 Pu	6.38E+03	5.54E+03	6.38E+03	5.54E+03	6.38E+03	5.54E+03
240 Pu	1.51E+03	1.33E+03	1.51E+03	1.32E+03	1.51E+03	1.32E+03
241 Pu	1.24E+02	1.12E+02	9.78E+01	8.78E+01	6.04E+01	5.43E+01
241 Am	5.52E+01	4.80E+01	8.12E+01	7.14E+01	1.17E+02	1.04E+02
Total ²	2.96E+04	2.95E+04	2.96E+04	2.95E+04	2.96E+04	2.95E+04

1. Units are grams per assembly. Actinides with mass below 10 grams per assembly are not listed.
2. Totals include all actinides present in one assembly.

Table 4.7: Mass of Selected Actinides in Fast Flux Test Facility Types 4.1 and 4.2 Driver Assemblies with 150 MWD/kg Exposure¹

Nuclide	Time Since Reactor Discharge					
	5 Years		10 Years		20 Years	
	Type 4.1	Type 4.2	Type 4.1	Type 4.2	Type 4.1	Type 4.2
235 U	2.44E+01	2.39E+01	2.52E+01	2.46E+01	2.68E+01	2.60E+01
238 U	2.05E+04	2.13E+04	2.05E+04	2.13E+04	2.05E+04	2.13E+04
237 Np	1.65E+01	1.88E+01	1.71E+01	1.93E+01	1.89E+01	2.09E+01
239 Pu	5.52E+03	4.85E+03	5.52E+03	4.85E+03	5.52E+03	4.85E+03
240 Pu	1.62E+03	1.43E+03	1.62E+03	1.43E+03	1.62E+03	1.43E+03
241 Pu	1.44E+02	1.32E+02	1.14E+02	1.04E+02	7.01E+01	6.41E+01
242 Pu	3.31E+01	2.90E+01	3.31E+01	2.90E+01	3.31E+01	2.90E+01
241 Am	6.27E+01	5.39E+01	9.29E+01	8.16E+01	1.34E+02	1.20E+02
Total ²	2.79E+04	2.78E+04	2.79E+04	2.78E+04	2.79E+04	2.78E+04

1. Units are grams per assembly. Actinides with mass below 10 grams per assembly are not listed.
2. Totals include all actinides present in one assembly.

DEA-1 -- This assembly was tested to determine the effects of sodium coolant flow on DFA performance. The assembly was only irradiated during the initial reactor start-up and the 2-day full power demonstration run.

HF021 through HF026 -- These assemblies were tested, together with non-fueled assemblies, to provide an early characterization of the core environment prior to the first full cycle of regular full-power operation of the reactor. The design of the assemblies allowed for the removal of up to 21 fuel pins for post irradiation analysis. Assemblies HF021 and HF024 were only irradiated for 24 hours at 4 MW reactor power, while the other four assemblies were only irradiated for 8 days at the full reactor power of 400 MW.

SRF-1 and SRF-2 -- These safety research fuel assemblies were tested to evaluate the use of uranium dioxide blankets in place of the fuel pin reflectors to simulate a fast breeder reactor and to evaluate the use of annular fuel and upper axial blanket pellets as an inherent shutdown mechanism in failure to scram accident conditions. Each assembly contains 25 annular fuel and upper blanket pins.

RNTT-1 -- The radionuclide trap assembly was tested to evaluate the capability for removing certain radioactive corrosion products from the sodium coolant. The assembly is identical to a DFA, except that the fuel pins have a shorter gas plenum to accommodate several rolled nickel sheets serving as the corrosion product trap.

FOTA-1 and FOTA-2 -- These fueled open test assemblies were tested to provide definitive data for verification of assembly thermal hydraulics during initial FFTF startup. They were instrumented DFAs positioned at two selected locations in the reactor core.

CV-1 -- The alternate cladding vendor assembly was tested to evaluate standard specification cladding supplied by three different vendors.

GF001, GF002, WF004, WF005 -- These assemblies were tested to obtain performance data on DFA fuel pins using two different types of grid-spacers rather than wire wrap spacing.

DE1-1R through DE1-6, DE2-1R through DE2-3, DE3-1 through DE3-4, DE4-1 through DE4-5 -- These driver evaluation assemblies were tested to ensure that the DFAs and fuel pin performance were as expected. Some of the assemblies were removed from the reactor core after each of the first four operating cycles and examined to determine actual performance during irradiation. The first number of the assembly designation indicates the cycle after which the assembly was to be removed from the core. The second number differentiates the assemblies within the group to be removed.

RTCB-1 through RTCB-7 -- These run-to-cladding-breach assemblies were tested to determine the lifetime capability of DFA fuel pins, to determine fuel pin breach mechanisms and characteristics, and to develop a statistical data base for development and verification of design procedures

and performance analysis. Assembly RTCB-4 has D9 duct material, five different types of stainless steel pin cladding (148 pins with cold worked 316 SS, 24 with cold worked D9, and 45 pins of United Kingdom stainless alloys M316, P316 and FV548), and a D9 grid rather than wire wrap to space the fuel pins (ANS 1990).

AW-1 -- The advanced wire wrap assembly was tested to evaluate the performance of a DFA containing some (55) wireless, non-adjacent pins, in case DFA performance was found to be limited by the interaction between the fuel pin bundle and the duct.

MW-1 through MW-6 -- These modified wire wrap assemblies were tested to improve coolant mixing, to improve hot spot and hot channel factors, and to determine the potential for vibration-induced wear in the reactor. This was done by determining the effects of different wire wrap pitch [6 inch (15 centimeter) and 12 inch (30 centimeter) axial pitch] and fuel bundle porosities (5 to 40 mil bundle-duct clearances) on assembly performance.

AAD-1 through AAD-7 -- These advanced alloy duct assemblies were tested to evaluate fuel pin bundle and duct interactions, to evaluate duct performance using D9 and HT9 ducts, and to evaluate the use of distributed wireless fuel pins as an alternative fuel pin spacing method (similar to AW-1).

DE-HTD -- The driver evaluation - high temperature fuel assembly was tested to evaluate assembly performance at high cladding temperatures and to define the exposure threshold for fuel pin breach at these temperatures. Higher temperatures were obtained by restricting the coolant flow through the assembly orifice (ANS 1990).

D9-1 through D9-4 -- These assemblies were tested to establish the performance capability of assemblies fabricated with D9 alloy pin cladding and duct materials. The initial objective of a peak pellet exposure in excess of 100 MWD/kg was achieved. The D9-3 was reclassified as a run-to-end-of-life (RTEOL) assembly (ANS 1990).

CRBR-1 -- This assembly was tested to simulate the design and operating conditions of the Clinch River Breeder Reactor and provide performance data and in-reactor behavior information on the fuel assemblies to support final licensing and operation. The plutonium enrichment of the fuel is slightly higher than a type 3.2 assembly, 22.5% versus 22.43%. Another difference is that 211 pins have depleted uranium dioxide axial blanket pellets in place of the fuel pin reflectors. This assembly is associated with the CRBR-3 and the CRBR-5 assemblies which have 33% plutonium enrichment.

C-1 -- This assembly was tested to evaluate the performance of fuel pins which were fabricated to the less stringent, revised Series III specifications recommended for the Secure Automated Fabrication facility. Assembly duct, cladding and wire wrap is made of D9 alloy (ANS 1990).

BUND-1 -- The large bundle porosity assembly was tested to determine the effects of bundle-duct clearance on vibration, wear, and cladding hot

spotting; provide an understanding of the hot spot conditions relative to bundle stiffness and to determine the effects of a reduced pitch on vibration and wear. The assembly is identical to a DFA, except for the use of smaller diameter and shorter axial pitch wire wrap.

DIPRES -- This assembly was tested to demonstrate the performance of mixed-oxide fuel pellets fabricated from direct press spheroidized (DIPRES) feed material. The DIPRES process is conducive to automated processing. The assembly contains 24 fuel pins with DIPRES fabricated pellets. The pellets conform to Type 4.2 driver fuel specifications other than being flat ended.

Table 4.8: Exposure of the Fast Flux Test Facility Test Driver Fuel Assemblies Through Operating Cycle 12B2 (March 1992)
(Page 1 of 3)

<u>Test Assembly</u>	<u>Assembly Type</u>	<u>Average Exposure MWD/KgHM</u>	<u>FFTF Operating Cycle Last Irradiated¹</u>	<u>Serial No.</u>
DEA-1	3.2	2.9	Nat. Circ.	16454
HF021	3.2	< 0.1	Low Power	2049
HF022	3.2	1.7	8 Day Run	2050
HF023	3.2	0.1	8 Day Run	2051
HF024	3.1	< 0.1	Low Power	2052
HF025	3.1	1.0	8 Day Run	2053
HF026	3.1	1.1	8 Day Run	2054
SRF-1	4.1	37.0	2B	8270
SRF-2	4.1	39.0	3	4191
RNTT-1	3.1	53.2	3	8193
FOTA-1	3.2	62.9	3	2100
FOTA-2	3.1	65.6	4	2101
CV-1	3.1	56.1	3	8250
GF001	3.2	61.3	5B	16438
GF002	3.1	71.1	4	8253
WF004	3.2	60.1	5B	16481
WF005	3.1	69.5	4	8251
DE1-1R	3.2	71.6	7C	16425
DE1-2	3.2	21.4	1B	16426
DE1-3	3.1	18.4	1B	8237
DE1-4	3.1	77.6	9C	4130
DE1-5	3.1	47.5	9A2	8218
DE1-6	3.1	22.9	1B	8243
DE2-1R	3.2	44.5	2B	16458
DE2-2	3.2	73.7	6D	16462
DE2-3	3.1	35.2	2B	8248
DE3-1	3.1	61.8	3	16397
DE3-2	3.1	55.2	3	8233
DE3-3	3.2	64.6	5B	16467
DE3-4	3.1	55.8	9A2	4143
DE4-1	3.2	81.6	4	16439
DE4-2	3.1	79.0	12B2	4101
DE4-3	3.1	64.9	8C	4116
DE4-4	3.1	83.1	7C	4171
DE4-5	3.1	67.9	10A4	4104
RTCB-1	3.2	63.4	3	16450
RTCB-2	3.2	76.8	5B	16444
RTCB-3	3.1	78.6	5B	8242
RTCB-4	4.2	86.4	6D	2060
RTCB-5	3.2	58.5	3	16416
RTCB-6	3.1	85.8	5B	8221

Table 4.8 continued: Exposure of the Fast Flux Test Facility Test Driver Fuel Assemblies Through Operating Cycle 12B2 (March 1992)
(Page 2 of 3)

<u>Test Assembly</u>	<u>Assembly Type</u>	<u>Average Exposure MWD/KgHM</u>	<u>FFTF Operating Cycle Last Irradiated¹</u>	<u>Serial No.</u>
RTCB-7	3.1	70.8	5B	4123
MW-1	4.2	71.3	5B	16518
MW-2	4.2	58.7	4	16494
MW-3	4.2	78.4	6D	16487
MW-4	4.2	78.9	6C	16490
MW-5	4.2	70.3	5B	16515
MW-6	4.2	70.8	5B	16510
AAD-1	3.1	68.5	5B	4148
AAD-2	4.2	65.4	7C	2140
AAD-3	3.2	68.4	5B	16435
AAD-4	4.2	80.9	4	16500
AAD-5	4.2	73.9	4	16489
AAD-6	4.1	96.0	5B	8256
AAD-7	4.2	68.7	7C	2141
AW-1	3.2	67.3	4	16440
D9-1	4.2	78.8	4	16502
D9-2	4.2	127.4	7A	16505
D9-3	4.1	145.3	10B	4215
D9-4	3.2	102.4	6D	16523
CRBR-1	3.2	54.7	6D	16501
C-1	3.2/4.2	88.1	6D	16497
BUND-1	4.2	54.0	4	16491
DE-HTD	4.1	59.8	6D	2067
DIPRES	4.2	82.7	10C3	16514

Notes

1. The last FFTF operating cycle during which the assembly was irradiated. "Nat. Circ." (for natural circulation), "8 Day Run", and "Low Power" (low power characterization test, were part of the initial FFTF operating periods where equipment operation was verified and certain operating capabilities were determined. For cycles 1 through 8, periods of (nominal) 100 days of effective full power were considered a single cycle. Core refueling and reactor maintenance were done between cycles. Letter designations were used to differentiate operating periods of a single cycle interrupted by short shutdowns, e.g. cycle 8 was interrupted twice resulting in cycle designations 8A, 8B, and 8C. For cycles 9 through 12, periods of (nominal) 300 effective full power days (EFPD) were considered a single cycle. Annual maintenance was done between cycles. Letter designations were used to differentiate the three nominal 100 EFPD periods within a cycle. Core refueling and reactor maintenance were done between these "part cycles." Additional number designations were used to differentiate operating periods of a single 100 EFPD part

Table 4.8 continued: Exposure of the Fast Flux Test Facility Test Driver Fuel Assemblies Through Operating Cycle 12B2 (March 1992)
(Page 3 of 3)

cycle interrupted by short shutdowns, e.g. cycle 10A was interrupted three times resulting in cycle designations 10A1, 10A2, 10A3, and 10A4. The dates on which the cycles ended are:

Cycle	Date	Cycle	Date	Cycle	Date
Nat. Circ.	3/81	5B	11/84	9C	10/87
8 Day Run	11/81	6C	6/85	10A4	5/88
Low Power	1/82	6D	6/85	10B	10/88
1B	11/82	7A	12/85	10C3	3/89
2B	5/83	7C	1/86	12B2	3/92
3	10/83	8C	7/86		
4	4/84	9A2	2/87		

Table 4.9: Non-Standard Duct and Pin Cladding Materials in the Fast Flux Test Facility Test Driver Fuel Assemblies

<u>Test Assembly</u>	<u>Assembly Type</u>	<u>Materials Assembly Duct</u>	<u>Pin Cladding</u>	<u>Serial No.</u>
RTCB-4	4.2	D9	SS ¹	2060
AAD-1	3.1	D9	316	4148
AAD-2	4.2	HT9	316	2140
AAD-3	3.2	D9	316	16435
AAD-4	4.2	D9	316	16500
AAD-5	4.2	D9	316	16489
AAD-6	4.1	D9	316	8256
AAD-7	4.2	HT9	316	2141
D9-1	4.2	D9	D9	16502
D9-2	4.2	D9	D9	16505
D9-3	4.1	D9	D9	4215
D9-4	3.2	D9	D9	16523
C-1	3.2/4.2	D9	D9	16497

Note 1. RTCB-4 fuel pins are clad with one of 5 different stainless steels.

Table 4.10: Fast Flux Test Facility Test Driver Fuel Assemblies with Fuel Pins Removed

<u>Test Assembly</u>	<u>Serial No.</u>	<u>Number of pins Remaining</u>
HF021	2049	196
HF022	2050	196
HF023	2051	196
HF024	2052	196
HF025	2053	196
HF026	2054	196

Table 4.11: Disassembled Fast Flux Test Facility Test Driver Fuel Assemblies

<u>Test Assembly Name</u>	<u>Serial No.</u>	<u>Number of pins at FSF</u>	<u>Storage Containers</u>	
			<u>Serial No.</u>	<u>Number of Pins</u>
DEA-1	16454	195	ID69002	44
			ID69042	42
			ID69043	109
DE1-2	16426	207	ID69005	32
			ID69036	109
			ID69042	66
DE1-6	8243	197	ID69002	15
			ID69007	109
			ID69015	73
DE2-1R	16458	182	ID69003	109
			ID69005	23
			ID69015	36
DE3-1	16397	204	ID69028	14
			ID69002	1
			ID69005	26
			ID69010	13
			ID69011	55
			ID69013	55
			ID69017	54
DE3-2	8233	206	ID69005	29
			ID69010	82
			ID69028	95
DE4-1	16439	182	ID69005	21
			ID69017	1
			ID69018	55
RTCB-7	4123	181	ID69021	105
			ID69025	70
			ID69031	96
AAD-5	16489	208	ID69032	15
			ID69023	133
			ID69024	18
D9-1	16502	202	ID69038	57
			ID69002	18
			ID69014	109
D9-2	16505	176	ID69025	75
			ID69038	69
			ID69044	107
D9-4	16523	187	ID69032	52
			ID69035	135
			ID69005	27
C-1	16497	204	ID69022	98
			ID69032	79
			ID69002	30
BUND-1	16491	216	ID69030	154
			ID69031	32

4.2.3 Test Fuel Assemblies

The TFAs contain fuel pins with different fuel compositions than the DFAs, although a few also contain some DFA fuel pins. The TFAs are also generally made from different materials, however, almost all have the same general structural features to interface with the design of the FFTF Reactor. The purposes of the TFAs are varied, but most support the fuel or material requirements for large scale breeder reactors.

Tables 4.12 through 4.14 summarize information on the TFAs. Table 4.12 lists the assembly exposure, fuel type, and pre-irradiation enrichment, as well as the FFTF operating cycle in which the assembly was last irradiated. Table 4.13 lists the duct and cladding material for the 54 TFAs, as well as the general reference drawing of each assembly.

Fuel pins were removed from some TFAs and shipped off site for examination. Some of these fuel pins were either returned whole in Ident 69 containers or as pieces in storage casks described in Section 6.2 of this report. Most of the remaining fuel pins will eventually be returned to Hanford. The other fuel pins from these TFAs, which were not shipped off site, are stored in Ident 69 containers, except those from one TFA (DEA-2) which remain in the assembly. Shipments of fuel pins from the last three disassembled TFAs have been delayed due to the status of FFTF program, and may not occur at all (WHC 1994c). Table 4.14 lists the TFAs from which fuel pins were removed, the original number of fuel pins in the TFA, the number at FSF and the distribution of fuel pins among Ident 69 containers (WHC 1992h, WHC 1992i).

The specific purposes of the TFAs are briefly described below. The plutonium and uranium enrichments described are all pre-irradiation enrichments. More information is available in the test assembly records of the Manager of Irradiation Test Management, stored in the 400 Area at the Hanford Site.

DEA-2 -- This assembly was tested to determine thermal performance capabilities of the DFAs. The assembly was subjected to a rapid rise in reactor power followed by 10 minutes of reactor operation at full power to minimize fuel restructuring. This created a maximum center fuel temperature and incipient fuel melting. The assembly contained all four driver fuel types -- 3.1, 3.2, 4.1, 4.2 -- as well as some type 4.2 pins with 3.7% ^{235}U enrichment (ANS 1990).

AB-1 -- This assembly was tested to characterize the behavior of a mixed oxide axially blanketed fuel and cladding system under prototypical operating conditions. The assembly contains 154 type 4.2 pins and 63 mixed oxide blanket pins, including 10 with fuel made using a co-precipitation process (ANS 1990).

WB018 and WB042 -- These assemblies were tested to determine blanket assembly performance under typical fast reactor conditions. Each assembly contains 61 wire wrapped, 0.503 inch (1.28 centimeter) diameter blanket rods; 52 rods contain depleted uranium dioxide, while 9 rods contain thorium-232 dioxide. The blanket rods are enclosed in a standard duct assembly, except for a flow mixing device at the top of the assembly to permit higher temperature operation.

DE-9 -- This assembly was tested to provide additional data on sustained driver operation at high power levels. The assembly contained 157 standard type 4.2 pins and 60 type 4.2 pins with 3.7% ^{235}U enrichment. The ^{235}U enriched pins are divided into 12 equal sets providing a variety of fuel-to-cladding gap sizes, fuel densities and oxygen-to-metal ratios (ANS 1990).

SRF-3 and SRF-4 -- These safety research fuel assemblies were tested to evaluate fast flux irradiated fuel pins having high fissile content for subsequent safety tests in the DOE Fast Reactor Safety program. Both assemblies contain two radial regions separated by a hollow-walled hexagonal duct enclosing a sodium layer. The assembly central section contains 91 highly enriched, mixed-oxide fuel pins, in four distinct fuel enrichment zones (42-89% ^{235}U). Pre-irradiation plutonium enrichment is 25%. The assemblies contain a 6.5 inch (16.5 centimeter) lower axial blanket and a 14.0 inch (35.6 centimeter) upper axial blanket. The outer regions contain boron carbide filter pins to provide neutron shielding. The two assemblies are identical, except that SRF-4 contains less instrumentation.

ACN-1 -- This assembly was tested to develop information on helium- and sodium-bonded mixed-carbide fuel pins with full length fuel columns at prototypic fluence and exposure conditions. Additionally, it tests the relative effects of 20% cold worked 316 SS and 25% cold worked D9 cladding on the carbide fuel pins. The assembly contains 18 sodium-bonded and 19 helium-bonded carbide fuel pins, enclosed in a 316 SS inner duct similar to the SRF-3. The outer region contains 90 standard driver fuel pins and is enclosed by a D9 duct.

FC-1 -- This assembly was tested to establish performance characteristics of a full size carbide fuel assembly. The assembly contains 91 large diameter [0.37 inch (0.94 centimeter)], D9 clad, wire wrapped, helium bonded fuel pins. The plutonium enrichment is 21.4 % in uranium carbide, with 6.5 inch (16.5 centimeter) top and bottom blankets.

CRBR-3 and CRBR-5 -- These assemblies were tested to simulate the design and operating conditions of the Clinch River Breeder Reactor and provide performance data and in-reactor behavior information on the fuel assemblies to support final licensing and operation. The assemblies have depleted uranium dioxide axial blanket pellets in place of the fuel pin reflectors. The duct, cladding, and wire wrap of the CRBR-3 is made of

316 SS, while that of the CRBR-5 is D9. The fuel is mixed plutonium-uranium oxide, with 33% plutonium enrichment. These tests are associated with the CRBR-1 assembly which has 22.5% plutonium enrichment. There are no CRBR-2 and CRBR-4 assemblies.

FO-1 and FO-2 -- These advance oxide fuel assemblies were tested to evaluate different fuel specifications for advanced oxide fuel designs for a commercial-scale liquid metal fast breeder reactor. The FO-1 is made of all D9 components, while the FO-2 is made of all HT9 components. Both assemblies contained 169 large diameter pins [0.27 inch (0.69 centimeter)]. The FO-1 contained fuel with two plutonium enrichments (19.0% and 22.4%) in natural uranium. The FO-2 contained three plutonium enrichments in depleted uranium, 10 pins with 18.0%, 139 pins with 22.5%, and 20 pins with 26.0% (ANS 1990).

ACO-1 through ACO-16 and ABA-1 through ABA-6 -- These advanced oxide fuel assemblies were tested to demonstrate the capability of a mixed plutonium and uranium oxide fuel and blanket system to achieve a three year life in a prototypic large scale breeder reactor environment and to evaluate the HT9 components in a core demonstration experiment (CDE). All assemblies have HT9 ducts. All have HT9 pin cladding, except the ACO-2 which has D9 pin cladding. The ACO-1 and the ACO-2 have some 316 SS components.

The purpose of assemblies ACO-1 and ACO-2 was to evaluate the HT9 and D9 material cladding within HT9 ducts. Six of the ACO-1 pins have subsequently been installed in the MFF-1A assembly to receive additional irradiation. The purpose of assemblies ACO-3 and ACO-4 was to evaluate entire fuel assemblies of HT9 duct, cladding, and wire wrap. The ACO-3 was operated at high temperature conditions, with the ACO-4 operated at nominal conditions. The purpose of assemblies ACO-5 through ACO-14 and blanket assemblies ABA-1 through ABA-6 was to simulate the large scale breeder reactor environment. The purpose of assemblies ACO-15 and ACO-16 is to provide performance data at low power, but at temperatures and exposures similar to the lead high-power CDE assemblies. Some of the ACO assemblies were being used as driver assemblies since that their test period was over.

All the ACO assemblies contain 169 large fuel pins [(0.27 inch (0.69 centimeter) diameter)]. The ACO-1 and ACO-2 fuel contains 22.6% plutonium enrichment in natural uranium and depleted uranium respectively. The ACO-3 and ACO-4 fuel contains 29.0% plutonium enrichment in depleted uranium. The ACO-5 through ACO-16 fuel contains 30.0% plutonium enrichment in depleted uranium. All the ABA assemblies contain ninety-one 0.39 inch (1 centimeter) diameter pins, with 49 inches (74 centimeters) of depleted uranium dioxide pellets (ANS 1990).

PO-1, PO-2, PO-4, PO-5 -- These advanced oxide fuel assemblies were tested to demonstrate fuel assembly qualification for a two year lifetime commercial scale, liquid metal reactor. Each assembly contains 169 large

diameter pins and all D9 alloy components. The fuel enrichment is 22.5% plutonium in depleted uranium. There is no PO-3 assembly (ANS 1990).

MFF-1 and MFF-1A -- These assemblies contain metal, uranium oxide, and mixed oxide fuel pins to allow a direct comparison of performance. The MFF-1 contains 153 CDE fuel pins (30% plutonium enrichment in depleted uranium), 10 fuel pins of enriched uranium dioxide (0% plutonium and 27.7% ²³⁵U enrichment), and 6 pins of 90% uranium-10% zirconium metal fuel (18.5% ²³⁵U enrichment). The MFF-1A contained six pins previously irradiated in ACO-1, plus, 143, 14 and 6, respectively, of the same types of pins as in MFF-1 (ANS 1990).

MFF-2 through MFF-6 -- These assemblies were part of the test series to qualify metal fuel for FFTF Series III fuel. The purpose of the MFF-2 is to obtain early verification of the performance of 90% uranium-10% zirconium metal alloy fueled pins. The MFF-3 is designed to operate at higher than normal temperature. The MFF-4, MFF-5, and MFF-6 use the new Series III fuel design. All assemblies contain 169 large fuel pins [0.27 inch (0.69 centimeter) diameter] and HT9 duct and cladding. The MFF-2, MFF-3, and MFF-4 contain 32.4% enriched uranium, while the MFF-5 and MFF-6 contain 31.0% enriched uranium (ANS 1990).

MFA-1, MFA-2, and MBA-1 -- The assemblies were tested to demonstrate the irradiation performance and lifetime capability of Japanese MONJU fuel and blanket assemblies under steady-state irradiation to high burn-up. The fuel enrichment in the MFA-1 and the MFA-2 is 31.2% plutonium in depleted uranium. The cladding, wire wrap, and duct are fabricated from Japanese 20% cold worked 316 SS for the MFA-1 and Japanese advanced austenitic SS for the MFA-2. Both assemblies contain 169 pins. The MBA-1 contains depleted uranium oxide in sixty-one 0.457 inch (1.16 centimeter) diameter fuel pins. The MBA-1 is fabricated of the same materials as the MFA-1. The MONJU duct tubes have smaller dimensions than the FFTF ducts.

UO-1 -- This uranium oxide fuel assembly is the first in a series which was to be tested to confirm the performance of a future uranium dioxide fueled FFTF core (Series IV). The purpose of the UO-1 is to provide fuel pins for transient testing in comparison with mixed plutonium-uranium oxide. The assembly contains four different types of fuel, seven pins of 42.9% enriched uranium oxide, 126 pins of type 3.1, 54 pins of type 4.1, and 30 pins of S-4 type (plutonium-uranium oxide with 29% Pu enrichment).

IFR-1 -- The integral fast reactor assembly was tested to support development of metal fuel. The test assembly contained 169 sodium-bonded fuel pins representing three metallic fuel alloys. The assembly contained 19 pins of 82% uranium-8% plutonium-10% zirconium alloy, 18 of pins of 71% uranium-19% plutonium-10% zirconium alloy, and 132 of pins of 90% uranium-10% zirconium binary alloy. The ²³⁵U enrichment of the uranium within these fuel alloys is 19%, 5.4%, and 25.5%, respectively. The 36-inch (91 centimeter) fuel section contains only two or three fuel

slugs. The duct, cladding, and wire-wrap is D9 stainless steel (ANS 1990).

AC-3 -- This assembly was tested in cooperative effort of the United States and Swiss governments and was part of the advanced liquid metal fast breeder reactor fuel program. The test compared performance of 66 pins containing pelletized fuel with that of 25 sphere-pac fuel pins at typical conditions of the breeder reactor. The pins are D9-clad, wire-wrapped, and were housed in a D9 duct. The fuel is mixed plutonium-uranium carbide with plutonium enrichments of 19.1% for the sphere-pac fuel and 19.7% for the pelletized fuel.

FSP-1 and FSP-1R -- These space program assemblies are the first in a series that was to be tested. They were tested to provide irradiation performance data on uranium nitride fuel pins operating at high temperatures and extended lifetimes to support space applications. The fuel pins are clad with a 99% niobium-1% zirconium alloy (Nb-1Zr). Although the assembly exteriors match that of a DFA, the internals are very unique. The fuel is 21.0% enriched uranium nitride in four inch (10.1 centimeter) and six inch (15 centimeter) long segments. The FSP-1 assembly has been disassembled. Twenty-two of its pins and 16 fresh pins comprise the fuel of FSP-1R.

Table 4.12: Exposure of the Fast Flux Test Facility Test Fuel Assemblies
Through Operating Cycle 12B2 (March 1992)
(Page 1 of 4)

<u>Test</u>	<u>Fuel Pins</u>	<u>Fuel Type¹</u>	<u>Average Exposure MWD/KgHM</u>	<u>FFTF Operating Cycle Last Irradiated²</u>	<u>Serial No.</u>
DEA-2	(6) (166) (12) (18) (15)	3.1 3.2 4.1 4.2 MOX ³	< 0.1	PWR to Melt	2110
AB-1	(154) (63)	4.2 MOX ⁴	7.2	1A	2114
WB018	(52) (9)	UO ⁵ ThO ⁶	-	6A	4183
WB042	(52) (9)	UO ⁵ ThO ⁶	-	6D	4184
DE-9	(157) (60)	4.2 MOX ³	73.0	5A	16527
SRF-3	(91)	MOX ⁷	37.0	4	2075
SRF-4	(91)	MOX ⁷	44.2	6D	2076
ACN-1	(90) (37)	3.2 Carbide ⁸	67.5	6B	2111
FC-1	(91)	Carbide ⁹	59.5	8A	2068
CRBR-3	(217)	MOX ¹⁰	60.3	4	8272
CRBR-5	(217)	MOX ¹⁰	77.0	6D	2074
FO-1	(169)	MOX ¹¹	46.9	6A	2073
FO-2	(10) (139) (20)	MOX ¹² MOX ¹³ MOX ¹⁴	46.8	8A	2143
ACO-1	(169)	MOX ¹⁵	93.9	8A	8252
ACO-2	(169)	MOX ¹⁶	103.4	10A4	2063
ACO-3	(169)	MOX ¹⁷	197.5	12B2	2144
ACO-4	(169)	MOX ¹⁷	199.4	12B2	2064
ACO-5	(169)	MOX ¹⁸	153.1	12B2	2145
ACO-6	(169)	MOX ¹⁸	134.6	11B1	2146
ACO-7	(169)	MOX ¹⁸	111.7	10C3	2147
ACO-8	(169)	MOX ¹⁸	148.2	12B2	2148
ACO-9	(169)	MOX ¹⁸	139.8	12B2	2159
ACO-10	(169)	MOX ¹⁸	140.1	11B1	2150
ACO-11	(169)	MOX ¹⁸	140.8	11B1	2151
ACO-12	(169)	MOX ¹⁸	147.0	12B2	2152
ACO-13	(169)	MOX ¹⁸	193.0	12B2	2153
ACO-14	(169)	MOX ¹⁸	133.9	11B1	2154
ACO-15	(169)	MOX ¹⁸	96.6	12B2	2149
ACO-16	(169)	MOX ¹⁸	78.0	12B2	2156
ABA-1	(91)	UO ⁵	16.4	10B	2160

Table 4.12 continued: Exposure of the Fast Flux Test Facility Test Fuel Assemblies Through Operating Cycle 12B2 (March 1992)
(Page 2 of 4)

<u>Test</u>	<u>Fuel Pins</u>	<u>Fuel Type¹</u>	<u>Average Exposure MWD/KgHM</u>	<u>FFTF Operating Cycle Last Irradiated²</u>	<u>Serial No.</u>
ABA-2	(91)	UO ⁵	29.1	11B1	2161
ABA-3	(91)	UO ⁵	32.1	11B1	2162
ABA-4	(91)	UO ⁵	19.5	10C3	2163
ABA-5	(91)	UO ⁵	29.5	11B1	2164
ABA-6	(91)	UO ⁵	8.2	10A1	2165
PO-1	(169)	MOX ¹³	88.3	10A3	2065
PO-2	(169)	MOX ¹³	68.6	9A1	2066
PO-4	(169)	MOX ¹³	80.0	10C2	2069
PO-5	(169)	MOX ¹³	78.0	10C2	2070
MFF-1	(153)	MOX ¹⁴	105.4	11B1	2158
	(10)	UO ¹⁹			
	(6)	Metal ²⁰			
MFF-1A	(143)	MOX ¹⁴	46.1	10A4	2155
	(14)	UO ¹⁹			
	(6)	Metal ²⁰			
	(6)	ACO-1 ¹⁵	117.1		
MFF-2	(169)	Metal ²¹	114.3	12B2	2168
MFF-3	(169)	Metal ²¹	114.2	12B2	2169
MFF-4	(169)	Metal ²¹	112.5	12B2	2088
MFF-5	(169)	Metal ²²	85.1	12B2	2089
MFF-6	(169)	Metal ²²	77.2	12B2	2090
MFA-1	(169)	MOX ²³	119.6	12A2	2170
MFA-2	(169)	MOX ²³	127.8	12B2	2171
MBA-1	(61)	UO ⁵	12.3	11C	2172
UO-1	(126)	3.1	34.3	11B2	2116
	(54)	4.1			
	(30)	MOX ²⁴			
	(7)	UO ²⁵			
IFR-1	(132)	Metal ²⁶	76.7	10B	16492
	(19)	Metal ²⁷			
	(18)	Metal ²⁸			
AC-3	(66)	Carbide ²⁹	66.7	10B	2071
	(25)	Carbide ³⁰			
FSP-1	(38)	Nitride ³¹	15.4	10C1	2086
FSP-1R	(16)	Nitride ³¹	41.7	12A2	2087
	(22)	Nitride ³¹	-		

Notes

1. Plutonium and uranium enrichments listed in the footnotes are all pre-irradiation enrichments.

Table 4.12 continued: Exposure of the Fast Flux Test Facility Test Fuel Assemblies Through Operating Cycle 12B2 (March 1992)
(Page 3 of 4)

2. The last FFTF operating cycle during which the assembly was irradiated. The "PWR to Melt" (for power to melt) operating period was part of the initial FFTF operating period where equipment operation was verified and certain operating capabilities were determined. For cycles 1 through 8, periods of (nominal) 100 days of effective full power were considered a single cycle. Core refueling and reactor maintenance were done between cycles. Letter designations were used to differentiate operating periods of a single cycle interrupted by short shutdowns, e.g. cycle 8 was interrupted twice resulting in cycle designations 8A, 8B, and 8C. For cycles 9 through 12, periods of (nominal) 300 effective full power days (EFPD) were considered a single cycle. Annual maintenance was done between cycles. Letter designations were used to differentiate the three nominal 100 EFPD periods within a cycle. Core refueling and reactor maintenance were done between these "part cycles." Additional number designations were used to differentiate operating periods of a single 100 EFPD part cycle interrupted by short shutdowns, e.g. cycle 10A was interrupted three times resulting in cycle designations 10A1, 10A2, 10A3, and 10A4. The dates on which the cycles ended are:

Cycle	Date	Cycle	Date	Cycle	Date
PWR to Melt	12/81	8A	4/86	10C2	1/89
1A	5/82	9A1	10/86	10C3	3/89
4	4/84	10A1	1/88	11B1	4/90
5A	8/84	10A3	3/88	11B2	10/90
6A	1/85	10A4	5/88	11C	3/91
6B	4/85	10B	10/88	12A2	9/91
6D	6/85	10C1	1/89	12B2	3/92

3. Plutonium-uranium oxide, 29.28% Pu/(Pu + U) with 3.7% ²³⁵U.
4. Mixed plutonium-uranium oxide.
5. Depleted uranium oxide.
6. ²³²Th thorium oxide.
7. Plutonium-uranium oxide, 25% Pu/(Pu + U), with the uranium containing between 42 and 89% ²³⁵U.
8. Plutonium-uranium carbide.
9. Plutonium-uranium carbide, with 21.4% Pu/(Pu + U).
10. Plutonium-uranium oxide, 33.0% PuO₂/(PuO₂ + UO₂), with depleted uranium.
11. Plutonium-uranium oxide, 19.0% and 22.4% Pu/(Pu + U), with natural uranium.
12. Plutonium-uranium oxide, 18.0% Pu/(Pu + U), with depleted uranium.
13. Plutonium-uranium oxide, 22.5% Pu/(Pu + U), with depleted uranium.
14. Plutonium-uranium oxide, 26.0% Pu/(Pu + U), with depleted uranium.
15. Plutonium-uranium oxide, 24.6% Pu/(Pu + U), with natural uranium.
16. Plutonium-uranium oxide, 22.5% Pu/(Pu + U), with natural uranium.

Table 4.12 continued: Exposure of the Fast Flux Test Facility Test Fuel Assemblies Through Operating Cycle 12B2 (March 1992)
(Page 4 of 4)

17. Plutonium-uranium oxide, 29.0% Pu/(Pu + U), with depleted uranium.
18. Plutonium-uranium oxide, 30.0% Pu/(Pu + U), with depleted uranium.
19. 27.2% ²³⁵U enriched uranium oxide.
20. 18.5% ²³⁵U enriched, 90% uranium/10% zirconium metal.
21. 32.4% ²³⁵U enriched, 90% uranium/10% zirconium metal.
22. 31.0% ²³⁵U enriched, 90% uranium/10% zirconium metal.
23. Plutonium-uranium oxide, 31.2% Pu/(Pu + U), with depleted uranium.
24. Plutonium-uranium oxide, 29% Pu/(Pu + U).
25. 42.9% ²³⁵U enriched uranium oxide.
26. 25.2% ²³⁵U enriched, 90% uranium/10% zirconium metal.
27. 18.0% ²³⁵U enriched, 82% uranium/8% plutonium/10% zirconium metal.
28. 5.4% ²³⁵U enriched, 71% uranium/19% plutonium/10% zirconium metal.
29. Plutonium-uranium carbide with 19.7% Pu/(Pu + U).
30. Plutonium-uranium carbide with 19.1% Pu/(Pu + U).
31. 21% ²³⁵U enriched uranium nitride.

Source: (WHC 1992f)

Table 4.13: Duct and Pin Cladding Material in the Fast Flux Test Facility Test
Fuel Assemblies
(Page 1 of 2)

<u>Test</u>	<u>Materials</u>		<u>General</u>	<u>Serial</u>
	<u>Assembly</u>	<u>Pin</u>	<u>Reference Drawing</u>	<u>No.</u>
	<u>Duct</u>	<u>Cladding</u>		
DEA-2	316	316	H4-21500 Rev 9	2110
AB-1	316	316	909E041 ¹	2114
WB018	316	316	766J708 ²	4183
WB042	316	316	766J981	4184
DE-9	316	316	H3-44979	16527
SRF-3	316 ³	316	H4-65103	2075
SRF-4	316 ³	316	H4-65103	2076
ACN-1	D9/316 ³	316/D9	H4-61485	2111
FC-1	D9	D9	90Y219562 Sub B ⁴	2068
CRBR-3	316	316	767J102 Sub12 ²	8272
CRBR-5	D9	D9	767J102 Sub14 ²	2074
FO-1	D9	D9	H3-46971	2073
FO-2	HT9	HT9	972E179 ¹	2143
ACO-1	HT9	HT9	H3-48574	8252
ACO-2	HT9	D9	H3-52610	2063
ACO-3	HT9	HT9	H3-53108	2144
ACO-4	HT9	HT9	H3-53108	2064
ACO-5	HT9	HT9	H3-53216 Rev 0	2145
ACO-6	HT9	HT9	H3-53216 Rev 0	2146
ACO-7	HT9	HT9	H3-53216 Rev 0	2147
ACO-8	HT9	HT9	H3-53216 Rev 0	2148
ACO-9	HT9	HT9	H3-53216 Rev 0	2159
ACO-10	HT9	HT9	H3-53216 Rev 0	2150
ACO-11	HT9	HT9	H3-53216 Rev 0	2151
ACO-12	HT9	HT9	H3-53216 Rev 0	2152
ACO-13	HT9	HT9	H3-53216 Rev 0	2153
ACO-14	HT9	HT9	H3-53216 Rev 0	2154
ACO-15	HT9	HT9	H3-57542	2149
ACO-16	HT9	HT9	H3-57511	2156
ABA-1	HT9	HT9	981E120 ¹	2160
ABA-2	HT9	HT9	981E120 ¹	2161
ABA-3	HT9	HT9	H3-53177	2162
ABA-4	HT9	HT9	H3-53177	2163
ABA-5	HT9	HT9	H3-53177	2164
ABA-6	HT9	HT9	H3-53177	2165
PO-1	D9	D9	972E164 ¹	2065
PO-2	D9	D9	972E164 ¹	2066
PO-4	D9	D9	972E164 ¹	2069
PO-5	D9	D9	972E164 ¹	2070
MFF-1	HT9	HT9	H3-57481	2158
MFF-1A	HT9	HT9	H3-57446	2155

Table 4.13: Duct and Pin Cladding Material in the Fast Flux Test Facility Test Fuel Assemblies
(Page 2 of 2)

<u>Test</u>	<u>Materials</u>	<u>General</u>	<u>Serial</u>
	<u>Assembly</u>	<u>Pin</u>	<u>No.</u>
	<u>Duct</u>	<u>Cladding</u>	
MFF-2	HT9	HT9	H3-57534 2168
MFF-3	HT9	HT9	H3-57534 2169
MFF-4	HT9	HT9	H3-58215 2088
MFF-5	HT9	HT9	H3-58254 2089
MFF-6	HT9	HT9	H3-58254 2090
MFA-1	M316 ⁵	M316 ⁵	H3-58011 2170
MFA-2	AA ⁶	AA ⁶	H3-58009 2171
MBA-1	M316 ⁵	M316 ⁵	H3-58027 2172
UO-1	316	316/D9	H3-60687 2116
IFR-1	D9	D9	M4000-0184-DE ⁷ 16492
AC-3	D9	D9	90Y-219562-E3 ³ 2071
FSP-1	316	Nb-1Zr ⁸	H3-56342 2086
FSP-1R	316	Nb-1Zr ⁸	H3-57750 2087

- Notes
1. General Electric drawing
 2. Westinghouse Advanced Reactors Division drawing
 3. Assemblies have an inner and outer duct
 4. Los Alamos National Laboratory drawing
 5. Japanese stainless steel
 6. Japanese austenitic steel
 7. Argonne National Laboratory drawing
 8. Binary metal alloy of 99% niobium and 1% zirconium

Source: (ANS 1990)

Table 4.14: Disassembled Fast Flux Test Facility Test Fuel Assemblies
(As of June 1994)

<u>Test Assembly Name</u>	<u>Serial No.</u>	<u>Number of pins</u>		<u>Storage Containers</u>	
		<u>Original</u>	<u>Current</u>	<u>Serial No.</u>	<u>No. of Pins</u>
DEA-2	2110	217	186	DEA-2 ¹	186
DE-9	16527	217	177	ID69-22	46
				ID69-26	131
FO-1	2073	169	169	ID69-01	91
				ID69-08	78
FO-2	2143	169	129	ID69-44	16
				ID69-45	113
ACO-1	8252	169	144	ID69-37	128
				ID69-44	16
ACO-3	2144	169	169	ID69-40	126
				ID69-27	12
				PB40-11 ²	22
				PB19-05 ³	9
PO-1	2065	169	152	ID69-24	103
				ID69-39	49
MFA-1	2170	169	151	ID69-12	56
				ID69-27	95
MFA-2	2171	169	169	ID69-33	151
				PB40-11 ²	18
MBA-1	2172	61	61	ID69-29	51
				PB19-05 ³	10
MFF-1A	2155	169	129	ID69-08	44
				ID69-39	85
AC-3	2071	91	72	ID69-46	72
IFR-1	16492	169	115	ID69-04	115
FSP-1	2086	38	0	none	0

Notes 1. DEA-2 had fuel pins removed, but was not disassembled.

2. PB19-05 and PB40-11 are fuel pin baskets that are intended to be shipped offsite.

4.3 FUEL INVENTORY

There are 8.6 metric tons (MT) uranium and 2.4 MT plutonium in the irradiated FFTF DFAs, TDFAs, and TFAs at the Hanford Site, including that in Ident 69 containers (WHC 1994d, WHC 1994e). Table 4.15 shows the breakdown of fuel between the DFAs, TDFAs, and TFAs. These quantities only include fuel currently stored at FFTF facilities.

Table 4.15: Fast Flux Test Facility Fuel Inventory

<u>Fuel Type</u>	<u>Uranium</u> MT	<u>Plutonium</u> MT	<u>Total</u> MT
DFAs	5.2	1.6	6.8
TDFAs	1.7	0.5	2.2
<u>TFAs</u>	<u>1.7</u>	<u>0.3</u>	<u>2.0</u>
Total	8.6	2.4	11.0

The plutonium and uranium content of the individual assemblies varies depending on the assembly exposure and their pre-irradiation content. Table 4.16 shows the ranges of uranium and plutonium content for the four different types of fuel in the DFAs and TDFAs, and for the variety of TFAs.

Table 4.16: Post-Irradiation Fuel Content of the Fast Flux Test Facility Assemblies

<u>Assembly Type</u>	<u>Uranium</u> Kg	<u>Plutonium</u> Kg	<u>Total</u> Kg
3.1	23.1-25.9	7.5-9.1	31.3 ¹ -35.0
3.2	24.4-27.3 ²	6.4-7.5	31.1 ¹ -34.8 ²
4.1	22.5-25.0 ²	7.2-9.8	29.7-34.8 ²
4.2	24.6-26.3	7.0-8.4	31.6-34.7
TFAs	1.8-63.4	0.0-9.7	1.8-65.0 ¹

Note 1. Minimum and maximum uranium and plutonium contents shown are not necessarily for the same assembly. Therefore their sum does not always equal the values shown as the range of the total uranium plus plutonium content.

2. One type 3.2 assembly and two type 4.1 assemblies have significantly higher uranium contents, but have not been included in the ranges to avoid misrepresenting the typical uranium content. Assembly CRBR-1 holds 36.1 kilograms of uranium and assemblies SRF-1 and SRF-2 hold 41.4 kilograms of uranium, due to the replacement of the standard high nickel alloy reflectors with UO₂ pellets in 211, 217, and 217 fuel pins, respectively. In addition, the SRF -1 and SRF-2 contain 25 fuel pins filled with 100% UO₂ fuel.

4.4 FUEL STORAGE

The FFTF assemblies are stored within several different facilities. Irradiated assemblies are stored at the 400 Area of Hanford within the reactor core, the in-vessel storage (IVS), the interim decay storage (IDS), and the fuel storage facility (FSF). Figure 4.6 shows the relative location of these storage facilities. Some irradiated fuel pins have been shipped off-site for detailed examination and a few assemblies are temporarily located within the IEM cell, while they are examined. Irradiated assemblies are handled remotely and those within the reactor core, the IVS, the IDS, and the FSF are cooled with liquid sodium. Assemblies were routinely moved between the reactor core, the IVS and the IDS; but once placed in the FSF, fuel assemblies have not been reinserted into the reactor.

4.4.1 The Reactor Core

The main purpose of the reactor is to provide the capability of safely testing candidate materials and fuel for the fast breeder reactor program in environments approximating those found in actual reactors. The reactor contains the fueled and non-fueled assemblies, and provides radiation shielding, cooling, and instrumentation to allow safe operation. The reactor is also equipped with instrumentation which permits monitoring of assembly temperature and coolant flow, but is not part of the plant safety system. The reactor is shown in Figure 4.3.

The reactor core contains fueled, non-fueled, and control rod assemblies, as well as radial reflectors and shield components, which function together to create a controlled, fast neutron environment. The core, shown schematically in Figure 4.4, consists of nine rows of vertical elements arranged in a hexagonal array. The active zone consists of the 91 positions of the six inner rows. It can be loaded with as many as 82 fueled and non-fueled-test assemblies, with the other nine positions occupied by control rods. The active zone is surrounded by three rows of reflector assemblies.

4.4.2 In-Vessel Storage

Storing some assemblies within the reactor vessel enables the reactor core to be reconfigured during an operating cycle, without opening the reactor vessel. There are three storage modules within the reactor, each serving one-third of the reactor core. Each module has 19 storage positions, but only 18 can be loaded with assemblies at the beginning of an operating cycle; one position must remain empty to receive the core assembly being replaced. This in-vessel storage is located within the core barrel, but outside of the core region. The IVS location is shown in Figure 4.3.

4.4.3 Interim Decay Storage

The Interim Decay Storage provides a controlled environment for temporary storage of irradiated assemblies and reactor core components between

39110033.1

Figure 4.6: FAST FLUX TEST FACILITY OVERVIEW

the periods they are used in the reactor core. The IDS also provides temporary storage of new assemblies and components enroute to the core. The IDS consists of a rotatable storage basket, contained in a stainless steel primary vessel. A carbon steel guard tank surrounds the primary vessel to help ensure adequate sodium cooling even if the primary vessel leaked. The IDS is located within the reactor containment building.

There are 122 storage positions for assemblies and core components, arranged in five concentric circles. The outer four circles can store 112 12-foot (3.7 meter) assemblies; while the inner circle can store ten special 40-foot (12 meter), non-fuel assemblies. In order to accommodate these different length assemblies, the IDS has a relatively small cylindrical lower section connected to a larger upper section. The decay heat of an assembly placed in the IDS must be no greater than 10 kilowatts. The IDS is shown schematically in Figure 4.7.

4.4.4 Fuel Storage Facility

The Fuel Storage Facility provides a controlled environment for longer term storage of irradiated assemblies and reactor core components. Similar to IDS, the FSF consists of a rotatable storage rack contained in a carbon steel primary vessel. The surrounding carbon steel guard tank ensures adequate sodium containment and cooling, in the event of a primary vessel leak. It is located within the 400 Area of Hanford.

There are 466 positions for assemblies and Ident 69 canisters (containers of irradiated assembly fuel or blanket pins) arranged in concentric circles, although only 380 are currently usable due to criticality considerations. The decay heat of an assembly placed in the FSF must be no greater than 1.4 kilowatts. The FSF is shown schematically in Figure 4.8.

4.4.5 Additional Assembly Storage

Additional storage capacity at the FFTF has been requested to support either early shutdown of the sodium-cooled storage facilities or continued reactor operation, depending on whether or not the reactor is started up again.

The proposed Interim Storage Cask (ISC) concept uses casks capable of storing six assemblies for up to 50 years. The design of the ISC is still being developed, so details are not definitively known. However in one likely possibility, they are approximately 15 feet (4.6 meters) tall, 7.5 feet (2.3 meters) in outside diameter and weigh 50 tons (45 metric tons). The internal cavity is expected to be approximately 12.5 feet (3.8 meters) high and 22 inches (56 centimeters) in diameter. A concrete/steel ISC concept is shown in Figure 4.9.

Irradiated fuel assemblies, with sufficient reactor exposure to meet security qualifications for self-protection and with decay heats of 250 watts

Figure 4.7: FAST FLUX TEST FACILITY INTERIM DECAY STORAGE

39110033.4

Figure 4.8: FAST FLUX TEST FACILITY FUEL STORAGE FACILITY

FAST FLUX TEST FACILITY INTERIM STORAGE CASK

(FBI0010A)

Figure 4.9

or less, would be removed from sodium-cooled storage, flushed with nitrogen, steam, and water to remove the sodium (at most a few grams will remain), and placed into an ISC for interim storage. These assemblies would be retrievable from the ISCs either for shipment to the geologic repository or for beneficial use, or for interim storage in the manner approved within the Record of Decision for the Hanford SNF Management EIS, if different from the ISCs.

Approximately 55 casks would be required for the current inventory of irradiated fuel assemblies within the Safeguards Category IV-E. (Another two ISCs would be required for the irradiated fuel assemblies within the Safeguards Category III-D.) The casks would be stored above ground on an existing concrete pad at the FFTF, with naturally convected air providing the necessary cooling. The remaining irradiated and unirradiated fuel assemblies, which are Safeguards Category I-C, must be stored in a vault within a protected area.

4.4.6 Storage Distribution of Fueled Assemblies

The storage locations of the FFTF irradiated fueled assemblies and Ident 69's as of October 1992 are summarized in Table 4.17 below (WHC 1992i). However, since the fuel assemblies are currently being relocated from the reactor, the distribution of fueled components is changing.

Table 4.17: Location of Fast Flux Test Facility Fuel Assemblies as of October 1992

<u>Number of FFTF Fuel Assemblies</u>				
	<u>DFA</u>	<u>TDFA</u>	<u>TFA</u>	<u>Total</u>
Irradiated Assemblies				
Location				
Reactor Core	63	1	15	79
In Vessel Storage	45	0	0	45
Interim Decay Storage	39	6	12	57
Fuel Storage Facility ¹	63	58	25	146
Off Site	0	0	1	1
Examination Cell	0	0	1	1
Total Irradiated Assemblies	210	65	54	329
Unirradiated Assemblies Location				
400 Area	32	0	0	32
Plutonium Finishing Plant	23	0	0	23
Total	55	0	0	55

Notes: 1.

Fuel Storage Facility inventories include one DFA, 14 TDFAs, and 9 TFAs that have been disassembled. The fuel pins from these assemblies have been removed and are stored in Ident 69.

4.4.7 Fuel Pin Cladding Leaks

There have been 13 assemblies which contained a few fuel pins whose cladding leaked while in the high temperature environment of the reactor (WHC 1992g, WHC 1992j). Nine of the assemblies only leaked radioactive gases, while four also leaked small amounts of fission products. The cladding failures were very small cracks, which may have closed at the cooler storage temperatures in the FSF.

Eight of these assemblies, four TDFAs and four TFAs, still contain all the fuel pins, including those that leaked. Of the other five, one TDFA and three TFAs were disassembled and some of the fuel pins shipped off-site for examination. Due to the difficulty in identifying the specific pins which leaked, it is uncertain whether the pin shipment included the pins that leaked in the FFTF Reactor. One DFA has been disassembled. The fuel pin which was suspected of leaking may be sent off-site after being examined in the IEM cell. The fuel pins remaining from all of these disassembled assemblies are stored in Ident 69 containers.

Table 4.18 lists the assemblies that contain fuel pins which leaked and the estimated number which leaked. Table 4.19 lists similar information for the disassembled assemblies.

Table 4.18: Fast Flux Test Facility Assemblies Containing Fuel Pins Which Leaked

<u>Assembly</u>	<u>Type</u>	<u>Estimate of Fuel Pins That Leaked</u>	<u>Serial No.</u>
HF025	TDFA	1	2053
RTCB-4	TDFA	1	2060
MW-4	TDFA	1	16490
AAD-6	TDFA	1	8256
AB-1	TFA	1	2114
ACN-1	TFA	2	2111
PO-2	TFA	3	2066
PO-4	TFA	5	2069

Table 4.19: Disassembled Fast Flux Test Facility Assemblies That Held Fuel Pins Which Leaked

<u>Assembly</u>	<u>Type</u>	<u>Estimated of Fuel Pins That Leaked</u>	<u>Pins Remaining In Ident 69 Containers</u>	<u>Serial No.</u>
16392	DFA	1	216	16392
D9-2	TDFA	1	176	16505
DE-9	TFA	1	177	16527
FO-1	TFA	1	169	2073
PO-1	TFA	2	152	2065

5.0 SINGLE PASS REACTOR FUEL

5.1 BACKGROUND

Eight single-pass water-cooled, graphite moderated reactors (SPR) operated at Hanford to produce plutonium, beginning in 1944 with the 105-B Reactor start-up and ending in 1971 with the 105-KE Reactor shut down (HAPO 1944, HAPO 1961, HAPO 1963a, HAPO 1963b, PNL 1991a). The reactors and their operating periods are listed in Table 5.1.

Table 5.1: Operating Periods of the Hanford Single Pass Reactors

<u>Reactor</u>	<u>Hanford Area</u>	<u>Operating Start up</u>	<u>Period Shut down</u>
105-B	100-B/C	1944	1968
105-C	100-B/C	1952	1969
105-D	100-D/DR	1944	1967
105-DR	100-D/DR	1950	1964
105-F	100-F	1945	1965
105-H	100-H	1949	1965
105-KW	100-KW/KE	1955	1970
105-KE	100-KW/KE	1955	1971

The older reactors (105-B, 105-C, 105-D, 105-DR, 105-F, and 105-H) were similar in design. Their graphite block cores were 36 feet (11 meters) high by 36 feet (11 meters) wide by 28 feet (8.5 meters) long. A total of 2004 horizontal aluminum process tubes held the fuel and contained the cooling water. Cooling water transferred the reaction heat from the uranium fuel elements to the secondary coolant water in steam generators. There were nine horizontal water-cooled, boron containing control rods running perpendicular to the process tubes, except in 105-C and 105-H Reactors which had 15 of these control rods. The control rods entered the reactor from the side and provided operating reactivity control, neutron flux shaping, and emergency shutdown control. A completely independent emergency shutdown control was provided by vertical channels penetrating the core which could be gravity filled with either boron-containing safety rods or boron-containing steel balls.

The basic design of the 105-KE and 105-KW Reactors was similar to the older reactors except as follows. Their graphite cores were 41 feet (12.5 meters) high by 41 feet (12.5 meters) wide by 35 feet (10.7 meters) long. There were 3220 horizontal aluminum process tubes, 2400 of which were later replaced with zirconium tubes, and there were 20 horizontal control rods.

5.2 FUEL DESCRIPTION (HAPO 1955, DUN 1967)

The single pass reactor fuel consisted of a machined uranium core that was sealed within and metallurgically bonded to an aluminum can. There were three basic designs for SPR fuel elements; solid core cylinders, hollow core cylinders, and tubes. All three types are shown in Figure 5.1. The solid cylinder fuel was used from 1944 until the late 1950's. Its design was adequate until increasing reactor power levels and fuel exposure caused significant fuel failures beginning in 1951. The hollow cored fuel was used concurrently with the solid fuel from late-1954 until 1957 in an attempt to reduce fuel failures, but the design was unsuccessful. The tube design, called an internally and externally cooled (I & E) fuel, was used beginning late-1955 and did successfully reduce fuel failures.

Many variations of these basic designs were created in attempts either to improve cooling of the fuel element during irradiation or to simplify the fabrication process. In one successful variation, called the tru-line design, one end of the fuel element had a raised face while the other end had a recessed face. The raised face mated with the recessed face of the adjacent fuel element in the tube to improve cooling through better alignment of the fuel elements within the reactor tubes. Tru-line features are shown on the I & E fuel element in Figure 5.1.

The SPR fuel was made primarily in two ^{235}U concentrations, natural (0.71%) and 0.947%; although other enriched fuel, including 1.25%, 1.7%, and 2.1% ^{235}U ; and depleted fuel, including 0.14% and 0.22% ^{235}U , were also made. The lengths of the fuel elements range from approximately 4.7 inches (12 centimeters) to 9.0 inches (23 centimeters); with specific lengths generally associated with specific ranges of enrichment. The diameters of most of the fuel elements vary from 1.36 inches (3.45 centimeters) to 1.43 inches (3.63 centimeters) for compatibility with slight differences in the process tubes of different reactors. The fuel for the 105-KE and 105-KW Reactors was generally the largest and even included some 2 inch (5.1 centimeters) diameter elements. The fuel for the 105-C Reactor was larger than that used in the other five reactors. These variations in length, diameter, and secondary features, such as tru-line ends, lead to a large number of similar, but distinct, fuel elements. To allow for easy differentiation, each fuel element is stamped with a six character code. The second letter of the code identifies the specific model of fuel element, including the enrichment, dimensions, and group of reactors to receive it.

The physical characteristics of some of the SPR fuel elements are listed in Table 5.2 and the reference drawings H-3-7350 and H-3-7352 (HAPO 1963b). Additional details are available in the reference drawings.

Table 5.2: Physical Characteristics of Single Pass Reactor Fuel¹

		Fuel Element					Uranium Core			Cap	
		Length	Diameter		Cladding Wall Thickness		Length	Diameter		Weight	Length
Code ²	²³⁵ U		Outer	Inner	Outer	Inner		Outer	Inner		
C2N	0.714	8.965	1.466	0.375	0.040	0.045	8.378	1.370	0.481	3.34	0.250
K4N	0.714	8.965	1.460	0.385	0.037	0.050	8.378	1.370	0.498	3.30	0.250
K5N	0.714	8.865	1.520	0.420	0.037	0.050	8.325	1.431	0.533	3.56	0.250
O3N	0.714	8.965	1.444	0.310	0.037	0.050	8.378	1.356	0.423	3.37	0.250
C6N	0.714	8.865	1.986	0.334	0.045	0.050	8.325	1.883	0.452	6.75	0.250
C3E	0.947	6.640	1.460	0.375	0.037	0.050	6.053	1.370	0.488	2.40	0.250
K5E	0.947	6.540	1.509	0.432	0.037	0.050	6.000	1.419	0.545	2.49	0.250
O3E	0.947	6.640	1.443	0.310	0.037	0.050	6.053	1.356	0.423	2.43	0.250
K4W	0.714	6.640	1.459	0.400	0.037	0.050	6.053	1.370	0.513	2.36	0.250
K5(1.25)	1.250	5.540	1.504	0.439	0.037	0.050	5.000	1.420	0.545	2.08	0.250

1. All dimensions are in inches, except that the uranium core weight is in kilograms.
2. The first letter of the code indicates in which reactors the fuel was used. "K" designates the 105-KE and 105-KW Reactors. "C" designates the 105-C Reactor. "O" designates the other five reactors. The number indicates the number of times the design has been modified. The third character indicates the general uranium content and other features. "N" indicates natural uranium content (0.714% ²³⁵U). "E" indicates 0.947% ²³⁵U enrichment. "W" indicates natural enrichment with a 2 inch cooling water mixing attachment. "D" indicates depleted uranium. Enrichments above 0.947% ²³⁵U are indicated by the specific enrichment enclosed within parentheses.

5.3 FUEL FABRICATION (HAPO 1955, HAPO 1958)

In the lead dip fabrication process used since 1954, a machined uranium core was sealed in and metallurgically bonded to an aluminum can. The core was first subjected to a degreasing and pickling operation to clean its surface. Then the core was heated to bonding temperature by immersion in a "duplex" bath of aluminum-silicon alloy (Al-Si) floating on lead. The core was preheated to brazing temperature by the lead layer and then pre-wet with the brazing alloy by brief agitation within the Al-Si layer. The core was then transferred to a heated Al-Si canning bath, where it was inserted into a preheated aluminum can and topped with an aluminum cap. After quenching, the fuel element was welded to create a continuous bond between the uranium core, the Al-Si braze, and the aluminum can. The braze layer was approximately 20 mils thick. Next, the ends of the fuel element were machined to the proper length and contour. Then, the exposed braze area was closed by a tungsten inert-gas weld bead. Finally, the completed fuel element was heated in an autoclave and inspected visually and by other non-destructive methods.

The typical impurities of the materials in the SPR fuel elements are shown below.

Table 5.3: Impurity Limits in Single Pass Reactor Fuel Element Metals¹

<u>Impurity</u>	<u>Uranium Alloy 501</u>	<u>Aluminum Alloy X-8001</u>	<u>Al-Si Braze</u>
Boron	NA ²	10	NA
Cadmium	NA	30	NA
Carbon	150-750	NA	NA
Chromium	65	NA	NA
Cobalt	NA	10	NA
Copper	NA	1500	1000
Iron	150	4500-7000	5000
Lithium	NA	80	NA
Magnesium	25	NA	NA
Manganese	25	NA	NA
Nickel	100	9000-13000	NA
Nitrogen	100	NA	NA
Silicon	75	1700	NA
Others, each	NA	500	500
Others, total	NA	1500	2000

1. Values shown are in parts per million.

2. N. A. indicates that the impurity is not present at any appreciable level.

5.4 FUEL INVENTORY

There are 3.38 MTHM of the SPR fuel at Hanford. The majority is 2.87 MT of highly depleted uranium and 8.7 kilograms of plutonium, contained in 779 fuel elements stored at PUREX (DUN 1972). The SPR fuel at PUREX is a portion of the fuel which was irradiated in the KE and KW reactors between December 1966 and May 1968 to produce high ^{240}Pu content plutonium (WHC 1993b). The pre-irradiation ^{235}U content of the fuel was 0.14% and the fuel was irradiated to an average exposure of 3000 MWD/MT. The post-irradiation ^{235}U content is 0.064% and the plutonium contains 26% ^{240}Pu .

The remaining 0.50 MTHM of the SPR fuel is natural or enriched uranium contained in 185 fuel elements; 0.40 MTHM in 138 elements within the 105-KE Basin and 0.10 MTHM in 47 elements within the 105-KW Basin (WHC 1991e). The fuel is predominately the I & E tru-line design. Most of this fuel is residual material from the 105-KE and the 105-KW Reactors, but also includes fuel from the clean out of the 105-C and the 105-D Storage Basins. The specific types and number of fuel elements are listed below.

Table 5.4: Single Pass Reactor Fuel Elements Stored in the 105-KE and the 105-KW Basins

<u>Fuel Type</u>	<u>Basin</u>	<u>Quantity</u>
C2N	KE	2
K4N	KE	3
K5N	KE	42
O3N	KE	15
C3E	KE	42
K5E	KE	29
O3E	KE	4
K4W	KE	1
Uncertain ¹	KW	47

- Existing records of these fuel elements do not indicate the specific fuel type, however their ^{235}U enrichment does not exceed 1.25% (WHC 1990c).

5.5 RADIONUCLIDE CONTENT

The radionuclide content of the depleted SPR fuel stored at PUREX, estimated using the ORIGEN2 and WIMS-E computer codes, is shown in Table 5.5 (WHC 1993b). Radionuclide concentration and decay heat of the SPR fuel stored in the KE and KW Basins are not available because of its small quantity, varied sources, and uncertain discharge and exposure history. However, since typical SPR fuel irradiation was less than 1000 MWD/MTU, nuclide

concentrations, and decay heat should be similar to that of N Reactor Mark IA fuel irradiated to a 6% ^{240}Pu content, which is shown in Table 2.4 (HAPO 1944). It has been at least 20 years since the fuel was irradiated, because the last SPR was shutdown in 1971.

Table 5.5: Activity of Selected Radionuclides in K Reactor Depleted Fuel

Radionuclide	Activity Ci/MT
^3H	208
^{63}Ni	20
^{85}Kr	217
$^{90}\text{Sr/Y}$	4,033
^{99}Tc	2
$^{113\text{m}}\text{Cd}$	5
^{134}Cs	2
^{137}Cs	10,310
^{151}Sm	120
^{154}Eu	139
^{155}Eu	34
^{238}Pu	43
^{239}Pu	120
^{240}Pu	190
^{241}Pu	5,000
^{231}Am	430
Total	20,923

5.6 FUEL STORAGE

The majority of the SPR fuel is stored at the PUREX storage basin, although smaller amounts are also stored at the 105-KE and the 105-KW Basins. The 105-KE and the 105-KW Basins are described in Section 2.6 of this report. The fuel stored in the PUREX storage basin may be transferred to the 105-KE or the 105-KW Basin prior to implementing the Record of Decision for the Hanford SNF Management EIS.

The PUREX storage basin is located at the east end of the 202A Building in the 200 East Area of Hanford. The storage basin dimensions are 31 feet (9.5 meters) long, 20 feet (6.1 meters) wide, and 17 feet (5.2 meters) deep. It is open to the PUREX canyon and contains 12 feet (3.7 meters) of water to limit the radiation level at the surface. The fuel is stored in four buckets, which are 17 inches (43 centimeters) square by 20.5 inches (52 centimeters) tall with a grid of 1/2-inch (1.27 centimeters) diameter drain holes in the bottom. Figure 5.2 is a schematic of the SPR fuel in the PUREX storage basin. Reference drawings for the SPR fuel storage buckets and yokes listed in Table 5.5.

Table 5.6 Reference Drawings for the Single Pass Reactor Fuel Storage Buckets and Yoke

SPR fuel storage bucket	HD-61514
Yoke for storage bucket	H-2-59728
PUREX storage basin	H-2-52096
	H-2-52110
	H-2-52169
	H-2-52182

Three of the four fuel buckets were lifted out of the water recently so that the condition of the fuel could be observed and videotaped from the crane. The buckets are full of fuel elements. There are several larger elements that appear to be either the 2 inch (5.1 centimeter) diameter fuel elements or overpack containers. Tru-line end features can be seen on a few elements, although end features on most elements cannot be seen. There is also sludge or corrosion product on many of the fuel elements.

6.0 MISCELLANEOUS IRRADIATED FUEL

Small amounts of a large variety of irradiated fuel are stored at several different locations on the Hanford Site, including the 324 Building Radiochemical Engineering Facility, the 325 Building Shielded Analytical Laboratory and 325-A Radiochemical Facility, the 327 Building Postirradiation Testing Laboratory, and the 308 Building Neutron Radiograph Facility, all within the 300 Area; and the Plutonium Finishing Plant and the Low Level Burial Grounds of the 200 West Area. This fuel and its storage locations are described in the following sections.

6.1 AT THE 324 BUILDING OF THE 300 AREA

6.1.1 Background

Irradiated fuel is stored at the 324 Building in the 300 Area of the Hanford Site to support the analyses of the Waste Technology Engineering laboratories in the building. The fuel was irradiated in several different commercial pressurized water reactors (PWR) and boiling water reactors (BWR). This fuel is typical of end-of-life fuel conditions from United States commercial nuclear reactors and is being characterized to support the investigation of different nuclear waste disposal forms for the DOE geologic repository project.

6.1.2 Fuel Inventory

There are slightly less than 2.3 MTHM of irradiated fuel stored in the Radiochemical Engineering Cells of the 324 Building, based on nuclear material safeguards reports. There are 7 full or nearly full assemblies, a few pins from another assembly and numerous pieces of fuel pins. Their source and quantity are listed in Table 6.1 (PNL 1992g).

6.1.3 Fuel Description

The following descriptions are specific to the Calvert Cliffs PWR assemblies and the Cooper Nuclear BWR assemblies, but are generally applicable to the other fuel as well.

6.1.3.1 Pressurized Water Reactor Assemblies (PNL 1984, PNL 1988a, PNL 1988b, PNL 1990) - The PWR assembly used in the Calvert Cliffs Reactor No. 1 is 157 inches (399 centimeters) long, contains 176 Zircaloy-4 clad fuel pins in a 14 by 14 square array, and weighs 1150 pounds (522 kilograms). The assembly is shown in Figure 6.1. The structural cage supporting the fuel pins consists of five guide tubes, eight spacer grids, and upper and lower fittings. There is no separate duct surrounding the fuel pin bundle. All structural components are made of Zircaloy-4, except the lowest spacer grid (high nickel alloy) and the upper and lower end fittings (stainless steel). The grid spacers with integral springs provide a lattice which aligns the fuel pins with each other and provides axial, lateral, and rotational restraint.

Table 6.1: Irradiated Fuel Stored at 324 Building

<u>Source Reactor</u>	<u>Type</u>	<u>Quantity of Fuel</u>			<u>Post-Irradiation ²³⁵U Content</u>
		<u>Fuel Pins</u>	<u>Uranium</u>	<u>Pu</u>	
			Kg	Kg	%
Calvert Cliffs #1 "A"	PWR	168 ¹	370.4	12.9 ²	0.69
Calvert Cliffs #1 "B"	PWR	126 ¹	256.2	---	0.62
Calvert Cliffs #1 "C"	PWR	13 ¹	36.4	---	0.32
Point Beach "A"	PWR	176	384.0	10.5 ²	0.89
Point Beach "B"	PWR	176	383.9	---	0.89
Point Beach "C"	PWR	176	383.4	---	0.88
Point Beach "D"	PWR	pieces	4.0	---	0.89
H. B. Robinson #2	PWR	pieces	30.2	0.3	0.72
Shippingport Naval ³	PWR	pieces	1.0	0.01	0.25
Cooper Nuclear "A"	BWR	46 ¹	181.7	1.5	0.67
Cooper Nuclear "B"	BWR	49	183.4	1.6	0.70
G. E. Vallecitos "A"	BWR	pieces	3.6	0.1 ²	1.0
G. E. Vallecitos "B"	BWR	pieces	3.9	---	1.23
G. E. Vallecitos "C"	BWR	pieces	1.2	---	1.04
G. E. Vallecitos "D"	BWR	pieces	1.6	---	1.88
G. E. Vallecitos "E"	BWR	pieces	0.8	---	1.73
G. E. Vallecitos other	BWR	pieces	28.5	---	0.67
Other	BWR/PWR	pieces	0.1	---	1.26-1.58

- Notes
1. The inventory also includes pieces of fuel pins.
 2. Plutonium totals shown are for the group of assemblies from a reactor rather than the for a specific assembly or fuel piece.
 3. The fuel pin pieces from the Shippingport Naval Reactor are from the first core blanket; not from the second core blanket described in Section 3 of this document.

during reactor operation. Any of the fuel pins can be removed individually from the assembly.

The fuel pins are also shown in Figure 6.1. They are 147 inch (373 centimeters) long and have a 0.44 inch (1.1 centimeter) outside diameter. They contain approximately 300 uranium oxide fuel pellets in a 136.7 inch (347 centimeter) stack. The pre-irradiation U^{235} enrichment varies between assemblies from 2.5% to 3.0%. The individual pellets are 0.45 inches (1.1 centimeters) long and have a 0.3765 inch (0.96 centimeters) diameter. There is an aluminum oxide spacer pellet at each end of the fuel stack. The gas plenum at the top of the pin was pressurized with helium before the fuel pin end caps were welded on.

The impurities of the uranium oxide used in the three Calvert Cliffs assembly pins are listed in Table 3.1 of reference (PNL 1988a) and Tables 4.1 of references (PNL 1988b, PNL 1990), while the impurities of the Zircaloy-4 used are listed in Table 3.2 and Tables 4.2 of the same references.

6.1.3.2 Boiling Water Reactor Assemblies (PNL 1989) - The BWR assembly used in the Cooper Nuclear Power Plant is 176 inches (447 centimeters) long, contains 49 Zircaloy-2 clad fuel pins in a 7 by 7 square array, and weighs 560 pounds (254 kilograms). The assembly is shown in Figure 6.2. The structural cage supporting the fuel pins consists of six spacer grids, eight tie pins (fuel pins with threaded ends), upper and lower tie plates, and a duct surrounding the fuel pin bundle. All structural components are made of Zircaloy-2, except the upper and lower tie plates (stainless steel) and the high nickel alloy springs within the grid spacers and between the upper plate and the locking nuts. The grid spacers provide a lattice which aligns the fuel pins with each other and provides axial, lateral, and rotational restraint during reactor operation.

Only limited data are available for the material properties and design specifications for the fuel and cladding, since the manufacturer considers most of the information proprietary. The fuel pins are also shown in Figure 6.2. There are three different types of fuel pins in the assembly: the eight tie pins, 40 standard pins, and one central segmented pin. The standard pins are similar to the tie pins with the omission of the threaded end extensions. These fuel pins are 0.563 inches (1.43 centimeters) in outside diameter, have an active fuel length of 146 inches (371 centimeters) and contain approximately 292 fuel pellets. The individual pellets have a 0.477 inch (1.21 centimeter) diameter and are approximately 0.5 inches (1.27 centimeters) long. The segmented pin consist of eight individual tubes of fuel pellets separated by Zircaloy-2 connectors. The pre-irradiation ^{235}U enrichment within the pins varies between 1.33% and 2.94%. Four of the standard rods and the segmented rod also contain either 3% or 4% gadolinium oxide, a burnable poison which helps to control initial reactor reactivity. The ^{235}U and gadolinium oxide content of the individual fuel pins are listed in Table 4.2 of reference (PNL 1989). The 14-inch (35.6 centimeter) gas plenum at the top of the pin was pressurized with helium during fabrication.

The impurities of the uranium oxide and the Zircaloy-2 used in the Cooper Nuclear assemblies pins were not available from the manufacturer.

6.1.4 Fuel Exposure and Radionuclide Content

The exposure of the fuel ranges from 28 to 42 megawatt days per kilogram of uranium (MWD/KgU) and the time since discharge from the reactors ranges from 11.5 to 30 years, as of November 1992 (BAPL 1973, PNL 1984, GE 1985, PNL 1988a, PNL 1988b, PNL 1989, DOE 1990, PNL 1990). Their exposure and decay time is listed below.

Table 6.2: Exposure and Decay Time of 324 Building Miscellaneous Fuel

<u>Fuel Source</u>	<u>Exposure MWD/KgU</u>	<u>Reactor Discharge Date</u>	<u>Age Years</u>
Calvert Cliffs No. 1 "A"	29.8	October 1980	13.5
Calvert Cliffs No. 1 "B"	41.9	April 1982	12.0
Calvert Cliffs No. 1 "C"	42.3	October 1980	13.5
Point Beach "A"	32.7	October 1981	12.5
Point Beach "B"	32.7	October 1981	12.5
Point Beach "C"	32.7	October 1981	12.5
Point Beach "D"	unknown	unknown	--
H. B. Robinson Unit 2	28.0	May 1972	22.0
Cooper Nuclear "A"	28.1	May 1982	12.0
Cooper Nuclear "B"	27.5	May 1982	12.0
Shippingport Naval	23.8	February 1964	30.0
G. E. Vallecitos "A"	28.7	April 1981	13.0
G. E. Vallecitos "B"	22.6	August 1980	13.5
G. E. Vallecitos "C"	32.6	September 1982	11.5
G. E. Vallecitos "D"	7.5	June 1976	18.0
G. E. Vallecitos "E"	12.0	March 1977	17.0
Other	unknown	unknown	--

The concentration of radionuclides present in these assemblies are a function of the original fuel composition, its irradiation history and the time since the fuel was discharged from the reactor. Short lived fission products have decayed to insignificant concentrations. The activity and mass of the fission products and actinides for most of these assemblies has been estimated by ORIGEN2 analysis. Table 6.3 lists the estimated activity of radionuclides in the Calvert Cliffs No. 1 "B" assembly, with a bounding PWR exposure of 45 MWD/KgU; and in the Cooper Nuclear "A" assembly, with a bounding BWR exposure of 30 MWD/KgU, for activities above ten curies per assembly. Table 6.4 lists the actinide mass of these same assemblies and exposures, for masses above one gram per assembly. A more complete listing of the activity and mass of the radionuclides present in these assemblies is

Table 6.3: Activity of Selected Radionuclides in Calvert Cliffs 45,000 MWD/MTU Assembly and Cooper Nuclear 30,000 MWD/MTU Assembly¹

Nuclide	Calvert Cliffs			Cooper Nuclear		
	Time Since Reactor Discharge			Time Since Reactor Discharge		
	10 Years	15 Years	20 Years	10 Years	15 Years	20 Years
³ H	4.16E+02	3.14E+02	2.37E+02	2.34E+02	1.77E+02	1.33E+02
⁸⁵ Kr	5.59E+03	4.05E+03	2.93E+03	3.52E+03	2.55E+03	1.84E+03
⁹⁰ Sr	6.80E+04	6.04E+04	5.36E+04	4.77E+04	4.23E+04	3.76E+04
⁹⁰ Y	6.80E+04	6.04E+04	5.36E+04	4.77E+04	4.23E+04	3.76E+04
⁹⁹ Tc	1.67E+01	1.67E+01	1.67E+01	1.21E+01	1.21E+01	1.21E+01
¹⁰⁶ Ru	5.86E+02	1.88E+01	6.05E-01	1.82E+02	5.85E+00	1.88E-01
¹⁰⁶ Rh	5.86E+02	1.88E+01	6.05E-01	1.82E+02	5.85E+00	1.88E-01
^{113m} Cd	5.87E+01	4.63E+01	3.65E+01	2.52E+01	1.99E+01	1.57E+01
¹²⁵ Sb	1.36E+03	3.89E+02	1.11E+02	5.13E+02	1.47E+02	4.20E+01
^{125m} Te	3.32E+02	9.48E+01	2.71E+01	1.25E+02	3.58E+01	1.03E+01
¹³⁴ Cs	7.78E+03	1.45E+03	2.70E+02	2.57E+03	4.79E+02	8.91E+01
¹³⁷ Cs	1.09E+05	9.70E+04	8.64E+04	6.90E+04	6.15E+04	5.48E+04
^{137m} Ba	1.03E+05	9.18E+04	8.18E+04	6.53E+04	5.82E+04	5.18E+04
¹⁴⁴ Ce	1.04E+02	1.21E+00	1.41E-02	3.85E+01	4.48E-01	5.21E-03
¹⁴⁴ Pr	1.04E+02	1.21E+00	1.41E-02	3.85E+01	4.48E-01	5.22E-03
¹⁴⁷ Pm	7.87E+03	2.10E+03	5.60E+02	5.26E+03	1.40E+03	3.75E+02
¹⁵¹ Sm	3.30E+02	3.18E+02	3.06E+02	2.80E+02	2.69E+02	2.59E+02
¹⁵² Eu	4.94E+00	3.83E+00	2.97E+00	1.37E+01	1.06E+01	8.22E+00
¹⁵⁴ Eu	7.74E+03	5.17E+03	3.46E+03	3.67E+03	2.45E+03	1.64E+03
¹⁵⁵ Eu	2.62E+03	1.30E+03	6.47E+02	1.18E+03	5.88E+02	2.92E+02
²³⁸ Pu	5.17E+03	4.97E+03	4.78E+03	2.69E+03	2.59E+03	2.49E+03
²³⁹ Np	4.95E+01	4.94E+01	4.94E+01	1.48E+01	1.48E+01	1.48E+01
²³⁹ Pu	3.18E+02	3.18E+02	3.18E+02	3.02E+02	3.02E+02	3.02E+02
²⁴⁰ Pu	6.37E+02	6.39E+02	6.41E+02	5.00E+02	5.00E+02	5.00E+02
²⁴¹ Pu	1.03E+05	8.06E+04	6.33E+04	6.87E+04	5.40E+04	4.24E+04
²⁴¹ Am	2.32E+03	3.03E+03	3.58E+03	1.83E+03	2.30E+03	2.67E+03
^{242m} Am	1.48E+01	1.44E+01	1.41E+01	2.61E+01	2.55E+01	2.49E+01
²⁴² Am	1.47E+01	1.44E+01	1.40E+01	2.59E+01	2.53E+01	2.48E+01
²⁴³ Am	4.95E+01	4.94E+01	4.94E+01	1.48E+01	1.48E+01	1.48E+01
²⁴² Cm	1.22E+01	1.19E+01	1.16E+01	2.15E+01	2.10E+01	2.05E+01
²⁴³ Cm	4.39E+01	3.89E+01	3.44E+01	1.86E+01	1.65E+01	1.46E+01
²⁴⁴ Cm	6.01E+03	4.96E+03	4.10E+01	1.04E+03	8.56E+02	7.07E+02
Total ²	5.01E+05	4.19E+05	3.61E+05	3.23E+05	2.73E+05	2.36E+05
Decay Heat ²	1.63E+03	1.40E+03	1.12E+03	9.28E+02	8.23E+02	7.50E+02

1. Units are curies per MTU, except decay heat is watts per MTU. Nuclides with activity below 10 curies per MTU are not listed.

2. Totals include effect of all fission products and actinides present in one MTU.

Table 6.4: Mass of Selected Actinides in Calvert Cliffs 45,000 MWD/MTU Assembly and Cooper Nuclear 30,000 MWD/MTU Assembly¹

Nuclide	Calvert Cliffs			Cooper Nuclear		
	Time Since Reactor Discharge			Time Since Reactor Discharge		
	10 Years	15 Years	20 Years	10 Years	15 Years	20 Years
²³⁴ U	1.47E+02	1.58E+02	1.69E+02	1.63E+02	1.69E+02	1.75E+02
²³⁵ U	3.74E+03	3.74E+03	3.74E+03	7.44E+03	7.44E+03	7.44E+03
²³⁶ U	4.09E+03	4.09E+03	4.09E+03	3.62E+03	3.62E+03	3.62E+03
²³⁸ U	9.34E+05	9.34E+05	9.34E+05	9.49E+05	9.49E+05	9.49E+05
²³⁷ Np	6.44E+02	6.50E+02	6.57E+02	3.96E+02	4.01E+02	4.07E+02
²³⁸ Pu	3.02E+02	2.90E+02	2.79E+02	1.57E+02	1.51E+02	1.45E+02
²³⁹ Pu	5.11E+03	5.11E+03	5.11E+03	4.86E+03	4.86E+03	4.86E+03
²⁴⁰ Pu	2.79E+03	2.81E+03	2.81E+03	2.19E+03	2.19E+03	2.20E+03
²⁴¹ Pu	9.94E+02	7.82E+02	6.14E+02	6.66E+02	5.24E+02	4.12E+02
²⁴² Pu	8.97E+02	8.97E+02	8.97E+02	4.02E+02	4.02E+02	4.02E+02
²⁴¹ Am	6.76E+02	8.83E+02	1.04E+03	5.33E+02	6.70E+02	7.77E+02
^{242m} Am	1.52E+00	1.48E+00	1.45E+00	2.68E+00	2.62E+00	2.56E+00
²⁴³ Am	2.48E+02	2.48E+02	2.48E+02	7.44E+01	7.44E+01	7.44E+01
²⁴⁴ Cm	7.42E+01	6.13E+01	5.06E+01	1.28E+01	1.06E+01	8.73E+00
²⁴⁵ Cm	4.92E+00	4.92E+00	4.92E+00	6.57E-01	6.57E-01	6.57E-01
Total ²	9.54E+05	9.54E+05	9.54E+05	9.69E+05	9.69E+05	9.69E+05

1. Units are grams per metric ton of unirradiated uranium. Nuclides with mass below 1 gram per MTU are not listed.
2. Totals include all actinides present from one metric ton of unirradiated uranium.

shown in the Appendices of references (PNL 1989, PNL 1990). The activity and mass of the radionuclides present in the Calvert Cliffs No. 1 "A" fuel, the H. B. Robinson fuel and the Cooper Nuclear "B" fuel are shown in the Appendices of References (PNL 1984, PNL 1988a, PNL 1988b). There is no specific radionuclide information for the Point Beach, Shippingport Naval, and G. E. Vallecitos fuel, however they do not represent a bounding case because of their lower exposures and older discharge dates, except for the G. E. Vallecitos "C" fuel which has only slightly a higher exposure and more recent discharge date than the Cooper Nuclear "A" BWR fuel.

6.1.5 Fuel Storage

The irradiated fuel is stored in the Radiochemical Engineering (RE) Cells of 324 Building (PNL 1991b). These cells are used primarily for engineering development work on highly radioactive materials. The full and partial assemblies are stored in the RE Cell B and the pieces are stored in the RE Cell D.

The walls of the RE Cells are constructed of normal and high density magnetite concrete that are nominally 4 feet thick (1.2 meters). The RE B-Cell has a stainless steel floor and a stainless steel liner on its walls. Its dimensions are 25 feet long (7.6 meters), 22 feet wide (6.7 meters) and 30.5 feet high (9.3 meters). The RE D-Cell has a stainless steel floor and mild steel walls with corrosion-resistant coatings. Its dimensions are 13 feet long (4.0 meters), 21 feet wide (6.4 meters) and 17 feet high (5.2 meters).

The fuel is stored dry in racks within the cells. Air provides the necessary cooling and is filtered by high efficiency particulate air filters before discharged through an exhaust stack. The fuel is handled remotely with cranes and master-slave manipulators.

6.2 AT THE 325 BUILDING OF THE 300 AREA

6.2.1 Background

Irradiated fuel is stored in the 325 Building in the 300 Area of the Hanford Site to support the process demonstration and analytical chemistry requirements of several DOE programs. Fuel is stored in two different locations: the 325-A Radiochemistry Facility and the Shielded Analytical Laboratory (PNL 1992a, PNL 1992b). Both locations contain fragments of the irradiated fuel stored in 324 Building, which are described in Section 6.1, and fragments of a few other commercial fuels. The Shielded Analytical Laboratory also contains fragments from other fuel, including FFTF fuel and N Reactor fuel.

6.2.2 Fuel Inventory

There are 5.6 kilograms uranium and 0.05 kilograms of plutonium in the irradiated fuel stored in the hot cells of 325-A Radiochemistry Facility, and 4.9 kilograms of uranium and 0.08 kilograms of plutonium in the irradiated fuel stored in the hot cells of the Shielded Analytical Laboratory, based on nuclear material safeguards records (PNL 1992c, PNL 1992d). Their source, quantity, and post-irradiation uranium content are listed in Tables 6.5 and 6.6 below.

Table 6.5: Irradiated Fuel Stored at the 325-A Radiochemistry Facility

<u>Source Reactor</u>	<u>Type</u>	<u>Quantity of Uranium Kg</u>	<u>Post-Irradiation ²³⁵U Content %</u>
Shippingport Naval ¹	PWR	2.9	0.25
Yankee	PWR	1.5	unknown
H. B. Robinson Unit 2	PWR	1.0	0.72
Calvert Cliffs No. 1 "A"	PWR	0.1	0.69
Calvert Cliffs No. 1 "B"	PWR	0.1	0.62
Point Beach	PWR	0.03	0.89

Note 1. The fuel pin pieces from the Shippingport Naval Reactor are from the first core blanket; not from the second core blanket described in Section 3 of this document.

Table 6.6: Irradiated Fuel Stored at the Shielded Analytical Laboratory

<u>Fuel Type</u>	<u>Quantity of Uranium (kilograms)</u>	<u>Post-Irradiation ²³⁵U Content (percent) Min. Max.</u>
PWR/BWR	4.1	0.15 - 2.98
FFTF	0.1	0.00 - 0.71
N Reactor	0.4	0.53 - 0.84
Other	0.3	0.22 - 97.01

6.2.3 Fuel Description

The description of the fuel in 324 Building (Section 6.1.3), FFTF fuel and, N Reactor fuel is applicable to most of the fuel in 325 Building. Approximately one-third of the fuel stored in the Shielded Analytical Facility has been chemically altered during analysis and now exists as a saltcake.

Additionally, some of the fuel has been dissolved and remains in liquid solution (PNL 1992a).

6.2.4 Fuel Exposure and Radionuclide Content

The exposure and radionuclide content of most of the fuel in the 325 Building is the same or similar to the fuel in the 324 Building, FFTF fuel and, N Reactor fuel described in other sections of this report.

6.2.5 Fuel Storage

The irradiated fuel is contained in the A and B hot cells of the 325-A Radiochemistry Facility and the hot cells of the Shielded Analytical Laboratory (PNL 1992a, PNL 1992b). These hot cells are used primarily for engineering development work and analytical chemistry on highly radioactive materials.

In the 325-A Radiochemistry Facility, the hot cell walls are constructed of high density magnetite concrete that are nominally 4 feet (1.2 meters) thick (PNL 1980). These hot cells have a stainless steel floor and a stainless steel liner on their walls. The nominal dimensions of the A cell are 6.6 feet (2.0 meters) long, 15 feet (4.6 meters) wide and 15 feet (4.6 meters) high. The nominal dimensions of the B cell are 6.8 feet (2.0 meters) long, 6 feet (1.8 meters) wide, and 15 feet (4.6 meters) high.

Fuel is stored dry on shelves within the 325-A Radiochemistry Facility hot cells. Air provides the necessary cooling and is filtered by high efficiency particulate air filters before discharged through an exhaust stack. The fuel is handled remotely with master-slave manipulators.

In the Shielded Analytical Laboratory, the hot cell construction and fuel storage arrangement are similar to that described above for the 325-A Radiochemistry Facility, although the specific dimensions of the hot cells differ.

6.3 AT THE 327 BUILDING OF THE 300 AREA

6.3.1 Background

Irradiated fuel is examined and stored in the Postirradiation Testing Laboratory within the 327 Building in the 300 Area of the Hanford Site to obtain the destructive examinations required to determine mechanical properties in support of several DOE programs (PNL 1988c).

6.3.2 Fuel Inventory

There are 24.5 kilograms heavy metal of irradiated fuel stored in the Postirradiation Testing Laboratory within the 327 building, based on nuclear

material safeguards records. (PNL 1992e). There are some intact fuel pins from the FFTF and many, many smaller pieces from the FFTF, the Experimental Breeder Reactor II (EBR II) at the Idaho National Engineering Laboratory, the commercial light water reactor fuel described in Section 6.1, and the Space Power 100 program (SP-100), as well as a small amount of N Reactor fuel. Their quantities and range of post-irradiation uranium contents are listed below.

Table 6.7: Irradiated Fuel Stored at 327 Building

<u>Fuel Type</u>	<u>Quantity of Fuel¹</u>		<u>Post-Irradiation</u>	
	<u>Uranium</u>	<u>Plutonium</u>	<u>²³⁵U Content</u>	
	Kg	Kg	Percent	
			Min.	Max.
PWR/BWR	7.4	0.1	0.32 -	1.23
FFTF	5.1	1.1	0.00 -	21.85
EBR II	8.4	1.3	0.14 -	94.00
SP-100	0.7	-	0.20 -	46.61
Other	0.2	-	0.20 -	65.67
<u>Total¹</u>	<u>21.8</u>	<u>2.5</u>		

Notes: The inventory of FFTF fuel contains 0.2 kilograms of irradiated thorium fuel not listed in the table.

6.3.3 Fuel Description

The PWR/BWR fuel and FFTF fuel have been previously described in Section 6.1 and Section 4. The EBR II is a test reactor similar to the FFTF in that it irradiates a variety of test fuel assemblies. Most EBR II fuel is highly enriched uranium oxide or plutonium oxide.

6.3.4 Fuel Exposure and Radionuclide Content

The exposure and radionuclide content of most of the fuel in the 327 Building is the same or similar to the fuel in the 324 Building and the FFTF fuel. The exposure and radionuclide content of the other fuel is not readily available, but could be determined by tracking through a large number of individual project files.

6.3.5 Fuel Storage

The irradiated fuel is stored in several different locations within the Postirradiation Testing Laboratory of 327 Building (PNL 1992f). The intact fuel pins are stored in a rack in a water basin and in various facility hot cells. The partial fuel pins and small fuel pieces are stored in small cans on shelves in the ten hot cells, and in a small dry storage vault.

Air provides the necessary cooling and is filtered by high efficiency particulate air filters before being discharged through an exhaust stack. The fuel is handled remotely with cranes and master-slave manipulators.

6.4 AT THE 308 BUILDING OF THE 300 AREA

6.4.1 Background

The 308 Building in the 300 Area of the Hanford Site contains a small Mark I TRIGA reactor (Test, Research, Isotope, General Atomics) and irradiated fuel (WHC 1992k, WHC 1994f). The TRIGA reactor was operated intermittently from the mid-1970's until May 1989. The reactor, called the Neutron Radiography Facility, was used primarily as the neutron source for radiographic non-destructive testing of FFTF fuel. It was also used for training nuclear operators from N reactor and from the Washington Public Power Supply System. The fuel assemblies and the reactor core are stored at the bottom of a water pool. The fuel assemblies are held in racks within the pool. There are 104 assemblies stored in the water pool. All but three of the assemblies have been irradiated to some extent.

6.4.2 Fuel Description and Inventory

There are 101 irradiated TRIGA reactor fuel assemblies made of uranium-zirconium hydride (8 to 8.5% uranium and 91.5% to 92% zirconium) (WHC 1992k, WHC 1994f). Sixty-six assemblies have aluminum cladding and 35 have stainless steel cladding. Ninety-nine fuel assemblies have an outside diameter of 1.47 inches (3.73 centimeters) and a length of 28.4 inches (72 centimeters). Two fuel assemblies have an outside diameter of 1.38 inches (3.51 centimeters) and a length of 45.8 inches (116 centimeters). The longer fuel assemblies are also fuel-followed control rods, having approximately 14.3 inches (36.3 centimeters) of boronated graphite above the fuel. These fuel-followed control rods were used at another site, where they had been inserted into a reactor at varying depths to control neutron flux in the reactor, however they were only stored in water pool storage racks at the TRIGA reactor at Hanford.

The uranium inventory in the irradiated TRIGA fuel assemblies is approximately 19.1 kilograms. The stainless steel clad fuel assemblies each originally contained about 192 grams of uranium, while the aluminum clad fuel assemblies each originally contained about 187 grams of uranium. The pre-irradiation ^{235}U enrichment of both types of fuel was about 20%.

Three stainless steel clad fuel assemblies, also stored in the TRIGA reactor pool, have zero operating time in the reactor; two are fuel-followed control rods from the Diamond Ordnance Reactor Facility (DORF), a deactivated Army reactor in Maryland, and one is a thermocouple instrumented fuel assembly from Oregon State University.

6.4.3 Fuel Exposure and Radionuclide Content

All aluminum clad fuel assemblies and 23 of the stainless steel clad fuel assemblies had been slightly irradiated before they arrived at the Hanford Site (WHC 1992k, WHC 1994f). Twelve stainless steel clad fuel assemblies were purchased unirradiated from General Atomics. Thirty-three aluminum clad assemblies came from the Armed Forces Research Institute Reactor in Maryland, and the other thirty-three came from the Nevada Test Site, which originated from a TRIGA Reactor in Vietnam that was defueled by American forces during the Vietnam conflict. Twenty-three of the slightly irradiated stainless steel fuel assemblies came from the DORF. Twenty-one of these are standard fuel assemblies and two are fuel-followed control rods.

Fifty-six of the aluminum clad irradiated fuel assemblies and the twelve previously unirradiated stainless steel clad fuel assemblies were irradiated in the TRIGA core at the Hanford Site during the thirteen year operating life. The other fuel assemblies received no additional irradiation since arriving at the Hanford Site.

Detailed fuel exposure or radionuclide inventory data has been generated, by using the ORIGEN2 computer code for the TRIGA fuel (WHC 1994g). The twelve previously unirradiated fuel assemblies received from General Atomics have data generated based on their operating history at the Hanford Site.

The calculated estimated of the fuel radioactivity of the 68 fuel assemblies that resided in the core of the TRIGA reactor is approximately 450 curies. The estimated radioactivity of the 33 fuel assemblies stored in the reactor water pool, but not irradiated at the Hanford Site is approximately 4 curies. The radioactivity doses of the fuel assemblies have been measured at 50-100 Rem per hour at contact for the fuel that was irradiated at the Hanford Site and at 5 Rem per hour at contact for the fuel stored in the water pool storage racks.

6.5 AT THE 200 AREA PLUTONIUM FINISHING PLANT

There are 84 grams of 93.2 percent enriched uranium fuel stored at the 2736-ZB Building at the Plutonium Finishing Plant in the 200 West Area of the Hanford Site (WHC 1992l). The fuel was irradiated at the University of Washington. The fuel is a uranium-aluminum alloy with aluminum cladding. The fuel is contained in a partial assembly of six plates, which are 65.1 centimeters in length, 7.24 centimeters in width and 0.18 centimeters in thickness. The six fuel plates are stored in one 55 gallon drum. The fuel has had very little exposure; its burnup is effectively zero, with surface dose rates of 0.3 mr/hr gamma and 0.8 mr/hr beta plus gamma.

6.6 AT THE 200 AREA LOW LEVEL BURIAL GROUNDS

6.6.1 Background

The Federal Government recognizes the need to provide safe storage of experimental fuel materials generated by government sponsored programs pending final disposal. Since 1974, experimental irradiated fuel materials have been placed in retrievable storage at the Low Level Burial Grounds in the 200 West Area of Hanford, in compliance with the applicable DOE regulations and normal operating practices. The fuel originates from a variety of reactors, including the Fast Flux Test Facility Reactor (FFTF), the 105-KE Reactor, and the 105-KW Reactor at Hanford; the Molten Plutonium Reactor (LAMPRE) at Los Alamos National Laboratory (LANL) in New Mexico, the EBR II at the Idaho National Engineering Laboratory (INEL) at Idaho Falls; the Test, Research, Isotope, General Atomics (TRIGA) at Oregon State University; and experimental light water reactors (RSWMS, 1991).

The fuel has been placed into sealed containers and then into retrievably stored packages. Currently, there are 75 containers stored in burial ground trenches: 39 containers, holding most of the fuel, are lead shielded, carbon steel EBR II casks, 22 containers are concrete-shielded casks holding four General Electric waste liners, 13 containers are lead-lined, concrete-filled 55 gallon drums, and one container is a zircaloy-hull container similar to the EBR II cask.

6.6.2 Fuel Inventory and Fuel Description

There are 0.327 metric tons of irradiated fuel stored in the 200 Area Low Level Burial Grounds (RSWMS, 1991). In most cases, the irradiated fuel is in the form of small fuel fragments and pellets; the only complete assemblies or fuel pins are those from the small TRIGA. The fuel is briefly described below, with the additional details listed in Table 6.8. More information is available in the records of Solid Waste Management stored in the 600 Area of Hanford in Trailer 57.

1. There are 45.2 kilograms of mixed plutonium and uranium from the General Electric Test Critical Facility and Southwest Experimental Fast Oxide Reactor at Vallecitos Nuclear Center in Pleasanton, California. The fuel is contained in 22 concrete shielded casks each holding four General Electric waste liners, weighing an average of 11,500 kilograms. The fuel was received at Hanford during 1974 and is stored covered with soil in Trench 8 in 218-W-3A.
2. There are 99 kilograms of mixed plutonium and uranium fuel from the EBR II at INEL and 7.8 kilograms of plutonium fuel from the LAMPRE at LANL. The stainless steel clad fuel is contained in 24 EBR II casks 3 holding the LAMPRE fuel and 21 holding the EBR II fuel), each weighing

an average of 2,360 kilograms. The fuel was received at Hanford during 1989 - 1990 and is stored uncovered in Trench 1 in 218-W-4C.

3. There are 49.1 kilograms of mixed plutonium and uranium from the EBR II at INEL. The stainless steel clad fuel is contained in four EBR II casks, each weighing an average of 2,500 kilograms. The fuel was received at Hanford during 1981 - 1982 and is stored uncovered in Trench 1 in 218-W-4C.
4. There are 63.9 kilograms of mixed plutonium and uranium from the Montecello Reactor, the Quad Cities Reactor, the Millstone Reactor and the General Electric Test Reactors. The fuel is contained in six EBR-II casks, each weighing an average of 2,500 kilograms. The fuel was received at Hanford during 1984 and is stored uncovered in Trench 1 in 218-W-4C.
5. There are 44.5 kilograms of mixed plutonium and uranium from the EBR II at INEL and the FFTF Reactor at Hanford. The stainless steel clad fuel is contained in 5 EBR II casks, each weighing on average 2,260 kilograms. The fuel was received at the Burial Grounds during 1987 and is stored covered with soil in Trench 6S in 218-W-3A.
6. There are 17.2 kilograms of uranium from the TRIGA at Oregon State University. The uranium hydride fuel in the TRIGA fuel element is mixed with zirconium hydride (8.5% uranium and 91.5% zirconium). Six or seven of the aluminum clad fuel assemblies are contained in each of 13 lead-lined, concrete filled 55 gallon drums. The average weight of the drums is 1,043 kilograms. Overall dimensions of the TRIGA fuel elements are 1.47 inches (3.73 centimeters) in outside diameter and 28.4 inches (72.1 centimeters) in length (WHC 1986). The fuel was received at Hanford during 1987 and is covered with soil in Trench 7 in 218-W-4C.
7. There are 0.12 kilograms of americium-plutonium contained in 12 americium target elements from one of the 105-K Reactors at Hanford (UNI 1979). The fuel is contained in one zircaloy hull container [30 inches (76 centimeters) in diameter by 60 inches (152 centimeters) high] weighing 3,277 kilograms. The fuel was discharged from the reactor in 1970 and is stored uncovered in Trench 1 in 218-W-4C.

6.6.3 Fuel Exposure and Radionuclide Content

There is limited information available on the exposure of the different fuel. The knowledge of the age of the fuel is limited to and by the date which it was received at the burial grounds. There is limited information on the radionuclide activity present at the time the fuel was received at Hanford. This activity is listed in Table 6.8.

6.6.4 Fuel Storage

Irradiated fuel is stored in the 218-W-3A and 218-W-4C Burial Grounds of the 200 West Area of Hanford (RHO 1985c, RSWMS 1991, WHC 1991f). The general layouts of the 218-W-3A and 218-W-4C Burial Grounds are shown in Figures 6.3 and 6.4. The 218-W-3A Burial Grounds contain trenches with V-shaped, gravel bottoms. The 218-W-4C Burial Grounds contain trenches with flat, gravel bottoms, and with asphalt bottoms. Figure 6.5 shows a schematic of a V-shaped and a flat bottom trench.

The 218-W-4C Burial Ground holds 34 EBR II casks and the zircaloy hull container uncovered in Trench No. 1. It also holds the 13 lead-lined, concrete filled 55 gallon drums in trench 7, covered with at least four feet of soil. The 218-W-3A Burial Ground holds the 22 concrete casks with General Electric waste liners in Trench 8 and five EBR II casks in Trench S6. The casks in 218-W-3A are also covered with at least four feet of soil.

The EBR-II cask is a 59.5 inch (151.1 centimeters) high, cylindrical steel vessel with a 17.5 inch (44.5 centimeter) outside diameter and two 30 inch (76.2 centimeter) diameter spacer rings. The internal cavity is 42 inches (106.7 centimeters) high with a 6 inch (15.2 centimeter) diameter. The cask is constructed of carbon steel with 5 inches (12.7 centimeters) of poured lead for shielding. The cask closure consists of a bolted lid and metallic gasket. A simplified schematic is shown in Figure 6.6. The reference drawing is H-2-68598.

There are two similar casks holding the General Electric waste liners. They are both rectangular concrete-shielded containers with 65.5 inch (166.4 centimeter) square tops. They differ in height; one is 48 inches (121.9 centimeters) high, while the other is 75.5 inches (191.8 centimeters) high. Each has four cavities to hold the General Electric waste liners. The shorter cask holds a 7.5 inch (19.1 centimeter) diameter by 15 inch (38.1 centimeter) long waste liner, while the taller cask holds a 5.56 inch (14.1 centimeter) diameter by 42 inch (106.7 centimeter) long waste liner (ARH 1974). Each well of both casks has a separate bolted lid and gasket. The reference drawing for the shorter cask is H-2-37511.

The 55 gallon TRIGA storage drum is a United States Department of Transportation Type 17C carbon steel drum with approximately 5 inches (12.7 centimeters) of lead and concrete shielding (HEDL 1982, WHC 1986). The external dimensions are 22.5 inches (57.1 centimeters) in diameter by 33.5 inches (85.1 centimeters) high, while internal dimensions are 5.04 inches (12.8 centimeters) in diameter and 24.4 inches (62.0 centimeters) high. The cask closure consists of a bolted lid and gasket. The reference drawing is H-4-101690.

Table 6.8: Irradiated Fuel Stored in Hanford Burial Grounds

Description	1	2	3	4	5	6	7
Source/ Building	GE/ -	PNL/ 324	WHC/ 327	GE/ 324	WHC/ 324	WHC/ 327	WHC/ 105K
Burial Facility	218-W- 3A	218-W- 4C	218-W- 4C	218-W- 4C	218-W- 3A	218-W- 4C	218-W- 4C
Trench	8	1	1	1	S6	7	1
Fuel, gm							
²³³ U	-	78	-	-	1	-	-
^D U	18,111	-	34,057	-	11,597	-	-
^N U	1,504	-	854	-	9,809	-	-
^E U	20,874	78,344	10,614	63,258	13,240	17,175	-
% ²³⁵ U	23%	66%	72%	2%	57%	19%	
Pu	4,704	28,446	3,602	593	9,811	13	74
²⁴³ Am	-	-	-	-	-	-	48
Total	45,193	106,868	49,127	63,851	44,458	17,188	122
Fission Products, Ci							
⁵¹ Cr	n.a. ¹	112	n.a.	n.a.	n.a.	n.a.	n.a.
⁵⁴ Mn	n.a.	26	n.a.	n.a.	n.a.	n.a.	n.a.
⁵⁸ Co	n.a.	55	n.a.	n.a.	n.a.	n.a.	n.a.
⁵⁹ Fe	n.a.	571	n.a.	n.a.	n.a.	n.a.	n.a.
⁶⁰ Co	2,232	452	n.a.	n.a.	n.a.	n.a.	n.a.
⁸⁵ Kr	3,730	n.a.	n.a.	n.a.	n.a.	n.a.	n.a.
⁹⁰ Sr/ ⁹⁰ Y	16,541	n.a.	n.a.	n.a.	n.a.	n.a.	n.a.
¹⁰⁶ Ru/ ¹⁰⁶ Rh	5,254	1,891	n.a.	n.a.	n.a.	n.a.	n.a.
¹³⁴ Cs	n.a.	859	n.a.	n.a.	n.a.	n.a.	n.a.
¹³⁷ Cs/ ¹³⁷ Ba	16,793	46,007	n.a.	n.a.	n.a.	n.a.	n.a.
¹⁴⁴ Ce/ ¹⁴⁴ Pr	52,984	1,282	n.a.	n.a.	n.a.	n.a.	n.a.
¹⁴⁷ Pm	40,234	n.a.	n.a.	n.a.	n.a.	n.a.	n.a.
¹⁵² Eu	n.a.	19	n.a.	n.a.	n.a.	n.a.	n.a.
¹⁵⁴ Eu	n.a.	303	n.a.	n.a.	n.a.	n.a.	n.a.
¹⁵⁵ Eu	38	n.a.	n.a.	n.a.	n.a.	n.a.	n.a.
Total	237,077	99,848	4,600	59,970	118,400	360	150
# Containers	22	24	4	6	5	13	1
Weight, kg	253,053	56,717	9,977	14,966	11,313	13,569	3,277
Volume, L	109,526	16,650	2,775	4,162	3,483	2,718	821

1. n/a = data not available

Figure 6.3: 200 WEST AREA 218-W-3A BURIAL GROUNDS

Reference Drawing H-2-34880 Sheet 1 of 3
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Figure 6.4: 200 WEST AREA 218-W-4C BURIAL GROUNDS

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Figure 6.5: TYPICAL BURIAL GROUND TRENCHES

EXPERIMENTAL BREEDER REACTOR II CASK

FIGURE 6.6

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The zircaloy hull container is similar in design to the EBR-II cask. It is a 70 inches (177.8 centimeters) high, cylindrical steel vessel with a 21.25 inch (54.0 centimeter) outside diameter and two 30 inch (76.2 centimeter) diameter spacer rings. The internal cavity is 56 inches (142.2 centimeter) high with a 6 inch (15.2 centimeter) diameter. The cask is constructed of carbon steel with 3.25 inches (8.3 centimeters) of poured lead and 3.5 inches (8.9 centimeters) of concrete for shielding. The cask closure consists of a bolted lid and gasket. The reference drawing is H-2-74587.

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Appendix 1: Shippingport Core II Blanket Fuel Assembly Fabrication Drawings

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REPUBLICAN AS BUILT DRAWINGS
OF THE ASSEMBLY

Appendix 2: Shippingport Core II Blanket Fuel Assembly Bulk Composition Calculations

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Shippingport Core II Blanket Fuel Assembly Bulk Composition Calculations

I. Determine weight of Zircaloy-4 and uranium oxide in a Shippingport Core II fuel assembly

References: Westinghouse Electric Corporation Drawing 921J987, sheets 1 & 2

A. VOLUME OF PLATE = $97.705" \times 3.653" \times 0.14" = 49.968 \text{ in}^3$

$97.705" \times 3.348" \times 0.14" = 45.796 \text{ in}^3$

VOLUME OF PLATES = $52 \text{ PL/assem} \times 49.968 \text{ in}^3/\text{pl} = 2598.34 \text{ in}^3$
IN ASSEMBLY

$= 8 \text{ PL/assem} \times 45.796 \text{ in}^3/\text{pl} = 366.37 \text{ in}^3$

$= 2964.70 \text{ in}^3$

B. VOLUME OF UO_2 fuel

1) RADIUS END WAFERS

$2 \times 9540 \text{ wafers} @ 1.48" \times 0.25" \times 0.10"$

$2 \times 9540 \times 0.0370 \text{ in}^3 = 705.96 \text{ in}^3$

2) RECTANGULAR WAFERS

$2 \times 10,176 \text{ wafers} @ 1.529" \times 0.25" \times 0.10"$

$2 \times 10,176 \times 0.03823 \text{ in}^3 = 777.96 \text{ in}^3$

Total volume of UO_2 $705.96 \text{ in}^3 + 777.96 \text{ in}^3 = 1483.9 \text{ in}^3$

C. Volume of Zr in assembly plates = $2964.7 - 1483.9 \text{ in}^3$

$= 1480.8 \text{ in}^3$

D. Volume of Zr in assembly shell end plates -

$2 \times (97.705" \times 7.376" \times 0.15") = 216.2 \text{ in}^3$

E. Volume of Zr in assembly sides. There are no "side plates." The sides are made when the ends of the fuel plates are welded together.

For each assembly

$$\begin{aligned}
 &4 \times 30 \text{ welds between fuel plates} \\
 &4 \times 1 \text{ weld between fuel plate and end plate} \\
 &4 \times 30 \times 0.058'' \times 97.705'' \times 0.091'' = 61.88 \text{ in.}^3 \\
 &4 \times 1 \times 0.205'' \times 97.705'' \times 0.066'' = \underline{5.29} \text{ in.}^3 \\
 &\text{total} = 67.17 \text{ in.}^3
 \end{aligned}$$

F. Volume of Zr in fuel cluster extensions -

In each cluster extension:

$$\begin{aligned}
 &2 \text{ sides} \times 3.398'' \times 7.376'' \times 0.3125'' = 15.67 \text{ in.}^3 \\
 &2 \text{ sides} \times 3.398'' \times (7.395'' - 2 \times 0.3125'') \times 0.3125'' = 14.38 \text{ in.}^3 \\
 &\text{total one extension} = 30.05 \text{ in.}^3 \\
 &\text{total two extensions} = 60.10 \text{ in.}^3
 \end{aligned}$$

G. Total Zircaloy-4 volume

Volume in assembly fuel plates + end plates + plate welds + cluster extensions

$$\begin{aligned}
 &1480.8 \text{ in.}^3 + 216.2 \text{ in.}^3 + 67.2 \text{ in.}^3 + 60.10 \text{ in.}^3 \\
 &= 1824.3 \text{ in.}^3
 \end{aligned}$$

H. Weight of Zircaloy-4 in fuel assembly

Density of Zircaloy-4 = 0.237 #/in.³ - Teledyne Wah Chang Zr literature

$$\begin{aligned}
 \text{Weight of Zircaloy-4 in fuel assembly} &= 0.237 \text{ \#/in.}^3 \times 1824.3 \text{ in.}^3 \\
 &= 432.36 \text{ \#/assembly} \\
 &= 196.11 \text{ kg/assembly}
 \end{aligned}$$

I. Weight of UO₂ in fuel assembly

Theoretical density of UO₂ is 10.97 g/cm³ at 100%

from Coroneos' notes
from pg. 415 Fast Breeder
Reactors Walter & Reynolds

Theoretical density of fuel is 96%

$$\text{Density of UO}_2 = 10.97 \text{ g/cm}^3 \times 0.96 \times (16.38 \text{ cm}^3/\text{in.}^3) \times (\text{\#/} 453.59\text{g}) = 0.38 \text{ \#/in.}^3$$

$$\begin{aligned}
 \text{Weight of UO}_2/\text{assembly} &= 0.38 \text{ \#/in.}^3 \times 1483.9 \text{ in.}^3 = 563.88 \text{ \#} \\
 &= 563.88 \text{ \#} \times 453.59 \text{ g/\#} \times (\text{kg}/1000\text{g}) = 255.77 \text{ kg UO}_2
 \end{aligned}$$

$$\begin{aligned}
 \text{Weight of U/assembly} &= 255.77 \text{ kg UO}_2 \times 238\text{g U} / 270\text{g UO}_2 \\
 &= 255.46 \text{ kg U (pre-irradiation)}
 \end{aligned}$$

$$\begin{aligned} \text{Zr: U} &= \frac{196.11 \text{ kg Zr/assembly}}{225.46 \text{ kg /assembly}} = 0.8698 \\ \frac{\text{Zr}}{\text{MTU}} &= \frac{196.11 \text{ kg Zr}}{0.22546 \text{ MTU}} = 869.8 \text{ kg Zr/MTU} \end{aligned}$$

II. Determine the amount of carbon coating around the Shippingport Core II fuel wafers

References:

- 1) (BAPL, 1962)
- 2) Westinghouse Electric Corporation Drawing 921J967, sheets 1 & 2

A. The average thickness of the pyrolytic carbon coating was determined statistically as 51 micro inches. (Reference 1, pg. 41)

B. Surface Area of Wafers

Within the blanket fuel assemblies, there are two different fuel wafers -- a rectangular wafer and a radius end wafer. (See Section I.A. for schematic).

1. Rectangular

$$\begin{aligned} 2 \text{ faces of } 1.529'' \times 0.25'' &= 0.7645 \text{ in.}^2/\text{wafer} \\ 2 \text{ faces of } 1.529'' \times 0.10'' &= 0.3058 \text{ in.}^2/\text{wafer} \\ \underline{2 \text{ faces of } 0.25'' \times 0.10''} &= \underline{0.05 \text{ in.}^2/\text{wafer}} \\ \text{Surface Area} &= 1.1203 \text{ in.}^2/\text{wafer} \end{aligned}$$

2. Radius end (approximate as a rectangle of similar dimensions)

$$\begin{aligned} 2 \text{ faces of } 1.48'' \times 0.25'' &= 0.74 \text{ in.}^2/\text{wafer} \\ 2 \text{ faces of } 1.48'' \times 0.10'' &= 0.296 \text{ in.}^2/\text{wafer} \\ \underline{2 \text{ faces of } 0.25'' \times 0.10''} &= \underline{0.05 \text{ in.}^2/\text{wafer}} \\ \text{Surface Area} &= 1.086 \text{ in.}^2/\text{wafer} \end{aligned}$$

C. Surface Area of Wafers in Fuel Plate

There are 2 x 10,176 radius end wafers/subassembly
2 x 9,540 rectangular wafers/subassembly

$$\begin{aligned} 2 \times 10,176 \text{ w/SA} \times 1.086 \text{ in.}^2/\text{w} &= 22,098 \text{ in.}^2/\text{A} \\ \underline{2 \times 9,540 \text{ w/SA} \times 1.12 \text{ in.}^2/\text{w}} &= \underline{21,370 \text{ in.}^2/\text{A}} \\ \text{Subtotal surface area} &= 43,465 \text{ in.}^2/\text{A} \end{aligned}$$

D. Volume of Carbon Coating

$$\begin{aligned} V &= 43,468 \text{ in.}^2/\text{assembly} \times 51 \times 10^{-6} \text{ in.} \\ V &= 2,217 \text{ in.}^3/\text{assembly} \end{aligned}$$

E. Weight of Carbon in Coating

$$\begin{aligned}\text{Density of Carbon Coating (pg 37 of reference 1)} &= 2.07 \text{ g/cm}^3 \\ \text{Carbon} &= 2.07 \text{ g/cm}^3 \times 16.38 \text{ cm}^3/\text{in.}^3 \times \#/453.59 \text{ g} \\ &= 0.0748 \text{ \#/in.}^3 \\ &= 2.07 \text{ g/cm}^3 \times 16.38 \text{ cm}^3/\text{in.}^3 \times (\text{kg}/1000\text{g}) \\ &= 0.0339 \text{ kg/in.}^3\end{aligned}$$

Weight of Carbon in assembly

$$\begin{aligned}&= 0.0748 \text{ \#/in.}^3 \times 2.217 \text{ in.}^3/\text{assem} = 0.116 \text{ \#/assembly} \\ &= 0.0339 \text{ kg/in.}^3 \times 2.217 \text{ in.}^3/\text{assem} = 0.0752 \text{ kg/assembly}\end{aligned}$$

F. Relative Carbon to Uranium

$$\begin{aligned}\text{Carbon in assembly} &= 0.0752 \text{ kg} \\ \text{Uranium in assembly} &= 225.46 \text{ kg} \\ \text{Carbon to Uranium} &= 0.0752 \text{ kg}/225.46 \text{ kg} = 0.00033 = .033 \text{ wt. \%} \\ \text{Per MTU} &= 0.0752 \text{ kg/assembly}/(225.46 \text{ kg} \times \text{MTU}/1000 \text{ kg}) \\ &= 0.333 \text{ kg Carbon/MTU}\end{aligned}$$

III. Determine amount of Stainless steel in Shippingport Core II blanket fuel assemblies
Reference: (BAPL, 1968)

A. Total weight of blanket fuel assembly = 1180#

(pg 108 of reference)

B. Blanket fuel assembly made from extended fuel cluster & top & bottom stainless steel extension bracelets

Weight of extended fuel cluster

wt. of Zircaloy-4	=	196.11 kg	=	432.36 #
wt. of Uranium oxide	=	255.77 kg	=	563.88 #
wt. of Carbon coating	=	0.08 kg	=	0.17 #
		<u>451.96 kg</u>		<u>996.41 #</u>

C. Weight of SS (by difference) = 1180 # - 996.41 #

$$= 183.59 \text{ #}$$

$$= 83.27 \text{ kg/assembly}$$

D. Weight of SS per MTU = $\frac{83.27 \text{ kg/assembly}}{0.22546 \text{ MTU/assembly}}$

$$= 369.35 \text{ kg SS/MTU}$$

Appendix 3: Exposure of Fast Flux Test Facility Standard Driver Fuel Assemblies

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Appendix 3: Exposure of Fast Flux Test Facility Standard Driver Fuel
Assemblies Through Operating Cycle 12B2 (March 1992)
(Page 1 of 6)

<u>Serial #</u>	<u>Type</u>	<u>Average Exposure MWD/KgHM</u>	<u>FFTF Operating Cycle Last, Irradiated¹</u>	<u>Former Test Assemblies</u>
4097	3.1	78.9	7C	
4098	3.1	64.8	8C	
4099	3.1	63.8	12B2	
4100	3.1	72.1	11B2	
4101	3.1	79.0	12B2	TDFA DE4-2
4102	3.1	68.8	12B2	
4103	3.1	74.5	12B2	
4104	3.1	67.9	10A4	TDFA DE4-5
4105	3.1	80.7	12B2	
4106	3.1	60.9	10C3	
4107	3.1	91.5	12B2	
4108	3.1	11.1	12B2	
4109	3.1	66.4	8C	
4110	3.1	86.9	12A2	
4111	3.1	67.0	9C	
4112	3.1	77.9	7C	
4113	3.1	41.9	11C	
4114	3.1	70.1	12B2	
4115	3.1	78.6	10C3	
4117	3.1	73.0	11B2	
4118	3.1	43.4	3	
4119	3.1	78.1	11B2	
4120	3.1	74.7	11B2	
4121	3.1	71.0	9A2	
4122	3.1	74.0	12B2	
4124	3.1	70.9	12A2	
4125	4.1	65.8	12A1	
4126	4.1	10.8	12B2	
4129	3.1	64.4	8C	
4130	3.1	77.6	9C	TDFA DE1-4
4131	3.1	10.8	12B2	
4133	4.1	25.1	12B2	
4134	3.1	71.7	11B2	
4135	4.1	90.3	12B2	
4136	4.1	21.9	12B2	
4140	3.1	88.1	12B2	
4142	4.1	59.2	11C	
4143	3.1	55.8	9A2	TDFA DE3-4
4144	3.1	74.1	9C	
4145	3.1	64.3	10C3	
4146	3.1	72.5	11B1	

Appendix 3 continued: Exposure of Fast Flux Test Facility Standard Driver Fuel
Assemblies Through Operating Cycle 12B2 (March 1992)
(Page 2 of 6)

<u>Serial #</u>	<u>Type</u>	<u>Average Exposure MWD/KgHM</u>	<u>FFTF Operating Cycle Last Irradiated¹</u>	<u>Former Test Assemblies</u>
4147	3.1	67.4	11B1	
4149	4.1	83.4	12B2	
4152	4.1	88.1	12B2	
4153	3.1	76.5	11C	
4155	3.1	74.1	7C	
4157	3.1	61.4	11A3	
4160	3.1	76.2	7C	
4163	3.1	79.3	12B2	
4164	3.1	74.6	10A4	
4168	3.1	84.0	7C	
4172	4.1	30.5	12B2	
4174	3.1	75.4	10C3	
4175	3.1	78.1	7C	
4176	3.1	100.3	12B2	
4177	4.1	32.2	12B2	
4186	3.1	81.0	12B2	
4189	4.1	76.0	7C	
4192	4.1	66.5	10C3	
4194	4.1	86.5	7C	
4196	4.1	81.2	12B2	
4197	4.1	79.0	12A2	
4198	4.1	78.6	7C	
4199	4.1	73.9	7C	
4200	4.1	69.9	7C	
4201	4.1	78.5	12B2	
4202	4.1	65.9	12B1	
4203	4.1	70.8	12A2	
4205	4.1	64.9	10C3	
4206	4.1	75.0	12B2	
4208	4.1	75.5	11B2	
4209	4.1	65.0	12A1	
4210	4.1	80.4	12A2	
4213	4.1	68.4	11B1	
4214	4.1	69.1	11B1	
4216	4.1	74.6	11B1	
4217	4.1	71.6	8C	
4219	4.1	72.9	12B2	
4220	4.1	57.7	10C3	
4221	4.1	81.4	12A2	
4223	4.1	62.0	11A2	
4224	4.1	72.5	12B2	

Appendix 3 continued: Exposure of Fast Flux Test Facility Standard Driver Fuel
Assemblies Through Operating Cycle 12B2 (March 1992)
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<u>Serial #</u>	<u>Type</u>	<u>Average Exposure MWD/KgHM</u>	<u>FFTF Operating Cycle Last, Irradiated¹</u>	<u>Former Test Assemblies</u>
8195	3.1	66.3	9A2	
8196	4.1	68.0	8C	
8198	3.1	83.5	7C	
8201	3.1	71.0	12B2	
8202	3.1	83.6	12A1	
8203	3.1	62.0	8C	
8204	3.1	64.0	12B2	
8205	3.1	63.7	12A2	
8206	3.1	74.7	8A	
8207	3.1	73.6	10B	
8208	3.1	74.1	10B	
8209	3.1	68.3	11B2	
8210	3.1	64.7	12B2	
8211	3.1	66.8	8C	
8212	3.1	72.4	12B2	
8213	3.1	73.5	8C	
8214	3.1	42.4	9C	
8216	3.1	51.4	12B2	
8218	3.1	47.5	9A2	TDFA DE1-5
8219	3.1	78.8	8C	
8220	3.1	18.8	12B2	
8223	4.1	73.0	11B2	
8224	4.1	74.5	11B2	
8227	4.1	36.4	12B2	
8228	3.1	66.9	10C3	
8230	3.1	72.9	8A	
8231	3.1	64.1	11A3	
8234	3.1	78.9	6D	
8236	3.1	76.9	12B2	
8238	3.1	60.6	8C	
8239	3.1	73.4	11B2	
8240	3.1	64.7	10C3	
8241	3.1	80.1	9A2	
8244	4.1	38.3	12B2	
8245	4.1	20.4	12B2	
8246	4.1	36.6	12B2	
8247	4.1	35.9	12B2	
8255	4.1	69.5	7C	
8257	4.1	71.1	11B2	
8258	4.1	65.9	10A4	
8259	4.1	92.3	12B1	

Appendix 3 continued: Exposure of Fast Flux Test Facility Standard Driver Fuel
Assemblies Through Operating Cycle 12B2 (March 1992)
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<u>Serial #</u>	<u>Type</u>	<u>Average Exposure MWD/KgHM</u>	<u>FFTF Operating Cycle Last Irradiated¹</u>	<u>Former Test Assemblies</u>
8260	4.1	73.2	10B	
8261	4.1	76.5	8C	
8262	4.1	66.0	12B2	
8263	4.1	84.2	11B2	
8264	4.1	98.7	12B2	
8265	4.1	69.0	12A1	
8267	4.1	94.7	12A2	
8268	4.1	71.3	11B2	
8269	4.1	70.1	11A3	
8271	4.1	59.3	12B2	
8273	4.1	74.1	11A3	
8274	4.1	75.5	11B2	
8275	4.1	82.8	7C	
8276	4.1	68.7	10C3	
8277	4.1	67.5	11A3	
8281	4.1	68.8	8C	
8282	4.1	62.7	10B	
8283	4.1	80.9	12A2	
8287	4.1	7.0	12B2	
16385	3.2	78.2	11B2	
16387	3.2	67.8	9C	
16388	3.2	75.1	12A2	
16389	4.2	87.8	12A1	
16390	3.2	76.3	11B2	
16391	4.2	71.4	12B2	
16392	4.2	84.7	7B	
16393	3.2	82.7	12A2	
16394	3.2	66.1	7C	
16395	4.2	71.2	10C3	
16396	3.2	78.5	12A1	
16398	3.2	57.4	8C	
16400	3.2	59.7	8C	
16401	4.2	78.2	6D	
16402	3.2	75.6	12B1	
16403	3.2	73.5	11B2	
16404	3.2	58.4	12A2	
16405	3.2	68.8	6D	
16406	4.2	75.1	11C	
16407	3.2	62.7	8C	
16408	3.2	63.8	8C	
16409	3.2	75.9	12B1	

Appendix 3 continued: Exposure of Fast Flux Test Facility Standard Driver Fuel
Assemblies Through Operating Cycle 12B2 (March 1992)
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<u>Serial #</u>	<u>Type</u>	<u>Average Exposure MWD/KgHM</u>	<u>FFTF Operating Cycle Last Irradiated¹</u>	<u>Former Test Assemblies</u>
16410	4.2	70.5	10C3	
16411	3.2	77.2	9A2	
16412	3.2	77.2	7C	
16413	3.2	70.2	12B2	
16414	4.2	74.0	10C3	
16415	4.2	73.0	11C	
16417	3.2	65.3	7C	
16418	4.2	74.5	11B1	
16420	4.2	77.3	12A2	
16421	4.2	76.5	12B2	
16422	3.2	70.3	12A1	
16424	3.2	79.5	12A2	
16425	3.2	71.6	7C	TDFA DE1-1R
16427	4.2	90.4	12A1	
16428	3.2	81.7	12B2	
16429	4.2	42.2	12B2	
16430	4.2	66.3	11A3	
16431	4.2	67.4	9A2	
16432	3.2	82.8	7B	
16441	3.2	60.5	5B	
16442	4.2	78.4	12B2	
16443	3.2	70.8	12B2	
16445	3.2	72.1	7C	
16446	3.2	72.9	12B2	
16448	4.2	22.4	12B2	
16451	3.2	64.9	5B	
16452	4.2	45.3	12B2	
16453	3.2	72.7	12B2	
16455	4.2	81.8	7C	
16456	3.2	73.6	9C	
16457	3.2	74.3	12B2	
16461	3.2	71.7	12A2	
16463	3.2	63.6	5B	
16464	3.2	63.6	5B	
16465	3.2	73.8	7C	
16467	3.2	64.6	5B	TDFA DE3-3
16469	3.2	69.5	12B2	
16470	3.2	73.0	6D	
16471	4.2	70.4	11B1	
16472	4.2	76.0	12B2	
16476	3.2	62.7	5B	

Appendix 3 continued: Exposure of Fast Flux Test Facility Standard Driver Fuel Assemblies Through Operating Cycle 12B2 (March 1992)
(Page 6 of 6)

<u>Serial #</u>	<u>Type</u>	<u>Average Exposure MWD/KgHM</u>	<u>FFTF Operating Cycle Last Irradiated¹</u>	<u>Former Test Assemblies</u>
16478	3.2	80.0	12A2	
16479	3.2	75.0	12B2	
16482	3.2	61.1	8C	
16483	3.2	79.6	12B1	
16484	4.2	89.9	12B1	
16486	4.2	65.4	12A2	
16488	4.2	25.4	12B2	
16493	4.2	74.3	12B2	
16495	4.2	7.6	12B2	
16498	4.2	42.6	12B2	
16499	4.2	17.3	12B2	
16503	4.2	8.0	12B2	

Notes

- The last FFTF operating cycle during which the assembly was irradiated. For cycles 1 through 8, periods of (nominal) 100 days of effective full power were considered a single cycle. Core refueling and reactor maintenance were done between cycles. Letter designations were used to differentiate operating periods of a single cycle interrupted by short shutdowns, e.g. cycle 8 was interrupted twice resulting in cycle designations 8A, 8B, and 8C. For cycles 9 through 12, periods of (nominal) 300 effective full power days (EFPD) were considered a single cycle. Annual maintenance was done between cycles. Letter designations were used to differentiate the three nominal 100 EFPD periods within a cycle. Core refueling and reactor maintenance were done between these "part cycles." Additional number designations were used to differentiate operating periods of a single 100 EFPD part cycle interrupted by short shutdowns, e.g. cycle 10A was interrupted three times resulting in cycle designations 10A1, 10A2, 10A3, and 10A4. The dates on which the cycles ended are:

Cycle	Date	Cycle	Date	Cycle	Date
3	10/83	9A2	2/87	11B1	4/90
5B	11/84	9C	10/87	11B2	10/90
6D	6/85	10A4	5/88	11C	3/91
7B	12/85	10B	10/88	12A1	9/91
7C	1/86	10C3	3/89	12B2	3/92
8A	4/86	11A2	6/89		
8C	7/86	11A3	9/89		