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Characteristics of Spent Fuel from Plutonium Disposition Reactors
Vol. 2: A General Electric Boiling-Water-Reactor Design

J. C. Ryman
O. W. Hermann

MANAGED AND OPERATED BY
LOCKHEED MARTIN ENERGY RESEARCH CORPORATION
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

ORNL-27 (3-96)

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ORNL/TM-13170/V2

Computational Physics and Engineering Division

CHARACTERISTICS OF SPENT FUEL FROM PLUTONIUM DISPOSITION REACTORS
VOL. 2: A GENERAL ELECTRIC BOILING-WATER-REACTOR DESIGN

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Date Published: April 1998

Prepared by the
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831-6370
Managed by
LOCKHEED MARTIN ENERGY RESEARCH CORP.
for the
U.S. DEPARTMENT OF ENERGY
under contract DE-AC05-96OR22464

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ACKNOWLEDGMENTS

The authors acknowledge the help of Brian Murphy, who performed the original SAS1 calculations for the MOX assembly and also reviewed the document. We appreciate the efforts of Trent Primm, who gave many valuable suggestions and reviewed the manuscript. The quality of the final report was greatly improved by Cathy Shappert, the technical editor. This work was supported by the Fissile Materials Disposition Program (FMDP).

ABSTRACT

Spent fuel characteristics of boiling-water-reactor (BWR) fuel assemblies have been computed for use in the Fissile Materials Disposition Program. Burnup and decay calculations and some simple dose rate calculations have been performed to provide data needed for decay heat, radiation shielding, criticality safety, and biological hazard analyses. The calculations were performed for a mixed-oxide (MOX) fuel cycle based on an existing BWR design and for a low-enriched-uranium fuel cycle typical of present-day BWR designs. Tables and plots of data for the two assembly designs are presented so that future analyses can be performed to provide the basis for selecting a method for plutonium disposition.

1. INTRODUCTION

Shortly after the instigation of the Fissile Materials Disposition Program (FMDP), program managers realized that to coordinate all parts of the program, documentation of neutronics parameters for all parts of the fuel cycle was needed. Data for radiation shielding, decay heat generation, criticality safety and biological hazard analyses were requested by fabrication facility designers, reactor facility operators, transportation analysts, and repository designers. In this report, these data are provided for two boiling-water-reactor (BWR) fuel cycles. An existing boiling-water reactor, BWR-5, is described in ref. 1 and is the basis for the BWR analyses. A uranium and plutonium mixed-oxide (MOX) fuel cycle case is described in Sect. 2. To provide a basis for comparison, an analysis of a low-enriched-uranium (LEU) fuel cycle case was conducted and is described in Sect. 3 was also run. Results of both cases are discussed in Sect. 4.

The SCALE[2] and AMPX[3] computer code systems were used to perform the calculations described here. All of the data presented in this report are from the program output. Even though the authors have endeavored to tabulate all data needed for fuel cycle analyses, all requests cannot be anticipated. The program output provides users access to a larger amount of data than can be tabulated easily. Furthermore, it is possible to calculate parameters for more than 1000 nuclides, many of which are not of interest to fuel cycle investigators. Therefore, the authors have chosen to exclude certain nuclides from program output. Since subsequent studies may reveal a need for data not included in the output, input datasets are available on the accompanying diskette. Users familiar with the SCALE and AMPX systems are encouraged to modify these datasets and rerun any cases of interest to obtain additional needed information.

Certain parameters calculated in these studies have been reported in ref. 1. Where available, those data are included here. This comparison is an indicator of the precision (and hopefully the accuracy) of these calculations.

2. THE URANIUM AND PLUTONIUM MIXED-OXIDE CASE

2.1 FRESH FUEL CHARACTERISTICS AND OPERATING PARAMETERS

Data and assumptions needed for the SCALE[1] calculations are taken from a combination of sources. Reference 1 is the General Electric (GE) report of an evaluation of plutonium disposition using existing GE boiling-water reactors (BWRs). A source for fuel rod dimensions is a report[4] characterizing potential repository wastes. Other sources[5,6] are useful in making reasonable assumptions of data, such as moderator densities and temperatures. Tables 2.1–2.4 include the fresh fuel composition and BWR-5 design and operating parameters applied in the analyses. Table 2.5 shows a summary of the plutonium disposition and fuel rod usage produced by utilizing an existing GE BWR-5 nuclear plant. Graphical representations of a fuel assembly are shown in Figs. 2.1 and 2.2.

Tables 2.1 and 2.2 show design data pertaining to the characteristics of the fuel assembly, the fuel rods, the guide tube, the shroud (assembly casing or tube), the channel water, and the ^{10}B equivalent content for the cruciform control rod. References, by item, are listed. In general, data pertaining to the MOX fuel composition and other differences from the existing reactors were taken from the plutonium disposition document.[1] Design data that were not contained in this report were taken from the vendor-supplied data[4] given as part of the repository waste characterization. Data in Tables 2.1 and 2.2 not available in refs. 1 and 4, such as material temperatures and effective water densities, were taken from other BWR analyses.[5,6] Most of the predicted results are not highly sensitive to the assumed data. The approximated guide tube data are insignificant.

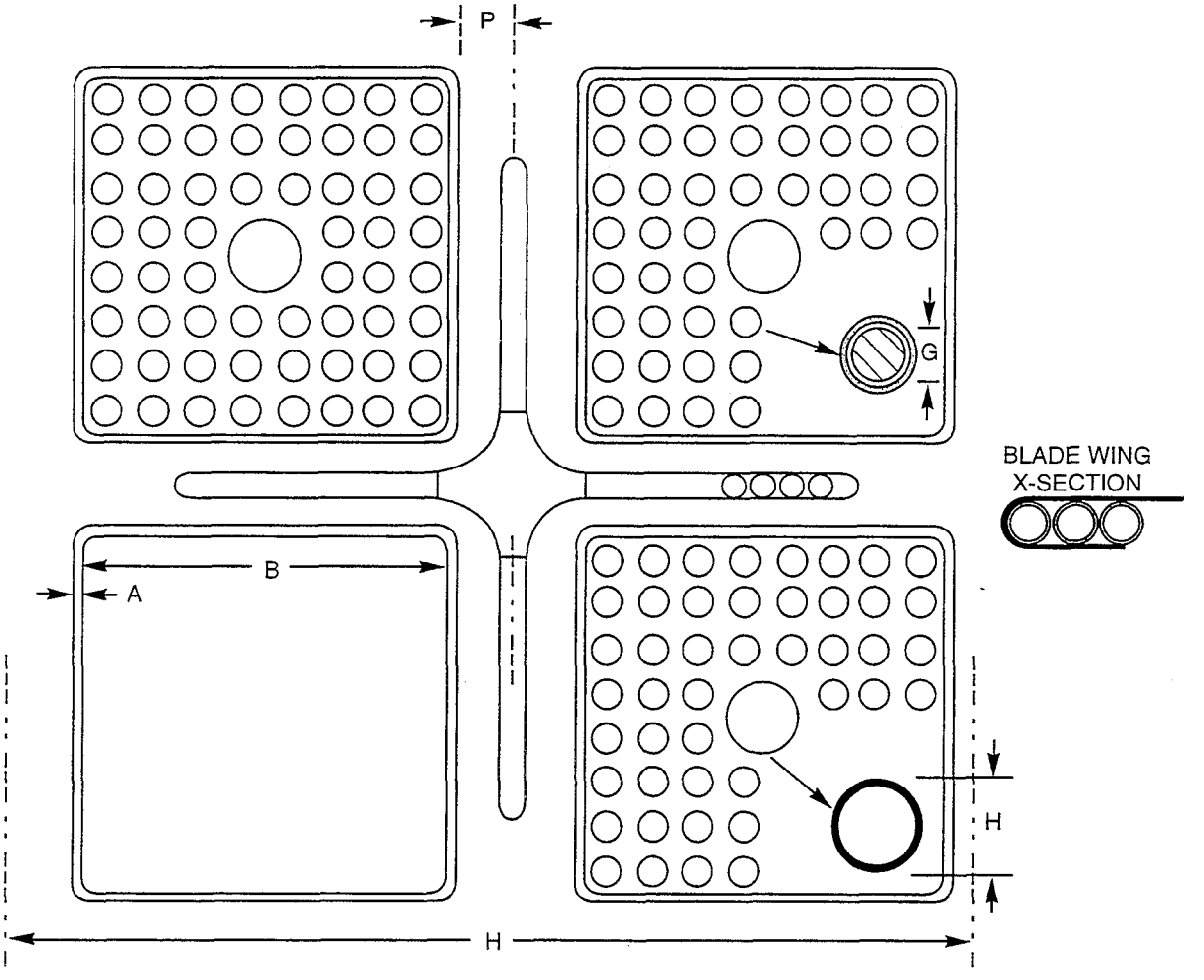
Table 2.3 fuel isotopic data were specified[7] for this project. Depleted uranium was specified for these analyses. In Table 2.4, the operating power history was derived from the cycle times, capacity factors and total burnup given[1] for the BWR. The first cycle was split into two intervals, 80 and 260 d, in order to apply more precisely the rapidly changing densities of the highly significant gadolinium isotopes. The common assumption of constant power in place of the more usual decreasing power tends to increase the nuclide activities at the earlier (<5-year) cooling times.

The major core parameters for the 3484 MWt reactor are listed in Table 2.5. The annual usages for plutonium and fuel rods were computed from the prior data in the table. Table 2.6 provides vendor-supplied data for fuel cycle analyses. Note that the data in Table 2.6 are reproduced unchanged from ref. 1 and that the composition data from this table are not used.

Table 2.1. GE BWR-5 full MOX fuel assembly design data

Parameter	Data
Assembly General Data	
Reactor type	Boiling water, BWR-5[a]
Designer	General Electric Co.[a]
Lattice	8 × 8[a]
Assembly pitch, cm (in.)	15.24 (6.0)[a]
Uranium weight, kg U	173.27[a]
Plutonium weight, kg Pu	5.334[a]
Gadolinium, avg wt % HM	0.5333[a]
Assembly water temperature, K	558[b]
Avg water density, g/cm^3	0.4323[b]
Number of fuel rods	60[a]
Number of guide tubes	1[a]
Fuel Rod Data	
Fuel pellet type	Mixed oxide (MOX)[a]
Fuel materials	UO_2 PuO_2, Gd_2O_3[a]
UO_2 stacked density, g/cm^3	10.095[c]
PuO_2 stacked density, g/cm^3	0.3106[c]
Gd avg density, g/cm^3	0.04892[d]
Rod pitch, cm (in.)	1.6256 (0.64)[e]
Rod OD, cm (in.)	1.2268 (0.483)[e]
Rod ID, cm (in.)	1.0643 (0.419)[e]
Pellet diameter, cm (in.)	1.0414 (0.410)[e]
Active fuel length, cm (in.)	381.0 (150.0)[e]
Clad material	Zircaloy-2[e]
Fuel temperature, K	840[b]
Clad temperature, K	620[b]
Guide Tube Data	
Inner radius, cm (in.)	1.3614 (0.536)[f]
Outer radius, cm (in.)	1.4630 (0.576)[f]
Tube material	Zircaloy-2[f]
Number of fuel rod positions	4[a]
Shroud and Channel Data	
Shroud (tube) thickness, cm (in.)	0.254 (0.10)[a]
Shroud outside flat-to-flat, cm (in.)	13.754 (5.415)[a]
Shroud material	Zircaloy-2[f]
Shroud temperature, K	558[b]
Channel water density, g/cm^3	0.669[b]
Channel water temperature, K	552[b]
Channel avg ^{10}B content, atoms/b-cm	2.6×10^{-6} [g]

[a]*Source*: ref. 1.
[b]*Source*: ref. 5.
[c]Computed from other data in this table.
[d]Computed from gadolinium concentration of each fuel pin given in ref. 1.
[e]*Source*: ref. 4.
[f]Assumed; the difference from correct data is probably insignificant to analysis.
[g]Reduced the 0.743 g-cm^3 bottom node density[6] by 10% to account for control rod displacement and small heat transfer in other nodes.



DIMENSION (in.)	
A	0.100
B	5.215
G	0.410
H	0.576
P	0.293
R	12.0

Fig. 2.1. Generic BWR fuel bundle and control blade configuration.

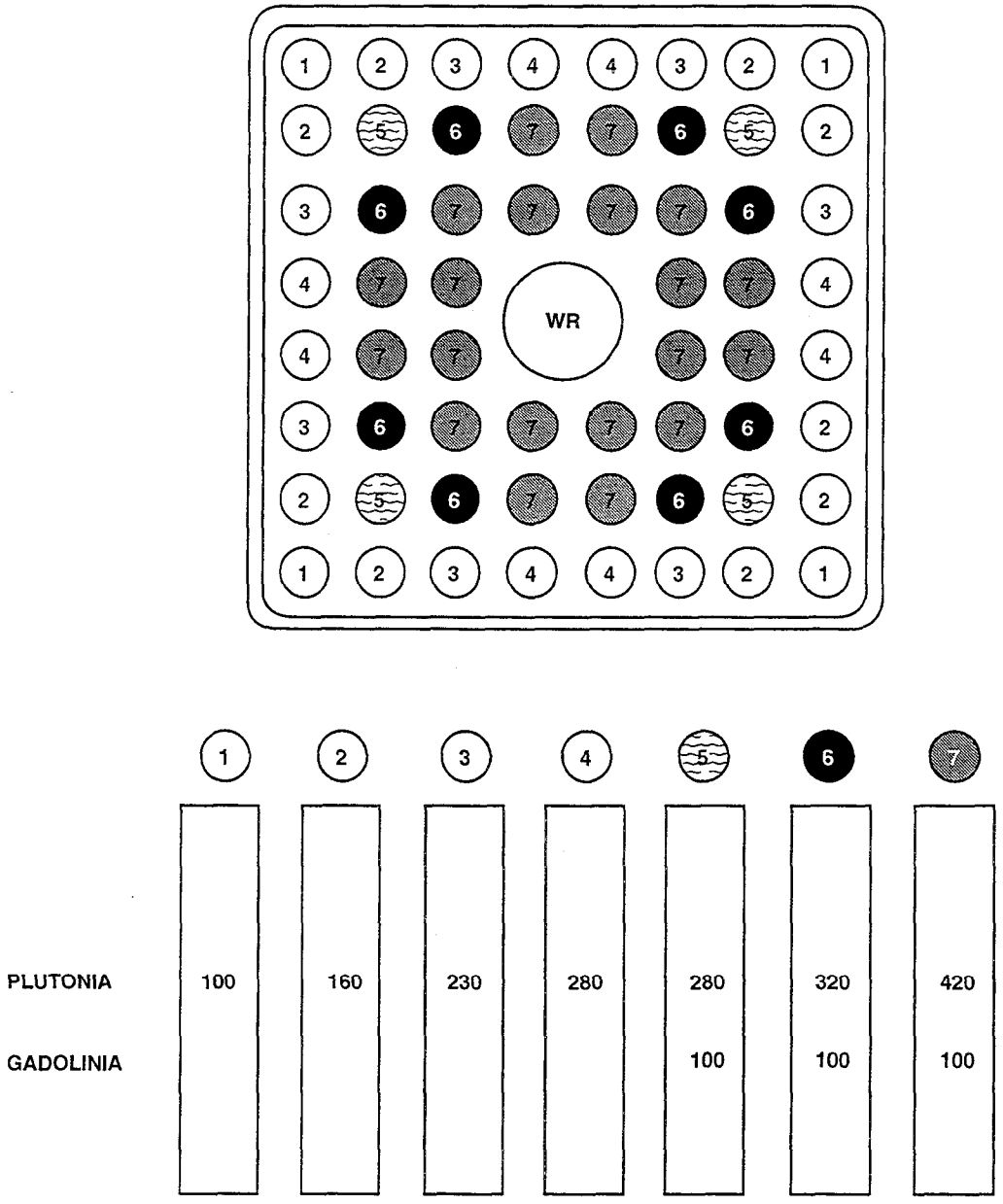


Fig. 2.2. Full MOX bundle design for generic BWR core.

Table 2.2. Axial water densities assumed[a] for GE BWR-5

Top of node, in.	Average density in node, g/cm^3	Average water temperature, K
30.83	0.743	552
43.17	0.600	558
55.50	0.494	558
67.83	0.417	558
80.17	0.360	558
98.67	0.309	558
123.33	0.264	558
148.00	0.234	558

[a]*Source*: ref. 6; these values were obtained as typical data.

Table 2.3. Full MOX GE BWR-5 fuel isotopic composition

Element	Isotope	Wt %
Uranium[a]	^{235}U	0.20
	^{238}U	99.80
Plutonium[a]	^{238}Pu	0.03
	^{239}Pu	93.75
	^{240}Pu	5.70
	^{241}Pu	0.50
	^{242}Pu	0.02
Gadolinium[b]	^{154}Gd	2.18
	^{155}Gd	14.80
	^{156}Gd	20.47
	^{157}Gd	15.67
	^{158}Gd	24.84
	^{160}Gd	21.86

[a]*Source*: ref. 7.
[b]*Source*: ref. 8.

Table 2.4. GE BWR-5 full MOX operating history

Cycle number	Capacity factor, %	Operating time, d		Power, Mw	Burnup,[c] GWd/MTHM	Channel ^{10}B,[d] atoms/b-cm
		Uptime[a]	Downtime[b]			
1A		80	0	4.5602[e]	2.043	0
1B	75	260	113		6.638	3×10^{-6}
2	75	340	113		8.681	3×10^{-6}
3	75	340	113		8.681	3×10^{-6}
4	75	340	113		8.681	3×10^{-6}
5	100	113	0		2.885	0
TOTALS		1473			37.609[a]	
Core (×764)				3484		

[a]*Source*: ref. 1.
[b]Computed from operating uptime and capacity factor.
[c]Cycle burnup increments computed from uptimes and power.
[d]Adjusted from SAS2H cases to give time-weighted $k_{eff} \doteq 1.0$ for an estimated escape probability from core.
[e]Computed from totals of burnup, uptime, and heavy metal (HM) weight of assembly (Table 2.1).

Table 2.5. Core parameters

Parameter	Data
Cycle operating period, EFPD	340[a]
Reload frequency or total cycle time, d	453.3[a]
Discharge burnup, GWd/MTHM	37,609[a]
Number of assemblies/core	764[a]
Reload batch size, assemblies	176[a]
Plant capacity factor, %	75[a]
Pu loading, kg/assembly	5.334[a]
Per usage, MT/year	0.756[b]
MOX fuel rod usage, rods/y	8509[c]

[a]*Source*: ref. 1.
[b]Computed from other data in table; compares with 0.75 in ref. 1.
[c]Computed from other data in table; precisely same value as ref. 1.

Table 2.6. Vendor-supplied data for BWR-5 full mixed-oxide fuel cycle

Full MOX	**Reactor plant identification**
	(DATA REPORTED FOR GENERIC 3 REACTOR CASE)
	I. Plant retrofit and operation schedule (relative to 10/1/94 project start)
Apr-01	1) time MOX burning operations start (mo) - FULL MOX BUNDLE LOADING
3484	2) thermal power (MWth) - SEE APPENDIX C FOR SPECIFIC PLANTS
1155	3) net electrical output (MWth) - SEE APPENDIX C FOR SPECIFIC PLANTS
75	4) capacity factor (average for 1991, 1992, and 1993) - SEE DISCUSSION IN SECTION 4.4
	5) remaining plant life (y) - SEE TABLE 1.2-1
	II. Fresh MOX fuel assembly characteristics
	1) reactor Pu inventory - first core (kg Pu) - TRANSITIONS FROM URANIA CORE WITH EACH RELOAD
	2) reactor U inventory - first core (kg U) - TRANSITIONS FROM URANIA CORE WITH EACH RELOAD
5.334	3) Pu content (kg Pu/fuel assembly)
0	4) Pu-238 fraction (kg Pu-238/kg Pu)
0.94	5) Pu-239 fraction (kg Pu-239/kg Pu)
0.057	6) Pu-240 fraction (kg Pu-240/kg Pu)
0.003	7) Pu-241 fraction (kg Pu-241/kg Pu)
0	8) Pu-242 fraction (kg Pu-242/kg Pu)
303	9) fuel assembly mass (kg)
4.55	10) fuel assembly length (m)
0.15	11) fuel assembly width (m)
173.27	12) U mass in MOX (kg U/fuel assembly)
0.0071	13) U-235 enrichment in MOX (kg U-235/kg U) - NATURAL URANIUM
-	14) U mass in UO2 fuel (kg U/fuel assembly) - ALL MOX RODS
-	15) U-235 enrichment in UO2 fuel (kg U-235/kg U) - ALL MOX RODS
	III. Equilibrium cycle MOX fuel recharge
938.78	1) Pu recharge (kg Pu) - PER REACTOR RELOAD
30495.52	2) U recharge (kg U) - PER REACTOR RELOAD
5.334	3) Pu content (kg Pu/fuel assembly)
0	4) Pu-238 fraction (kg Pu-238/kg Pu)
0.94	5) Pu-239 fraction (kg Pu-239/kg Pu)
0.057	6) Pu-240 fraction (kg Pu-240/kg Pu)
0.003	7) Pu-241 fraction (kg Pu-241/kg Pu)
0	8) Pu-242 fraction (kg Pu-242/kg Pu)
173.27	9) U mass in MOX (kg U/fuel assembly)
0.0071	10) U-235 enrichment in MOX (kg U-235/kg U)
-	11) U mass in UO2 fuel (kg U/fuel assembly)
-	12) U-235 enrichment in UO2 fuel (kg U-235/kg U)
176	13) number of MOX fuel assemblies
14.9	14) cycle length (mo) - IRRADIATION TIME AT FULL POWER/CAPACITY FACTOR
0.23	15) average fraction of core removed per discharge

Table 2.6 (continued)

	IV. MOX fuel assembly characteristics at discharge
3.388	1) Pu mass discharged per cycle in MOX fuel (kg Pu) - PER BUNDLE IN EACH OF 176 BUNDLES
-	2) Pu mass discharged per cycle in UO2 fuel (kg Pu)
0.0099	3) Pu-238 fraction in MOX fuel (kg Pu-238/kg Pu)
0.4209	4) Pu-239 fraction in MOX fuel (kg Pu-239/kg Pu)
0.3528	5) Pu-240 fraction in MOX fuel (kg Pu-240/kg Pu)
0.1508	6) Pu-241 fraction in MOX fuel (kg Pu-241/kg Pu)
0.0656	7) Pu-242 fraction in MOX fuel (kg Pu-242/kg Pu)
-	8) Pu-238 fraction in UO2 fuel (kg Pu-238/kg Pu)
-	9) Pu-239 fraction in UO2 fuel (kg Pu-239/kg Pu)
-	10) Pu-240 fraction in UO2 fuel (kg Pu-240/kg Pu)
-	11) Pu-241 fraction in UO2 fuel (kg Pu-241/kg Pu)
-	12) Pu-242 fraction in UO2 fuel (kg Pu-242/kg Pu)
261500	13) heat generation in one fuel assembly (W)
1.89E+07	14) gamma radiation field (rem/h at surface, midplane of fuel assembly)
1473	15) effective full power days (d)
37.609	16) burnup (MWd/kgHM)
0.01968	17) fraction of Pu in heavy metal (kg Pu/kgHM)
	V. MOX fuel characteristics 1 year after discharge
3.382	1) Pu mass in MOX fuel (kg Pu) - PER BUNDLE
-	2) Pu mass in UO2 fuel (kg Pu)
0.0123	3) Pu-238 fraction in MOX fuel (kg Pu-238/kg Pu)
0.4245	4) Pu-239 fraction in MOX fuel (kg Pu-239/kg Pu)
0.3535	5) Pu-240 fraction in MOX fuel (kg Pu-240/kg Pu)
0.1439	6) Pu-241 fraction in MOX fuel (kg Pu-241/kg Pu)
0.0658	7) Pu-242 fraction in MOX fuel (kg Pu-242/kg Pu)
-	8) Pu-238 fraction in UO2 fuel (kg Pu-238/kg Pu)
-	9) Pu-239 fraction in UO2 fuel (kg Pu-239/kg Pu)
-	10) Pu-240 fraction in UO2 fuel (kg Pu-240/kg Pu)
-	11) Pu-241 fraction in UO2 fuel (kg Pu-241/kg Pu)
-	12) Pu-242 fraction in UO2 fuel (kg Pu-242/kg Pu)
1898	13) heat generation in one fuel assembly (W)
1.30E+05	14) gamma radiation field (rem/h at surface, midplane of fuel assembly)
1473	15) effective full power days (d)
37.609	16) burnup (MWd/kgHM)
0.01964	17) fraction of Pu in heavy metal (kg Pu/kgHM)

Table 2.6 (continued)

	VI. MOX fuel characteristics 10 years after discharge
3.213	1) Pu mass in MOX fuel (kg Pu) - PER BUNDLE
-	2) Pu mass in UO2 fuel (kg Pu)
0.0128	3) Pu-238 fraction in MOX fuel (kg Pu-238/kg Pu)
0.4468	4) Pu-239 fraction in MOX fuel (kg Pu-239/kg Pu)
0.3729	5) Pu-240 fraction in MOX fuel (kg Pu-240/kg Pu)
0.0983	6) Pu-241 fraction in MOX fuel (kg Pu-241/kg Pu)
0.0692	7) Pu-242 fraction in MOX fuel (kg Pu-242/kg Pu)
-	8) Pu-238 fraction in UO2 fuel (kg Pu-238/kg Pu)
-	9) Pu-239 fraction in UO2 fuel (kg Pu-239/kg Pu)
-	10) Pu-240 fraction in UO2 fuel (kg Pu-240/kg Pu)
-	11) Pu-241 fraction in UO2 fuel (kg Pu-241/kg Pu)
-	12) Pu-242 fraction in UO2 fuel (kg Pu-242/kg Pu)
241	13) heat generation in one fuel assembly (W)
1.78E+04	14) gamma radiation field (rem/h at surface, midplane of fuel assembly)
1473	15) effective full power days (d)
37.609	16) burnup (MWd/kgHM)
0.01866	17) fraction of Pu in heavy metal (kg Pu/kgHM)
	VII. MOX fuel characteristics 100 years after discharge
2.876	1) Pu mass in MOX fuel (kg Pu) - PER BUNDLE
-	2) Pu mass in UO2 fuel (kg Pu)
0.0074	3) Pu-238 fraction in MOX fuel (kg Pu-238/kg Pu)
0.4982	4) Pu-239 fraction in MOX fuel (kg Pu-239/kg Pu)
0.4156	5) Pu-240 fraction in MOX fuel (kg Pu-240/kg Pu)
0.0014	6) Pu-241 fraction in MOX fuel (kg Pu-241/kg Pu)
0.0774	7) Pu-242 fraction in MOX fuel (kg Pu-242/kg Pu)
-	8) Pu-238 fraction in UO2 fuel (kg Pu-238/kg Pu)
-	9) Pu-239 fraction in UO2 fuel (kg Pu-239/kg Pu)
-	10) Pu-240 fraction in UO2 fuel (kg Pu-240/kg Pu)
-	11) Pu-241 fraction in UO2 fuel (kg Pu-241/kg Pu)
-	12) Pu-242 fraction in UO2 fuel (kg Pu-242/kg Pu)
94	13) heat generation in one fuel assembly (W)
1.40E+03	14) gamma radiation field (rem/h at surface, midplane of fuel assembly)
1473	15) effective full power days (d)
37.609	16) burnup (MWd/kgHM)
0.0167	17) fraction of Pu in heavy metal (kg Pu/kgHM)

Table 2.6 (continued)

	VIII. Pu inventory and handling at fuel fabrication facility (for identified reactor)
	(ref. Management and Disposition of Excess Weapons Plutonium,
	National Academy of Sciences, 1994, Table 3.1
C	1) DOE attractiveness level of PuO2 feed material
D	2) DOE attractiveness level of MOX fuel pellets
360	3) average inventory of PuO2 feed material (kg Pu)
2124	4) throughput of PuO2 feed material (kg Pu/y)
0.17	5) PuO2 feed material residence time or duration (y)
1	6) number of manual movements of material (e.g. between gloveboxes)
10.7	7) average inventory of MOX fuel pellets (kg Pu)
2124	8) throughput MOX fuel pellets (kg Pu/y)
0.005	9) MOX fuel pellet residence time or duration (y)
0	10) number of manual movements of MOX fuel pellets (e.g. between gloveboxes)
4248	11) average inventory of MOX fuel assemblies (kg Pu)
2124	12) throughput MOX fuel assemblies (kg Pu/y)
0.62	13) MOX fuel assembly residence time or duration prior to fuel load (y)
7	14) number of movements of MOX fuel assemblies (ASSUMES EX-REACTOR FUEL STORAGE)
220	15) staffing level of MOX fuel fabrication facility (number of persons)
1.12E+06	16) effort at MOX fuel fabrication facility (person-h/y)
98	17) average inventory of MOX fuel rods (kg Pu/y)
2124	18) throughput MOX fuel rods (kg Pu/y)
0.01	19) MOX fuel rod residence time (y)
0	20) number of manual movements of MOX fuel rods
	IX. Environmental, Safety, and Health
Y	1) requires Am removal (respond with "y" or "n")
OPTIONAL	1a) requires Am removal (respond with "y" or "n") - COULD BE LEFT IN FUEL, SEE SECTION 8.2-4
507	2) dose for fuel fab operations and intrasite transport (person-rem)
1311	2a) dose for fuel fab operations and intrasite transport without Am removal (person-rem)
31400	3) dose for reactor operations (person-rem)
110	3a) Additional dose for reactor operations with MOX (person-rem)
3384	4) low level waste generated for fuel fab operations (m^3)
16905	5) low level waste generated for reactor operations during Pu disposition (m^3)
47	6) mixed waste generated for fuel fab operations (m^3)
N/A	7) mixed waste generated for reactor operations during Pu disposition (m^3)
9328	8) spent fuel generated (no. of spent MOX fuel assemblies)
6251	9) TRU waste generated, excluding packaging (m^3)
6.25E+06	10) TRU waste generated, excluding packaging (kg)
4	11) number of spent fuel movements to end of disposition operations (including last 10 years)

Table 2.6 (continued)

	X. Other measures
Y	1) physical protection system meets standards in DOE orders (respond with "y" or "n")
Y	2) material control and accountability system meets standards in DOE orders ("y" or "n")
Y	3) is IAEA monitoring practical (i.e. nothing to prohibit monitoring) ("y" or "n")
0	4) number of reactors of similar design under construction in the former republics of the USSR
0	5) number of reactors of similar design in operation in the former republics of the USSR

2.2 CALCULATIONS PERFORMED

The model of a full MOX-fueled GE BWR-5, operated at a core power level of 3484 MWt was prepared and calculations were performed with the SAS2H, ORIGEN-S, and SAS1 modules of the SCALE code system.[2]

2.2.1 SAS2H MODEL

The SAS2H module was used to model the fuel depletion of the GE BWR-5 assembly, with cross sections being taken from the 27-energy-group SCALE depletion library known as 27BURNUPLIB. This library is based primarily on ENDF/B-IV, but has a large number of fission product nuclides with data taken from ENDF/B-V. Details can be found in Sects. M4 and M8 of the SCALE document.[2]

During the SAS2H calculations, cross sections for the ORIGEN-S library were updated for the 147 fission product nuclides given in Table 2.7. The cross section for the light element nuclide ^{59}Co was also updated since ^{60}Co is an important source of gamma radiation for structural materials. The actinides and burnable poison nuclides of the fuel for which cross sections were updated are given in Table 2.8.

The SAS2H module is described in detail in Sect. S2 of the SCALE document.[2] A summary description of the neutronics and depletion models is given here. The basic schema is to (1) start with nuclide densities for a fresh fuel assembly; (2) perform resonance self-shielding calculations with the BONAMI and NITAWL modules; (3) perform neutronics calculations to compute the flux that is then used to cell weight cross sections, (4) use the COUPLE module to update the cross sections in a binary ORIGEN-S library, and (5) use that library in an ORIGEN-S calculation to determine nuclide densities at the midpoint of the first irradiation interval. This process is repeated for each irradiation interval until cross-section-updated ORIGEN-S binary libraries have been produced for all intervals. Then, a final ORIGEN-S case is run for the entire irradiation history, using the updated libraries produced earlier, and includes any desired assembly cooling time, after which neutron and photon sources are calculated.

The neutronics calculations of step (3) above use a model comprised of two computational paths to develop the updated ORIGEN-S binary libraries used for the final depletion calculation. This two-path model is necessary to model the two-dimensional (2-D) effects of fuel assemblies that contain different kinds of fuel pins or water holes. This approach is used both for PWR and BWR assemblies.

The model used for the first computational path, called Path A, represents the fuel by an infinite lattice of fuel pins. The Path A cell model for the MOX fuel has a fuel pellet diameter of 1.0414 cm, a gap outer diameter (OD) of 1.06426 cm, a clad OD of 1.22682 cm, and a pitch of 1.6256 cm. Resonance self-

shielding cross-section processing is handled by the BONAMI and NITAWL modules. Then, a one-dimensional (1-D) discrete ordinates transport calculation of the neutron flux in a unit cell with white boundary conditions is performed by the XSDRNPM module. Cell-weighted cross sections produced by this Path A model are then used in the fuel region of a larger unit cell, modeled with a second computational path, called Path B, that represents part or all of a fuel assembly.

The use of cell-weighted cross sections from Path A for the fuel region of a larger unit cell in the Path B analysis is an approximate method for evaluating 2-D effects found in assemblies that contain different kinds of fuel rods (e.g., burnable poison) or water holes. In path B, BONAMI and NITAWL are again used for resonance self-shielding cross section processing, and XSDRNPM is used to compute the flux in the larger unit cell of Path B. The cell-weighted cross sections from the Path B model are then used by COUPLE to update the ORIGEN-S binary library.

For the MOX fuel cycle analysis, the Path B cell model describes the entire assembly and has a central moderator region (radius: 1.3614 cm), a guide tube (radius: 1.4630 cm), a second moderator region (radius: 1.8343 cm), a homogenized fuel region from the Path A cell (radius: 7.3372 cm), an outer moderator region (radius: 7.4733 cm), an assembly tube (radius: 7.7599 cm), and a final (in between assemblies) moderator region (radius: 8.5982 cm). These radii are derived by conserving the cross-sectional area of the square assembly as shown in Fig. 2.2.

Following the final ORIGEN-S depletion calculation of SAS2H, discharge isotopic compositions of the fuel and other actinides, the fission products, and the clad and structural materials were written to an output file by SAS2H.

Table 2.7. Fission products for which cross sections were updated

^{109}Ag	^{111}Ag	^{75}As	^{134}Ba	^{138}Ba
^{140}Ba	^{81}Br	^{110}Cd	^{111}Cd	^{112}Cd
^{113}Cd	^{114}Cd	^{115m}Cd	^{116}Cd	^{140}Ce
^{141}Ce	^{142}Ce	^{143}Ce	^{144}Ce	^{133}Cs
^{134}Cs	^{135}Cs	^{136}Cs	^{137}Cs	^{160}Dy
^{161}Dy	^{162}Dy	^{163}Dy	^{164}Dy	^{153}Eu
^{154}Eu	^{155}Eu	^{156}Eu	^{73}Ge	^{76}Ge
^{127}I	^{129}I	^{130}I	^{131}I	^{135}I
^{115}In	^{82}Kr	^{83}Kr	^{84}Kr	^{85}Kr
^{86}Kr	^{139}La	^{140}La	^{95}Mo	^{96}Mo
^{97}Mo	^{98}Mo	^{99}Mo	^{100}Mo	^{95}Nb
^{142}Nd	^{143}Nd	^{144}Nd	^{145}Nd	^{146}Nd
^{147}Nd	^{148}Nd	^{150}Nd	^{104}Pd	^{105}Pd
^{106}Pd	^{107}Pd	^{108}Pd	^{110}Pd	^{147}Pm
^{148}Pm	^{149}Pm	^{151}Pm	^{141}Pr	^{142}Pr
^{143}Pr	^{85}Rb	^{86}Rb	^{87}Rb	^{103}Rh
^{105}Rh	^{100}Ru	^{101}Ru	^{102}Ru	^{103}Ru
^{104}Ru	^{105}Ru	^{106}Ru	^{121}Sb	^{123}Sb
^{124}Sb	^{125}Sb	^{126}Sb	^{77}Se	^{78}Se
^{80}Se	^{147}Sm	^{148}Sm	^{149}Sm	^{150}Sm
^{151}Sm	^{152}Sm	^{153}Sm	^{154}Sm	^{115}Sn
^{116}Sn	^{117}Sn	^{118}Sn	^{119}Sn	^{120}Sn
^{122}Sn	^{123}Sn	^{124}Sn	^{125}Sn	^{89}Sr
^{90}Sr	^{181}Ta	^{159}Tb	^{160}Tb	^{99}Tc
^{122}Te	^{123}Te	^{124}Te	^{125}Te	^{126}Te
^{127m}Te	^{128}Te	^{129m}Te	^{130}Te	^{132}Te
^{128}Xe	^{129}Xe	^{130}Xe	^{131}Xe	^{132}Xe
^{133}Xe	^{134}Xe	^{135}Xe	^{136}Xe	^{89}Y
^{90}Y	^{91}Zr	^{92}Zr	^{93}Zr	^{94}Zr
^{95}Zr	^{96}Zr			

Table 2.8. Actinide and burnable poison nuclides for which cross sections were updated

^{233}U	^{235}U	^{238}U	^{237}Np	^{238}Pu
^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	^{241}Am
^{243}Am	^{154}Gd	^{155}Gd	^{156}Gd	^{157}Gd
^{158}Gd	^{160}Gd			

2.2.2 ORIGEN-S MODEL

Using the nuclide concentration from the SAS2H output file as input, ORIGEN-S calculations were performed to provide a decay analysis at cooling times of 0, 1, and 90 d; 1, 5, 10, 100, 300, 1,000, 10,000, 30,000, 100,000, and 250,000 years. Output tables from the ORIGEN-S calculations included nuclide concentrations (g), activities (Ci), total decay heat (W), air and water toxicities (m^3 to dilute to radioactivity concentration guides), neutron spectra from spontaneous fission and (α,n) reactions, and photon source spectra, including gamma lines, X rays, and bremsstrahlung from oxide fuel. The cutoff for listing nuclide concentrations was 0.01 g at a 10-year cooling time. All other tables had cutoffs of 0.05% of the table total at a 10-year cooling time. An additional ORIGEN-S case was run with an absolute cutoff of 10^{-35} for all tables to get discharge activities for nuclides of interest for repository criticality safety calculations, since the original cutoffs left out some nuclides for which information was desired.

2.2.3 SAS1 MODEL

Dose analyses were performed with the SAS1 (or SAS1X) module. Neutron and photon dose rates at the midplane between the ends of the bare spent fuel assembly were computed at 1 m from the outside surface of the shroud of the fuel bundle. Sources for the dose calculations were taken from the ORIGEN-S analysis. Neutron and photon dose rates were also computed for a model of a shipping cask similar to a Transnuclear TN12 cask. The doses were computed at the midplane of a cask holding 20 assemblies, both at the cask surface and at a 1-m radius from the surface.

3. THE LOW-ENRICHED-URANIUM CASE

3.1 FRESH FUEL CHARACTERISTICS AND OPERATING PARAMETERS

The MOX assembly design, described in Sect. 2.1, is a modification of the GE-9 fuel bundle, which has one large central water hole that displaces four fuel pins. The fuel pin data for the MOX calculations were taken both from ref. 1 and ref. 4. The data from ref. 4 were the dimensions for a fuel pin from a GE-5 fuel bundle, because the exact data for a GE-9 bundle were not available. For the LEU calculations, no LEU-specific data could be found in ref. 1. Since complete data for the GE-9 design were not available from ref. 4 (specifically, the number of Gd_2O_3 burnable poison pins), data for a GE-5 design were used instead. The major difference between the GE-9 and GE-5 design, for the purposes of the SAS2H model used in this work, is the number of water holes (one large central hole occupying four pin locations for the GE-9 design vs two water holes for the GE-5 design) and the number of burnable poison pins (unknown for the GE-9 design vs two for the GE-5 design). Because the GE-5 design has been widely used and constitutes a significant fraction of present-day fuel assemblies, it should be adequate as a basis for comparison with the MOX design of ref. 1 in terms of environmental impact.

3.2 CALCULATIONS PERFORMED

The same general calculational procedure as described in Sect. 2.2 for the MOX assembly was followed for the LEU assembly.

3.2.1 SAS2H MODEL

The changes for the SAS2H LEU model are as follows. The GE-5 design has 0.183 metric tonnes of heavy metal (MTHM) per assembly, whereas the GE-9 MOX design has 0.176 MTHM. The four burnable poison pins used in the LEU model had 2.17 wt % Gd_2O_3, as described in ref. 9. The specific power of the SAS2H LEU model was changed to 4.724 MW per assembly to achieve the same burnup as in the SAS2H MOX model (i.e., 37.6 GWd per MTHM). The irradiation periods were kept the same as in the MOX calculations. As for the MOX calculations, the neutronics calculations in SAS2H use a model comprised of two computational paths to better model the 2-D effects of fuel assemblies containing different kinds of fuel pins or water holes. The model for the first computational path (Path A) represents the fuel by an infinite lattice of fuel pins. The dimensions for the Path A unit cell were the same as for the MOX case. The cell-weighted cross sections from Path A are used for the fuel region of a larger unit cell in a second computational path (Path B). The Path B unit cell in the SAS2H LEU BWR model was based on one-quarter of a fuel bundle, centered around a single burnable poison pin. It consists of a burnable poison pin (radius: 0.53213 cm), surrounded by a Zircaloy cladding (outer radius: 0.61341 cm), a water moderator (radius: 0.91715 cm), a homogenized fuel mixture from the Path A unit cell (radius: 3.49239 cm), an assembly casing (outer radius: 3.64531 cm), and a final moderator region (radius: 4.08860 cm). These radii were derived by conserving the cross-sectional area of one-quarter of the square assembly.

The initial average enrichment for an assembly was chosen to be a value that was reasonable to achieve the desired burnup of 37.6 GWd per MTHM. A plot of average initial enrichment vs fuel burnup, using data from ref. 5, shows that an enrichment of 3.25 wt % ^{235}U matched this burnup. This enrichment is

nearly an exact match for the higher (3.23 wt %) of the two average enrichments actually used in reload studies for the Laguna Verde nuclear power plant.[9] Cross sections were updated for the same nuclides as in the MOX case (see Table 2.7).

As in the analysis of the MOX fuel cycle, discharge isotopic compositions of the fuel and other actinides, the fission products, and the clad and structural materials were written to an output file by SAS2H.

3.2.2 ORIGEN-S MODEL

The ORIGEN-S calculations were performed in exactly the same manner as for the MOX fuel cycle analysis.

3.2.3 SAS1 MODEL

The SAS1 calculations were performed in the same manner as for the MOX fuel cycle analysis.

4. SPENT FUEL CHARACTERISTICS AND COMPARISONS

This study seeks to find major differences between the characteristics of the spent fuel from a GE BWR-5 system burning weapons-grade plutonium and a GE BWR-5 system burning conventional (i.e., low-enriched) uranium fuel. The characteristics of the spent fuel are needed to assess hazards associated with handling, transportation, and storage. The characteristics of interest for comparison of the two kinds of assemblies are isotopic concentrations, nuclide activity, decay heat, gamma and neutron doses for bare and cask-shielded assemblies, and potential ingestion and inhalation hazards. The flux profiles within the fuel pellets are of interest for comparison with future experiments.

Tables 4.1 through 4.9, Tables 4.20, 4.22, and 4.23, and Figs. 4.1 through 4.13 describe the characteristics of a full MOX BWR-5 assembly. Those of an LEU BWR-5 assembly are found in Tables 4.10 through 4.12, Tables 4.14 through 4.19, Table 4.21, and Figs. 4.14 through 4.26. Table 4.13 contains data for both kinds of assemblies.

4.1 ISOTOPIC COMPOSITION OF SPENT FUEL

Table 4.1 and Figs. 4.1 through 4.3 show the actinide concentrations in the full MOX BWR-5 assembly. The concentrations for the LEU BWR-5 assembly are given in Table 4.10 and Figs. 4.14 through 4.16. Since the uranium assembly is charged with much more ^{235}U, it burns more ^{235}U, and will therefore contain more ^{236}U and ^{237}Np at discharge. At very long decay times, the amounts of ^{235}U, ^{236}U, and ^{237}Np increase significantly in the MOX assembly as compared with the LEU assembly, due to the alpha decay of ^{239}Pu, ^{240}Pu, and ^{241}Am (which decays from ^{241}Pu), with the ^{237}Np becoming absolutely greater in the MOX assembly.

The MOX assembly has about 1.8 times as much plutonium at discharge as the LEU assembly. This is due to the amount of plutonium present in each assembly at the start of irradiation, since the amount of ^{239}Pu generated during operation is small. However, the ^{239}Pu constitutes only 40% of the plutonium in the MOX assembly at discharge, whereas it was about 94% in the fresh fuel. All the americium and curium isotopes show higher concentrations at discharge in the MOX fuel. This is expected because of the greater amounts of plutonium initially present in the MOX assembly.

Light-element and fission-product concentrations are given in Table 4.2 for the MOX assembly and in Table 4.11 for the LEU assembly.

4.2 ACTIVITIES

The activities from the light elements, actinides, and fission products in a MOX BWR-5 assembly as a function of cooling time are shown in Table 4.3 and Fig. 4.4. For the LEU BWR-5 assembly, the comparable data are found in Table 4.12 and Fig. 4.17. Table 4.13 gives discharge activities for nuclides of interest in both assemblies. Most of the contribution to activity comes from actinides and fission products, with light elements contributing 1% or less.

The activity from structural materials is comparable for both assemblies. If everything else were equal, the light-element contribution to activity would only be proportional to burnup. The amount of light

elements per MTHM is the same for both assemblies, and the burnup is the same. However, because of the use of the BWR-5 design for LEU fuel, there is 0.183 MTHM in the LEU assembly and 0.176 MTHM in the MOX assembly, which causes slightly more light-element activity in the LEU assembly. A major contributor to light-element activity is ^{60}Co. The final amount of ^{60}Co depends on the initial amount of ^{59}Co in the assembly. The amount of ^{59}Co per MTHM of both assemblies was the same, having been set to a value that is considered typical for a BWR assembly.

Fission-product activities are comparable, with the LEU assembly having somewhat higher values (4 to 5%) than the MOX assembly. These ratios of fission product activity (LEU/MOX) are just about the same as the ratio of heavy-metal content (0.183 MTHM/0.176 MTHM = 1.04) for the MOX and LEU assemblies, since the burnups are equal. The slight differences from the value of 1.04 are due to differences in the fission yields of the uranium and plutonium mixtures in the two assemblies.

The actinide activity is slightly greater at discharge (4%) and 1-d decay (6%) for the LEU assembly, because of higher activities for the short-lived nuclide ^{239}Np and its daughter ^{239}U. For decay times from 1 year to 10,000 years, the larger quantities of long-lived actinides in the MOX assembly cause its actinide activity to be greater by about a factor of 2. Beyond that decay time, the ratio of MOX to LEU actinide activity slowly decreases to a value of about 1.1 at 250,000 years.

4.3 DECAY HEAT

The decay heat from the light elements, actinides, and fission products in the MOX BWR-5 assembly is shown in Table 4.3 and Fig. 4.5. The decay heat data for the LEU BWR-5 assembly are given in Table 4.12 and Fig. 4.18. The light-element decay heat for the LEU assembly is consistently slightly greater since the quantity of structural materials was set to be the same per unit loading of heavy metal (0.183 MTHM for the LEU assembly vs 0.176 MTHM for the MOX assembly) and the burnups are the same. The actinide decay heats are comparable at discharge, but are consistently larger for the MOX assembly at long decay times. The fission product decay heat is generally less in the MOX assembly for decay times less than 300 years, and greater past that time. The total decay heat at discharge is about 6% larger for the LEU assembly, since it comes primarily from fission products at that time. For cooling times of 1 year or greater, the total is greater for the MOX assembly and is comprised principally of actinide decay heat for times greater than 100 years.

4.4 DOSE RATES

Table 4.5 and Fig. 4.6 show the neutron dose rate as a function of decay time at a radial distance of 1 m from the axial midplane of a bare MOX assembly. The dose rate 1 m from a bare LEU assembly is given in Table 4.15 and Fig. 4.19. The dose rate from the MOX assembly is about 2.5 times greater than that from the LEU assembly until 30,000 years decay time, when the ratio drops to about 2.2. The greater dose from the MOX assembly is due to the greater amounts of curium and americium nuclides in that fuel compared with the LEU assembly.

Table 4.6 and Fig. 4.7 give the gamma dose rate as a function of decay time at a radial distance of 1 m from the axial midplane of a bare MOX assembly. The dose rate 1 m from a bare LEU assembly is found in Table 4.16 and Fig. 4.20. The gamma dose rates from both assemblies are comparable (ratios from 1.07 to 0.95) from discharge until 300 years. Then, the dose rate from the LEU assembly is significantly

less (ratios of 0.51 to 0.87) until 30,000 years decay time, after which it becomes greater (ratios of 1.15 to 1.27).

The neutron and gamma dose rates at the surface of the cask and at 1 m from the cask are found in Table 4.7 and Figs. 4.8 and 4.9 for the MOX assembly and in Table 4.17 and Figs. 4.21 and 4.22 for the LEU assembly. The major points to note are twofold: (1) the neutron dose rates are insignificant compared with the gamma dose rates and (2) the ratio of gamma dose rates follow about the same trend as for the bare assemblies.

4.5 POTENTIAL INHALATION AND INGESTION HAZARDS

The ORIGEN calculations give a rough estimate of the potential hazards from inhalation and ingestion of radionuclides in the fuel assemblies. These potential hazards are quantified as the amount of air or water necessary to dilute the amount of radionuclides to the rcg (radioactivity concentration guide) limits of the pre-1993 version of 10 CFR 20. It must be emphasized that these are potential hazards, since thorough pathway and dosimetry analyses would be necessary to estimate the doses and risks for particular exposure incidents. With that qualification in mind, the inhalation and ingestion hazards for the MOX assembly are given in Tables 4.8 and 4.9 and in Figs. 4.10 and 4.11. Hazards for the LEU assembly are shown in Tables 4.18 and 4.19 and in Figs. 4.23 and 4.24. Broadly speaking, the MOX assembly is potentially more hazardous for inhalation because of the greater quantities of alpha-emitting and bone-seeking actinides. The potential ingestion hazard is slightly greater for the nuclides from the LEU assembly because of the importance of light-element and fission-product nuclides.

4.6 NUCLIDES OF INTEREST FOR CRITICALITY SAFETY IN A REPOSITORY

Assessment of repository safety must include criticality analyses. Earlier work[10,11] has determined the principal nuclides of interest (e.g., fissile nuclides, neutron-emitting actinides, and nuclides with significant absorption cross sections). Quantities of these nuclides as a function of decay time are presented in Table 4.20 for the MOX assembly and Table 4.21 for the LEU assembly.

4.7 RADIAL FLUX DISTRIBUTIONS WITHIN THE FUEL PELLETS

During future work in the fissile materials disposition program, measurements are planned of the burnup as a function of radial distance across a fuel pellet. To allow for future comparison to those data, plots are given here of the radial flux distributions across the fuel pins, as computed by XSDRNPM during the Path A calculation of the SAS2H module. These calculations were performed at the midpoints of the cycles for which ORIGEN-S binary libraries were generated. In the plots, the thermal flux is taken to be the sum of the flux in groups 15 through 27 (energies below 3.05 eV) and the fast flux is taken to be the sum of the flux in groups 1 through 14 (energies above 3.05 eV). The 3.05 eV boundary between groups 14 and 15 is the upper limit of thermal upscatter in the 27-group SCALE depletion library.

The thermal and fast flux in the MOX fuel pin for burnups corresponding to the six ORIGEN-S libraries generated for the MOX assembly are shown in Figs. 4.12 and 4.13. The thermal and fast flux for the LEU fuel pin are given in Figs. 4.25 and 4.26. Note that the flux curves for the different burnups are not normalized to the same fission rate, so they cannot be compared with each other, even for the same

assembly model. The datum of interest for each flux curve is its behavior as a function of radius within the fuel pellet. Note also that the entire pellet is treated as a single fuel zone; radially dependent depletion is not possible with SAS2H.

4.8 COMPARISONS OF ORNL AND GE DEPLETION AND DECAY RESULTS

This section gives comparisons between the depletion results of the discharged BWR-5 MOX assembly as calculated by ORNL and General Electric. The ORNL results were computed with SAS2H and ORIGEN-S from SCALE version 4.2p, using the models described in Sect. 2.2. The methods used by GE are described briefly below.

The methods and data used by GE to model the heterogeneous BWR-5 MOX core are summarized here in four categories from information in ref. 1: (1) the 3-D nuclear model, (2) the lattice physics methods, (3) the thermal-hydraulic model, and (4) the model qualification. These four categories can be described further as:

1. The PANACEA BWR core simulation code uses a 3-D coupled nuclear and thermal-hydraulic, time-independent model of power distributions and thermal performance. It takes into account control rod position, coolant flow, and other significant hydraulic, operational, and design parameters.

2. The TGBLA code applies lattice physics methods to compute few-group neutron cross sections. It performs neutron spectra calculations and determines fuel rod power distributions while considering the effects of fuel exposure in tracking up to 100 nuclides. The cross sections are fed into the PANACEA code.

3. The thermal-hydraulic model includes void-quality correlation, bypass region calculations, total core energy balance, and thermal limits calculations. This model is coupled to the neutronics model of PANACEA.

4. The models applied by GE were qualified by comparisons against data obtained from three BWRs, using nine different operating fuel cycles. The average computed k_{eff} was 1.002 with a root-mean-square error of 0.002.

Table 4.22 presents the discharged quantity (g per assembly) of the plutonium isotopes, the total plutonium, and the fraction of plutonium in the discharged heavy metal as computed by the SAS2H/ORIGEN-S model (from Table 4.1) and the PANACEA/TGBLA models (from Table 2.6). Note that the difference for ^{239}Pu was only -0.4%. The largest difference was 9.3% for ^{242}Pu. The comparison for total plutonium is quite good, with a difference of 0.9%.

Table 4.23 gives the comparisons of decay heat (W per assembly) for several times after shutdown. The difference is only 2.2% at shutdown, -10.2% after 1 year, and about 6% at 10 and 100 years.

Gamma dose rates were computed at different distances from the bare assembly by ORNL and GE; therefore, they are not easily comparable. However, if similar photon spectra data were used, gamma dose rates at the same distance from the assembly would tend to follow the decay heat comparisons.

Overall, there are no highly significant discrepancies between the results computed by ORNL and GE.

4.9 SUMMARY

Burnup and decay calculations have been performed to provide data needed for decay heat, radiation shielding, criticality safety, and biological hazard analyses for a MOX fuel cycle based on an existing BWR design that has been proposed for use in the Fissile Materials Disposition Program. To provide a basis for comparison, the same calculations have been performed for a low-enriched-uranium fuel cycle case typical of present-day BWR designs. In addition to quantifying the nuclide inventories themselves, the data needed to evaluate other hazards have been discussed. These tabulations of data provide the input data for analyses on which to base decisions for selecting a method for plutonium disposition.

In terms of gross hazard parameters characterizing a spent fuel assembly (e.g., decay heat and overall activity), the MOX and LEU assemblies are not greatly different. In terms of actual nuclide inventories, the weapons-grade plutonium used to fuel the MOX assembly is definitely not weapons grade at discharge.

One item of interest not addressed in the present work is the uncertainty associated with the calculated quantities. Future work might include a thorough uncertainty analysis, by which is meant a determination of the uncertainties in input quantities, and their propagation through the suite of calculations.

Table 4.1. Actinide concentrations in a BWR-5 MOX assembly (g per assembly)

Nuclide	Time after discharge						
	Discharge	1 d	1 year	5 year	10 years	30 years	100 years
^{210}Pb	6.25E-12	6.26E-12	6.55E-12	6.04E-12	7.28E-12	1.17E-10	8.04E-09
^{214}Bi	3.73E-19	3.74E-19	8.18E-19	8.92E-18	4.79E-17	9.70E-16	2.89E-14
^{210}Po	7.61E-14	7.63E-14	1.05E-13	1.03E-13	1.24E-13	2.00E-12	1.37E-10
^{225}Ra	1.48E-12	1.42E-12	1.88E-13	1.94E-13	2.06E-13	3.19E-13	1.99E-12
^{226}Ra	1.68E-11	1.69E-11	3.65E-11	3.98E-10	2.14E-09	4.33E-08	1.29E-06
^{227}Ac	5.85E-11	5.86E-11	7.94E-11	1.77E-10	3.27E-10	1.16E-09	5.35E-09
^{229}Th	3.69E-08	3.70E-08	3.71E-08	3.83E-08	4.08E-08	6.31E-08	3.95E-07
^{230}Th	1.40E-06	1.41E-06	3.11E-06	1.95E-05	6.16E-05	4.53E-04	4.04E-03
^{231}Pa	1.06E-06	1.06E-06	1.16E-06	1.57E-06	2.08E-06	4.11E-06	1.14E-05
^{233}U	3.64E-05	3.64E-05	4.74E-05	9.02E-05	1.43E-04	3.93E-04	2.04E-03
^{234}U	4.57E-01	4.58E-01	7.78E-01	2.18E+00	3.88E+00	1.00E+01	2.54E+01
^{235}U	1.04E+02	1.04E+02	1.04E+02	1.04E+02	1.04E+02	1.05E+02	1.08E+02
^{236}U	4.39E+01	4.39E+01	4.40E+01	4.45E+01	4.51E+01	4.75E+01	5.60E+01
^{238}U	1.68E+05	1.68E+05	1.68E+05	1.68E+05	1.68E+05	1.68E+05	1.68E+05
^{237}Np	3.08E+01	3.09E+01	3.13E+01	3.20E+01	3.36E+01	4.53E+01	1.03E+02
^{239}Np	1.13E+01	8.47E+00	5.13E-05	5.13E-05	5.13E-05	5.12E-05	5.08E-05
^{238}Pu	3.66E+01	3.67E+01	4.40E+01	4.46E+01	4.29E+01	3.67E+01	2.13E+01
^{239}Pu	1.40E+03	1.41E+03	1.42E+03	1.42E+03	1.42E+03	1.41E+03	1.41E+03
^{240}Pu	1.16E+03	1.16E+03	1.16E+03	1.16E+03	1.16E+03	1.17E+03	1.16E+03
^{241}Pu	5.57E+02	5.57E+02	5.31E+02	4.37E+02	3.44E+02	1.31E+02	4.44E+00
^{242}Pu	2.43E+02	2.43E+02	2.43E+02	2.43E+02	2.43E+02	2.43E+02	2.43E+02
^{241}Am	4.20E+01	4.21E+01	6.82E+01	1.61E+02	2.53E+02	4.54E+02	5.22E+02
^{242m}Am	1.17E+00	1.17E+00	1.16E+00	1.14E+00	1.11E+00	1.01E+00	7.14E-01
^{243}Am	5.97E+01	5.97E+01	5.97E+01	5.97E+01	5.96E+01	5.95E+01	5.91E+01
^{242}Cm	9.70E+00	9.70E+00	2.07E+00	7.13E-03	2.92E-03	2.64E-03	1.87E-03
^{243}Cm	3.99E-01	3.99E-01	3.90E-01	3.54E-01	3.13E-01	1.93E-01	3.51E-02
^{244}Cm	2.01E+01	2.01E+01	1.94E+01	1.66E+01	1.37E+01	6.38E+00	4.37E-01
^{245}Cm	9.74E-01	9.74E-01	9.74E-01	9.74E-01	9.73E-01	9.72E-01	9.66E-01
^{246}Cm	1.64E-01	1.64E-01	1.64E-01	1.63E-01	1.63E-01	1.63E-01	1.61E-01

Table 4.1 (continued)

Nuclide	Time after discharge (years)						
	300	1000	3000	10000	30000	100000	250000
^{210}Pb	2.43E-07	5.20E-06	4.28E-05	2.55E-04	8.94E-04	2.04E-03	2.43E-03
^{214}Bi	5.46E-13	8.99E-12	7.40E-11	4.41E-10	1.55E-09	3.53E-09	4.20E-09
^{210}Po	4.12E-09	8.83E-08	7.27E-07	4.34E-06	1.52E-05	3.46E-05	4.13E-05
^{225}Ra	3.19E-11	8.14E-10	1.21E-08	1.45E-07	8.76E-07	3.51E-06	7.21E-06
^{226}Ra	2.44E-05	4.01E-04	3.30E-03	1.97E-02	6.91E-02	1.57E-01	1.88E-01
^{227}Ac	1.93E-08	7.86E-08	3.03E-07	1.69E-06	8.75E-06	3.34E-05	4.62E-05
^{229}Th	6.31E-06	1.61E-04	2.39E-03	2.87E-02	1.73E-01	6.96E-01	1.43E+00
^{230}Th	2.40E-02	1.13E-01	3.68E-01	1.22E+00	3.29E+00	7.51E+00	8.97E+00
^{231}Pa	3.30E-05	1.20E-04	4.63E-04	2.59E-03	1.34E-02	5.11E-02	7.07E-02
^{233}U	1.34E-02	1.02E-01	4.73E-01	1.83E+00	5.46E+00	1.58E+01	2.89E+01
^{234}U	4.22E+01	4.68E+01	4.66E+01	4.59E+01	4.39E+01	3.77E+01	2.78E+01
^{235}U	1.16E+02	1.43E+02	2.20E+02	4.57E+02	9.33E+02	1.47E+03	1.55E+03
^{236}U	8.00E+01	1.60E+02	3.58E+02	7.99E+02	1.15E+03	1.20E+03	1.19E+03
^{238}U	1.68E+05	1.68E+05	1.68E+05	1.68E+05	1.68E+05	1.68E+05	1.68E+05
^{237}Np	2.45E+02	4.98E+02	6.15E+02	6.19E+02	6.15E+02	6.02E+02	5.73E+02
^{239}Np	4.99E-05	4.67E-05	3.87E-05	2.00E-05	3.05E-06	4.23E-09	1.32E-12
^{238}Pu	4.54E+00	2.80E-02	6.20E-07	7.01E-22	0.00E+00	0.00E+00	0.00E+00
^{239}Pu	1.41E+03	1.38E+03	1.31E+03	1.09E+03	6.28E+02	8.46E+01	1.13E+00
^{240}Pu	1.14E+03	1.06E+03	8.57E+02	4.09E+02	4.95E+01	3.04E-02	1.52E-08
^{241}Pu	1.86E-03	1.49E-03	1.27E-03	7.17E-04	1.40E-04	4.65E-07	2.26E-12
^{242}Pu	2.43E+02	2.43E+02	2.42E+02	2.39E+02	2.30E+02	2.02E+02	1.53E+02
^{241}Am	3.82E+02	1.24E+02	5.09E+00	2.17E-02	4.23E-03	1.40E-05	7.17E-11
^{242m}Am	2.67E-01	8.55E-03	4.59E-07	5.22E-22	0.00E+00	0.00E+00	0.00E+00
^{243}Am	5.80E+01	5.43E+01	4.50E+01	2.33E+01	3.55E+00	4.92E-03	1.53E-06
^{242}Cm	7.01E-04	2.25E-05	1.21E-09	1.37E-24	0.00E+00	0.00E+00	0.00E+00
^{243}Cm	2.71E-04	1.09E-11	8.16E-33	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{244}Cm	2.06E-04	4.69E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{245}Cm	9.51E-01	8.98E-01	7.63E-01	4.31E-01	8.43E-02	2.79E-04	1.36E-09
^{246}Cm	1.57E-01	1.41E-01	1.05E-01	3.78E-02	2.02E-03	7.09E-08	3.35E-17

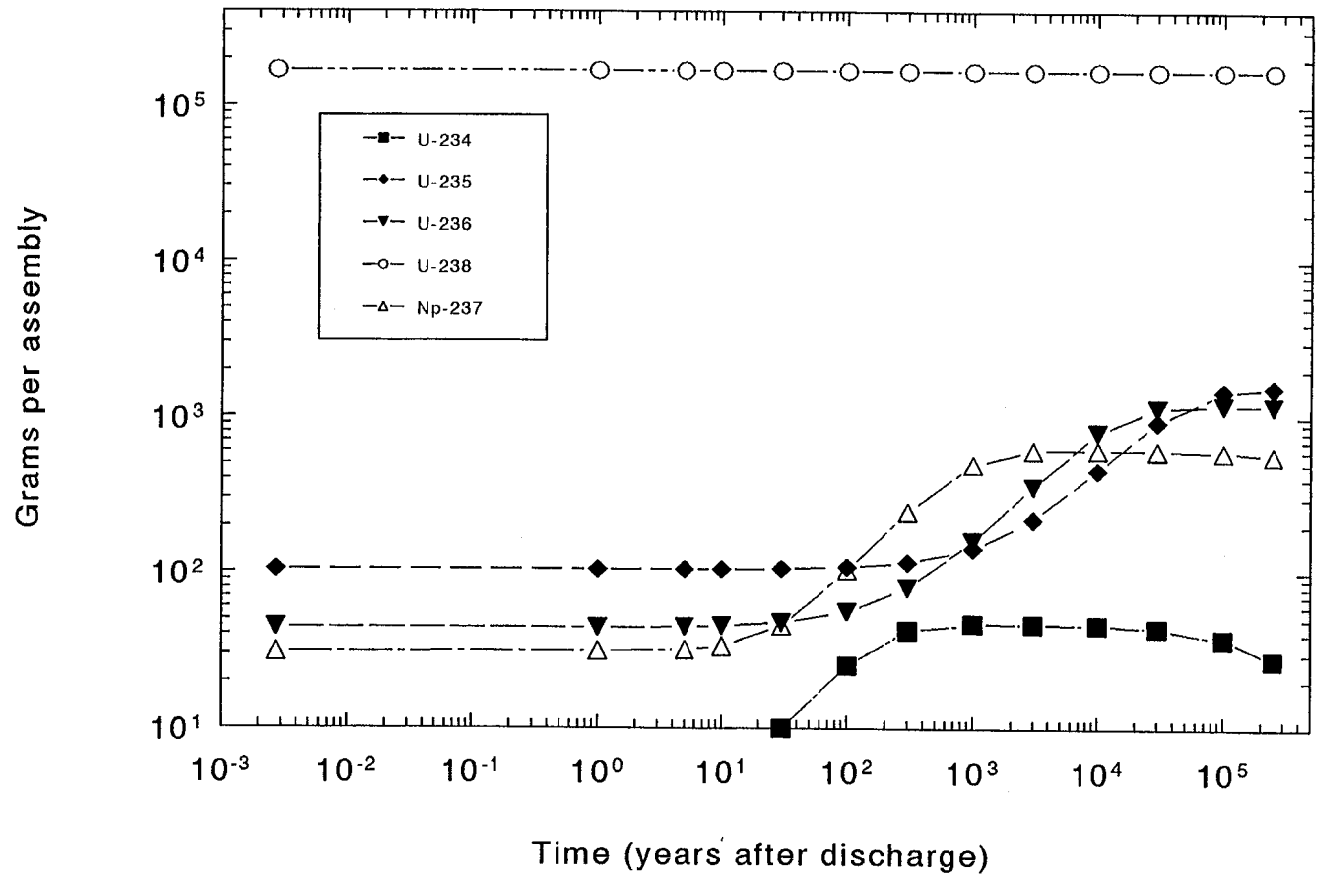


Fig. 4.1. Uranium and neptunium decay from a BWR-5 MOX fuel assembly.

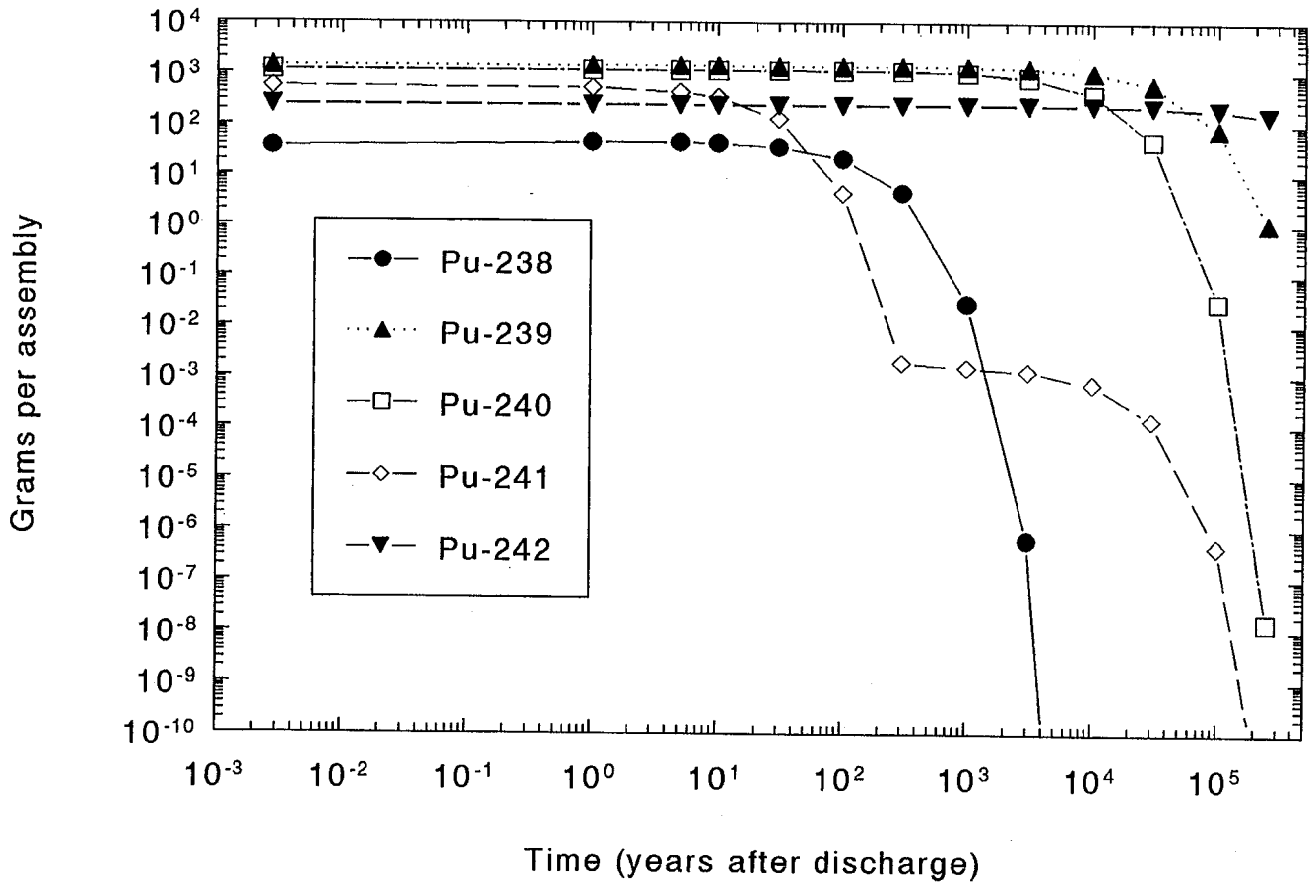


Fig. 4.2. Plutonium decay from a BWR-5 MOX fuel assembly.

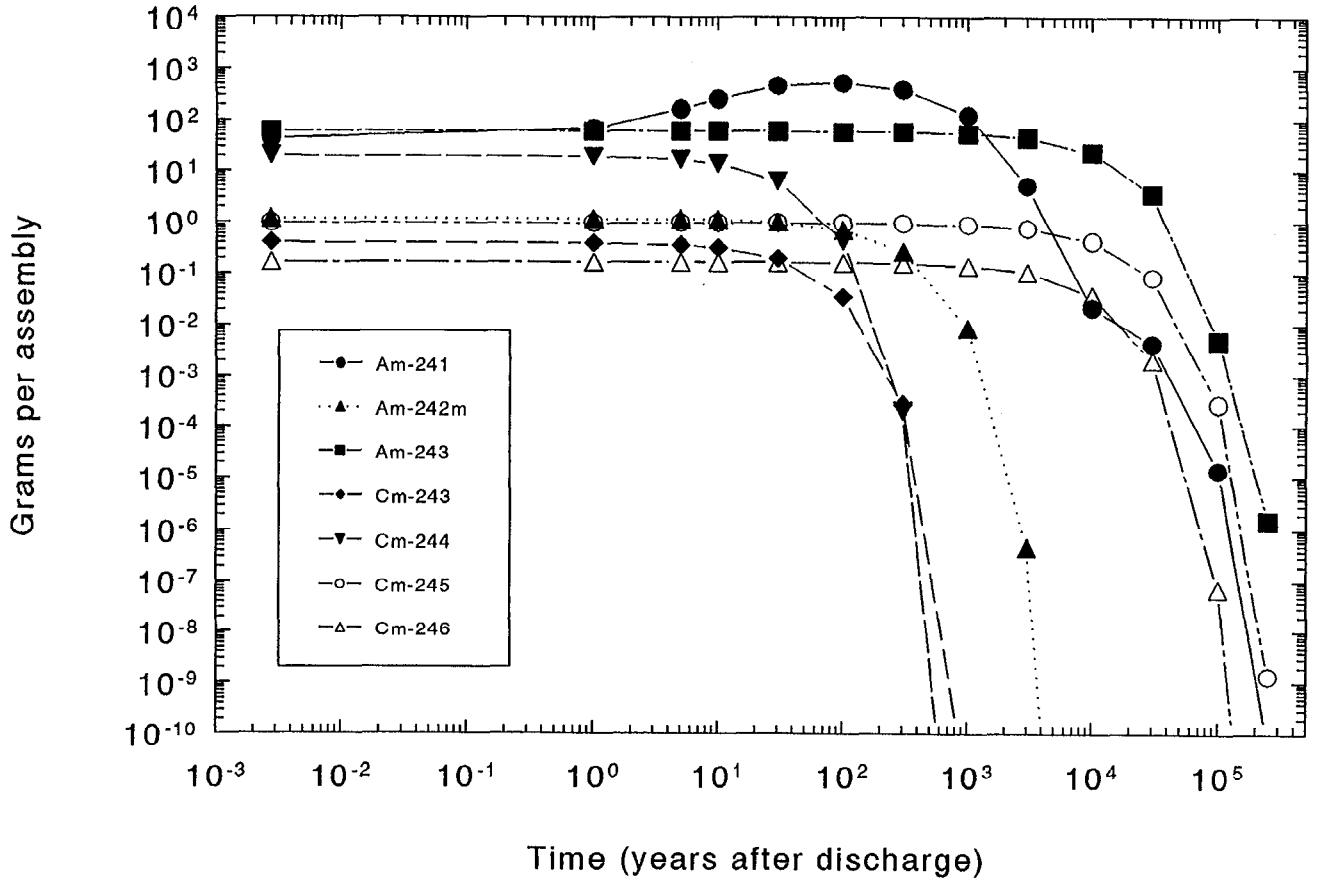


Fig. 4.3. Americium and curium decay from a BWR-5 MOX fuel assembly.

Table 4.2. Light-element and fission-product concentrations in a BWR-5 MOX assembly (g per assembly)

Nuclide	Time after discharge						
	Discharge	1 d	1 year	5 year	10 years	30 years	100 years
^{3}H	1.05E-02	1.05E-02	9.92E-03	7.92E-03	5.98E-03	1.94E-03	3.79E-05
^{14}C	7.11E-03	7.11E-03	7.11E-03	7.10E-03	7.10E-03	7.08E-03	7.02E-03
^{60}Co	4.89E-01	4.89E-01	4.29E-01	2.53E-01	1.31E-01	9.45E-03	9.47E-07
^{59}Ni	2.27E+00	2.27E+00	2.27E+00	2.27E+00	2.27E+00	2.27E+00	2.27E+00
^{63}Ni	4.42E-01	4.42E-01	4.39E-01	4.27E-01	4.12E-01	3.59E-01	2.21E-01
^{85}Kr	2.49E+00	2.49E+00	2.33E+00	1.80E+00	1.30E+00	3.58E-01	3.87E-03
^{90}Sr	5.35E+01	5.35E+01	5.22E+01	4.73E+01	4.18E+01	2.56E+01	4.56E+00
^{90}Y	1.57E-02	1.54E-02	1.36E-02	1.23E-02	1.09E-02	6.64E-03	1.18E-03
^{94}Nb	1.65E-04	1.65E-04	1.65E-04	1.65E-04	1.65E-04	1.65E-04	1.64E-04
^{95}Mo	1.25E+02	1.25E+02	1.33E+02	1.34E+02	1.34E+02	1.34E+02	1.34E+02
^{99}Tc	1.52E+02	1.52E+02	1.52E+02	1.52E+02	1.52E+02	1.52E+02	1.52E+02
^{101}Ru	1.63E+02	1.63E+02	1.63E+02	1.63E+02	1.63E+02	1.63E+02	1.63E+02
^{106}Ru	3.47E+01	3.46E+01	1.75E+01	1.15E+00	3.18E-02	4.61E-08	8.96E-29
^{103}Rh	1.23E+02	1.23E+02	1.29E+02	1.29E+02	1.29E+02	1.29E+02	1.29E+02
^{106}Rh	3.36E-05	3.21E-05	1.63E-05	1.07E-06	3.54E-08	4.28E-14	8.32E-35
^{105}Pd	1.45E+02	1.46E+02	1.46E+02	1.46E+02	1.46E+02	1.46E+02	1.46E+02
^{109}Ag	3.71E+01	3.71E+01	3.71E+01	3.71E+01	3.71E+01	3.71E+01	3.71E+01
^{110m}Ag	2.62E-01	2.61E-01	9.51E-02	1.65E-03	1.04E-05	1.63E-14	0.00E+00
^{131}Xe	9.42E+01	9.43E+01	9.53E+01	9.53E+01	9.53E+01	9.53E+01	9.53E+01
^{133}Cs	2.24E+02	2.24E+02	2.25E+02	2.25E+02	2.25E+02	2.25E+02	2.25E+02
^{134}Cs	2.04E+01	2.04E+01	1.46E+01	3.81E+00	7.09E-01	8.52E-04	5.13E-14
^{137}Cs	2.46E+02	2.46E+02	2.41E+02	2.19E+02	1.96E+02	1.23E+02	2.44E+01
^{137m}Ba	3.78E-05	3.76E-05	3.68E-05	3.35E-05	2.99E-05	1.88E-05	3.73E-06
^{144}Ce	3.27E+01	3.26E+01	1.35E+01	3.85E-01	4.53E-03	8.68E-11	0.00E+00
^{141}Pr	2.01E+02	2.02E+02	2.08E+02	2.08E+02	2.08E+02	2.08E+02	2.08E+02
^{144}Pr	1.39E-03	1.37E-03	5.67E-04	1.62E-05	1.91E-07	3.65E-15	0.00E+00
^{143}Nd	1.38E+02	1.38E+02	1.40E+02	1.40E+02	1.40E+02	1.40E+02	1.40E+02
^{145}Nd	1.16E+02	1.16E+02	1.16E+02	1.16E+02	1.16E+02	1.16E+02	1.16E+02
^{147}Pm	2.55E+01	2.56E+01	2.03E+01	7.07E+00	1.89E+00	9.57E-03	8.87E-11
^{147}Sm	2.08E+01	2.08E+01	2.69E+01	4.02E+01	4.54E+01	4.73E+01	4.73E+01
^{149}Sm	4.44E-01	4.93E-01	6.28E-01	6.28E-01	6.28E-01	6.28E-01	6.28E-01
^{150}Sm	6.17E+01	6.17E+01	6.17E+01	6.17E+01	6.17E+01	6.17E+01	6.17E+01
^{151}Sm	3.39E+00	3.40E+00	3.40E+00	3.30E+00	3.17E+00	2.72E+00	1.59E+00
^{152}Sm	3.11E+01	3.11E+01	3.11E+01	3.11E+01	3.11E+01	3.11E+01	3.11E+01
^{151}Eu	6.90E-03	6.97E-03	3.32E-02	1.36E-01	2.61E-01	7.14E-01	1.85E+00
^{153}Eu	2.87E+01	2.87E+01	2.89E+01	2.89E+01	2.89E+01	2.89E+01	2.89E+01
^{154}Eu	1.15E+01	1.15E+01	1.06E+01	7.68E+00	5.13E+00	1.02E+00	2.60E-03
^{155}Gd	1.39E-01	1.41E-01	7.45E-01	2.44E+00	3.54E+00	4.49E+00	4.54E+00
^{157}Gd	1.65E-01	1.67E-01	1.68E-01	1.68E-01	1.68E-01	1.68E-01	1.68E-01

Table 4.2 (continued)

Nuclide	Time after discharge (years)						
	300	1000	3000	10000	30000	100000	250000
^{3}H	4.96E-10	4.02E-27	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{14}C	6.85E-03	6.29E-03	4.94E-03	2.12E-03	1.88E-04	3.95E-08	5.18E-16
^{60}Co	3.55E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{59}Ni	2.26E+00	2.25E+00	2.21E+00	2.07E+00	1.72E+00	9.00E-01	2.25E-01
^{63}Ni	5.53E-02	4.34E-04	4.19E-10	3.72E-31	0.00E+00	0.00E+00	0.00E+00
^{85}Kr	9.36E-09	2.06E-28	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{90}Sr	3.31E-02	1.08E-09	4.39E-31	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{90}Y	8.60E-06	2.80E-13	1.14E-34	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{94}Nb	1.63E-04	1.59E-04	1.49E-04	1.17E-04	5.92E-05	5.42E-06	3.23E-08
^{95}Mo	1.34E+02	1.34E+02	1.34E+02	1.34E+02	1.34E+02	1.34E+02	1.34E+02
^{99}Tc	1.52E+02	1.52E+02	1.51E+02	1.47E+02	1.38E+02	1.10E+02	6.70E+01
^{101}Ru	1.63E+02	1.63E+02	1.63E+02	1.63E+02	1.63E+02	1.63E+02	1.63E+02
^{106}Ru	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{103}Rh	1.29E+02	1.29E+02	1.29E+02	1.29E+02	1.29E+02	1.29E+02	1.29E+02
^{106}Rh	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{105}Pd	1.46E+02	1.46E+02	1.46E+02	1.46E+02	1.46E+02	1.46E+02	1.46E+02
^{109}Ag	3.71E+01	3.71E+01	3.71E+01	3.71E+01	3.71E+01	3.71E+01	3.71E+01
^{110m}Ag	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{131}Xe	9.53E+01	9.53E+01	9.53E+01	9.53E+01	9.53E+01	9.53E+01	9.53E+01
^{133}Cs	2.25E+02	2.25E+02	2.25E+02	2.25E+02	2.25E+02	2.25E+02	2.25E+02
^{134}Cs	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{137}Cs	2.40E-01	2.27E-08	1.93E-28	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{137m}Ba	3.67E-08	2.47E-15	2.95E-35	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{144}Ce	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{141}Pr	2.08E+02	2.08E+02	2.08E+02	2.08E+02	2.08E+02	2.08E+02	2.08E+02
^{144}Pr	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{143}Nd	1.40E+02	1.40E+02	1.40E+02	1.40E+02	1.40E+02	1.40E+02	1.40E+02
^{145}Nd	1.16E+02	1.16E+02	1.16E+02	1.16E+02	1.16E+02	1.16E+02	1.16E+02
^{147}Pm	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{147}Sm	4.73E+01	4.73E+01	4.73E+01	4.73E+01	4.73E+01	4.73E+01	4.73E+01
^{149}Sm	6.28E-01	6.28E-01	6.28E-01	6.28E-01	6.28E-01	6.28E-01	6.28E-01
^{150}Sm	6.17E+01	6.17E+01	6.17E+01	6.17E+01	6.17E+01	6.17E+01	6.17E+01
^{151}Sm	3.40E-01	1.55E-03	3.16E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{152}Sm	3.11E+01	3.11E+01	3.11E+01	3.11E+01	3.11E+01	3.11E+01	3.11E+01
^{151}Eu	3.09E+00	3.43E+00	3.43E+00	3.43E+00	3.43E+00	3.43E+00	3.43E+00
^{153}Eu	2.89E+01	2.89E+01	2.89E+01	2.89E+01	2.89E+01	2.89E+01	2.89E+01
^{154}Eu	3.53E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{155}Gd	4.54E+00	4.54E+00	4.54E+00	4.54E+00	4.54E+00	4.54E+00	4.54E+00
^{157}Gd	1.68E-01	1.68E-01	1.68E-01	1.68E-01	1.68E-01	1.68E-01	1.68E-01

Table 4.3. Radioactivity from a BWR-5 MOX assembly (Ci per assembly)

Time after discharge (years if not specified)	Light elements	Actinides	Fission products	Total
Discharge	1.07E+05	5.58E+06	2.07E+07	2.64E+07
1 d	5.85E+04	2.12E+06	4.17E+06	6.35E+06
1	3.10E+03	6.47E+04	3.15E+05	3.83E+05
5	6.00E+02	4.83E+04	7.67E+04	1.26E+05
10	2.45E+02	3.87E+04	5.06000E+04	8.95E+04
30	3.22E+01	1.66E+04	2.86000E+04	4.52E+04
100	1.31E+01	3.05E+03	5.47000E+03	8.53E+03
300	3.49E+00	1.77E+03	6.28000E+01	1.84E+03
1000	3.53E-01	7.78E+02	3.73000E+00	7.82E+02
3000	3.19E-01	3.14E+02	3.66000E+00	3.18E+02
10000	2.94E-01	1.73E+02	3.56000E+00	1.77E+02
30000	2.56E-01	5.50E+01	3.31000E+00	5.86E+01
100000	1.86E-01	1.02E+01	2.63000E+00	1.30E+01
250000	1.247E-01	6.32E+00	1.72000E+00	8.16E+00

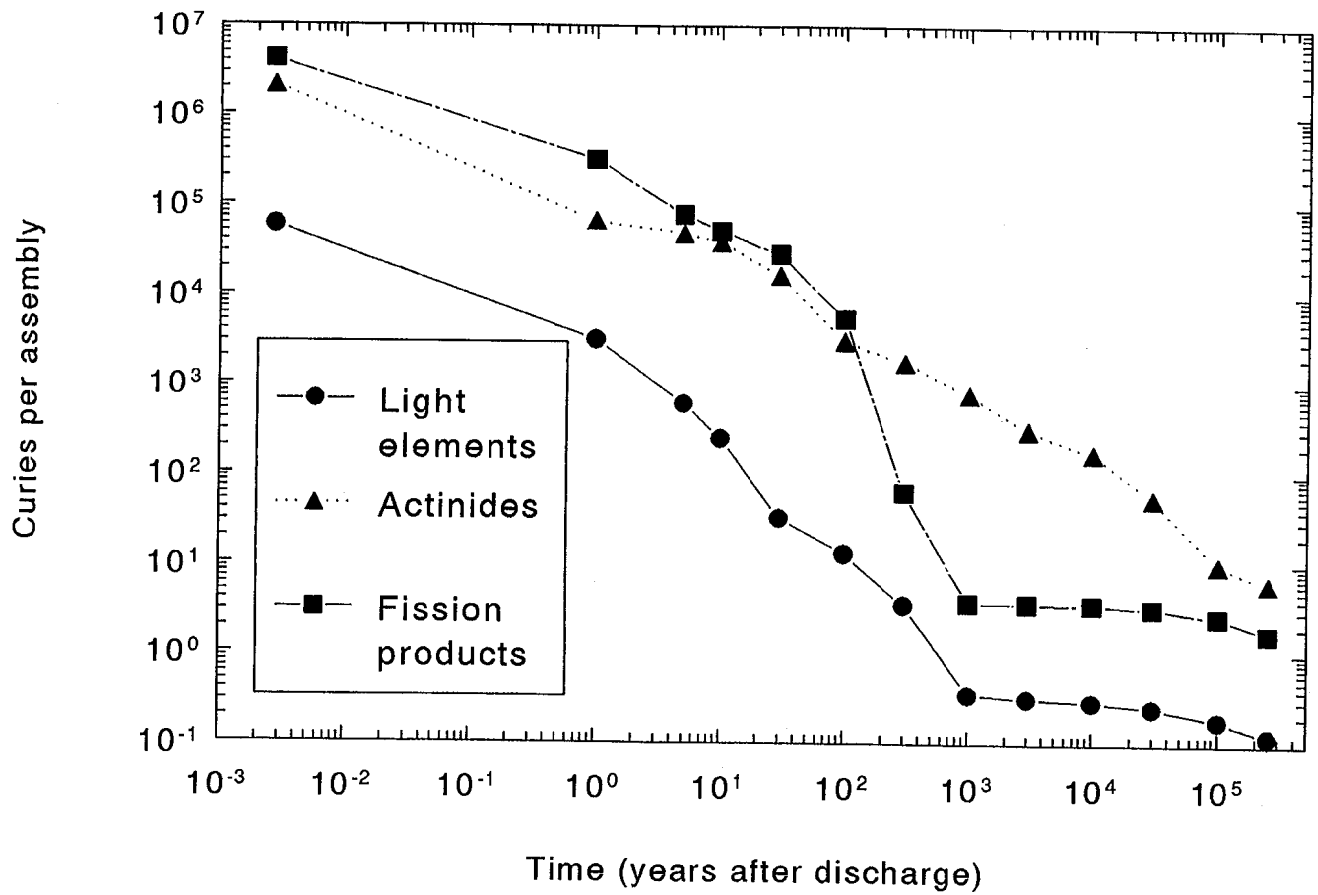


Fig. 4.4. Activity from a BWR-5 MOX fuel assembly.

Table 4.4. Decay heat from a BWR-5 MOX assembly (W per assembly)

Time after discharge (years if not specified)	Light elements	Actinides	Fission products	Total
Discharge	5.64E+02	1.56E+04	2.51E+05	2.67E+05
1 d	2.70E+02	6.41E+03	1.87E+04	2.54E+04
1	1.29E+01	3.52E+02	1.34E+03	1.70E+03
5	4.79E+00	1.05E+02	2.59E+02	3.69E+02
10	2.39E+00	1.06E+02	1.47E+02	2.55E+02
30	1.67E-01	1.03E+02	7.86E+01	1.82E+02
100	1.40E-03	8.48E+01	1.46E+01	9.94E+01
300	3.59E-04	5.76E+01	1.38E-01	5.77E+01
1000	3.55E-05	2.49E+01	4.83E-03	2.49E+01
3000	3.10E-05	9.55E+00	4.76E-03	9.55E+00
10000	2.68E-05	5.23E+00	4.57E-03	5.23E+00
30000	2.29E-05	1.66E+00	4.07E-03	1.66E+00
100000	1.92E-05	2.78E-01	2.74E-03	2.81E-01
250000	1.58E-05	1.49E-01	1.27E-03	1.50E-01

Table 4.5 Neutron dose rate 1 m from a bare BWR-5 MOX assembly at the axial midplane

Time after discharge (years if not specified)	Dose rate (rem h^{-1})
Discharge	2.47E-01
1 d	2.47E-01
1	1.44E-01
5	1.01E-01
10	8.38E-02
100	5.44E-03
300	2.41E-03
1000	1.80E-03
10000	6.90E-04
30000	2.74E-04
100000	1.95E-04
250000	1.47E-04

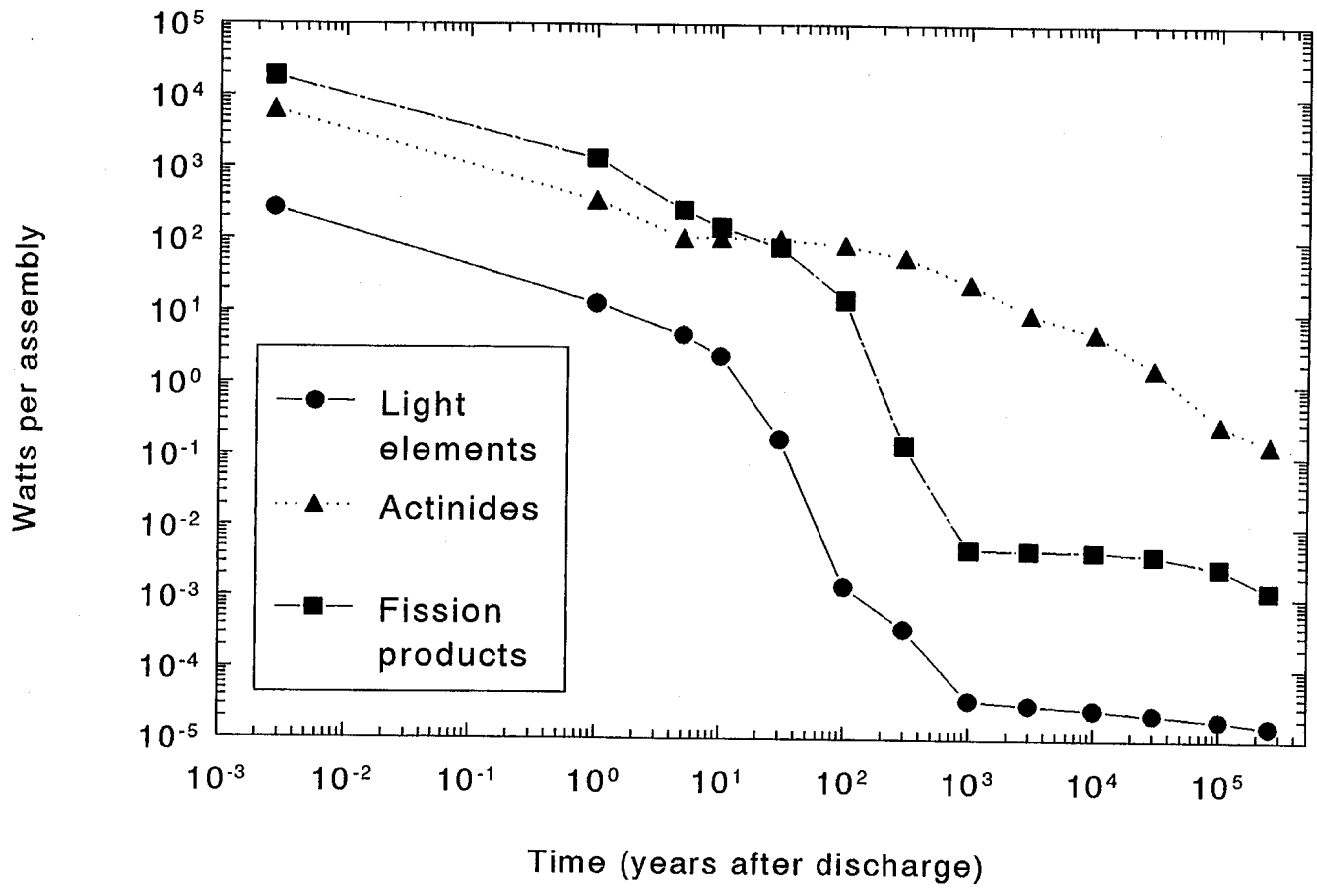


Fig. 4.5. Decay heat from a BWR-5 MOX fuel assembly.

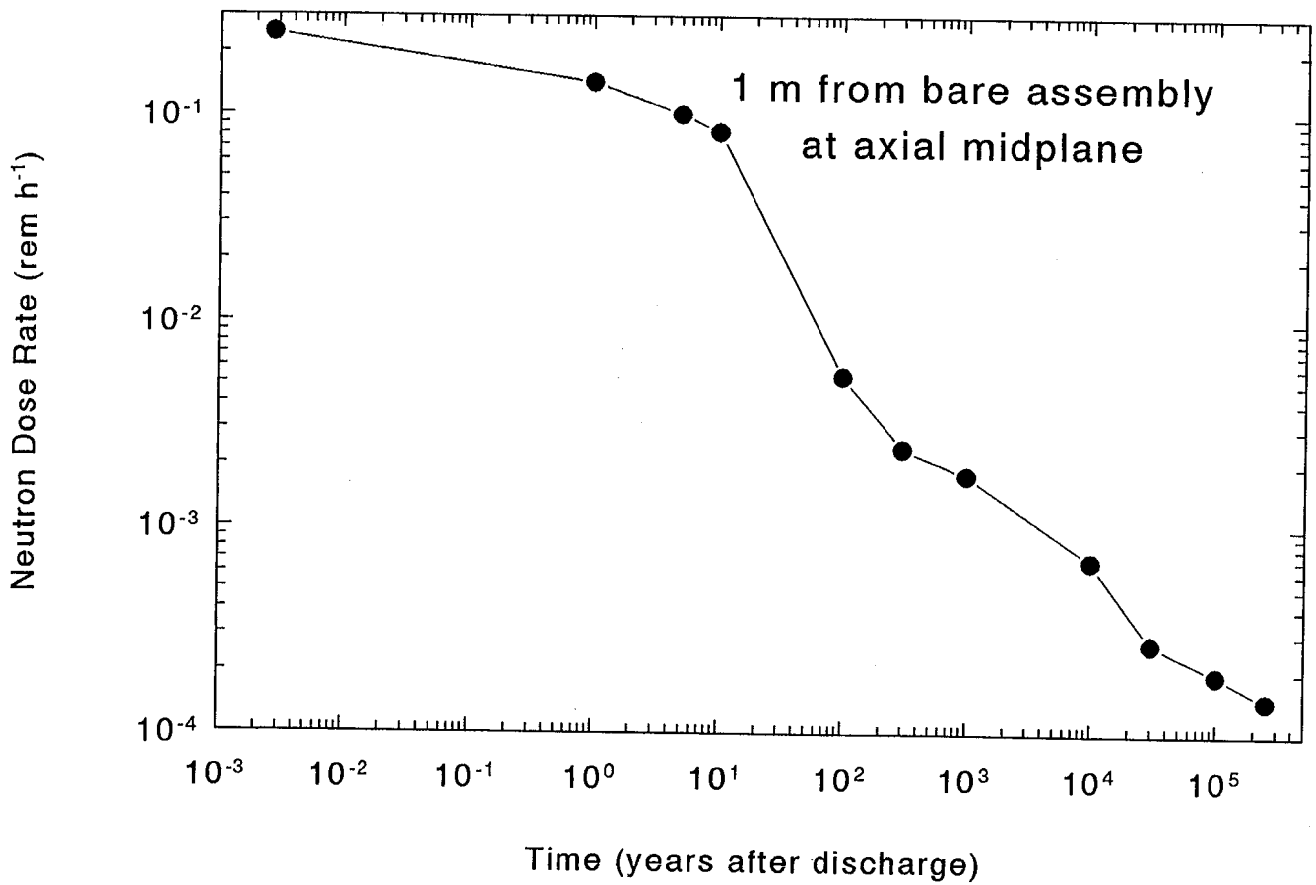


Fig. 4.6. Neutron dose rate 1 m from a BWR-5 MOX fuel assembly at the axial midplane.

Table 4.6. Gamma dose rate 1 m from a Bare BWR-5 MOX assembly at the axial midplane

Time after discharge (years if not specified)	Dose rate (rem h^{-1})
Discharge	1.55E+06
1 d	1.39E+05
1	5.28E+03
5	1.68E+03
10	9.68E+02
100	8.51E+01
300	9.27E-01
1000	6.31E-02
10000	3.81E-02
30000	3.52E-02
100000	4.04E-02
250000	3.83E-02

Table 4.7. BWR cask dose rates (rem h^{-1}) from 20 BWR-5 MOX assemblies at the axial midplane

Time after discharge (years if not specified)	Surface neutron	Surface gamma	1-m neutron	1-m gamma
1 d	1.24E-02	1.34E+01	4.79E-03	6.14E+00
1	7.16E-03	1.80E-01	2.76E-03	8.23E-02
5	5.00E-03	2.67E-02	1.93E-03	1.19E-02
10	4.15E-03	1.15E-02	1.60E-03	5.05E-03
100	2.76E-04	2.84E-04	1.06E-04	1.20E-04
300	1.24E-04	6.02E-05	4.77E-05	2.47E-05
1000	9.13E-05	4.42E-05	3.52E-05	1.81E-05
10,000	3.45E-05	1.77E-05	1.33E-05	7.30E-06
30,000	1.37E-05	9.52E-06	5.28E-06	4.04E-06
100,000	9.64E-06	1.11E-05	3.72E-06	4.86E-06
250,000	7.29E-06	1.12E-05	2.81E-06	4.96E-06

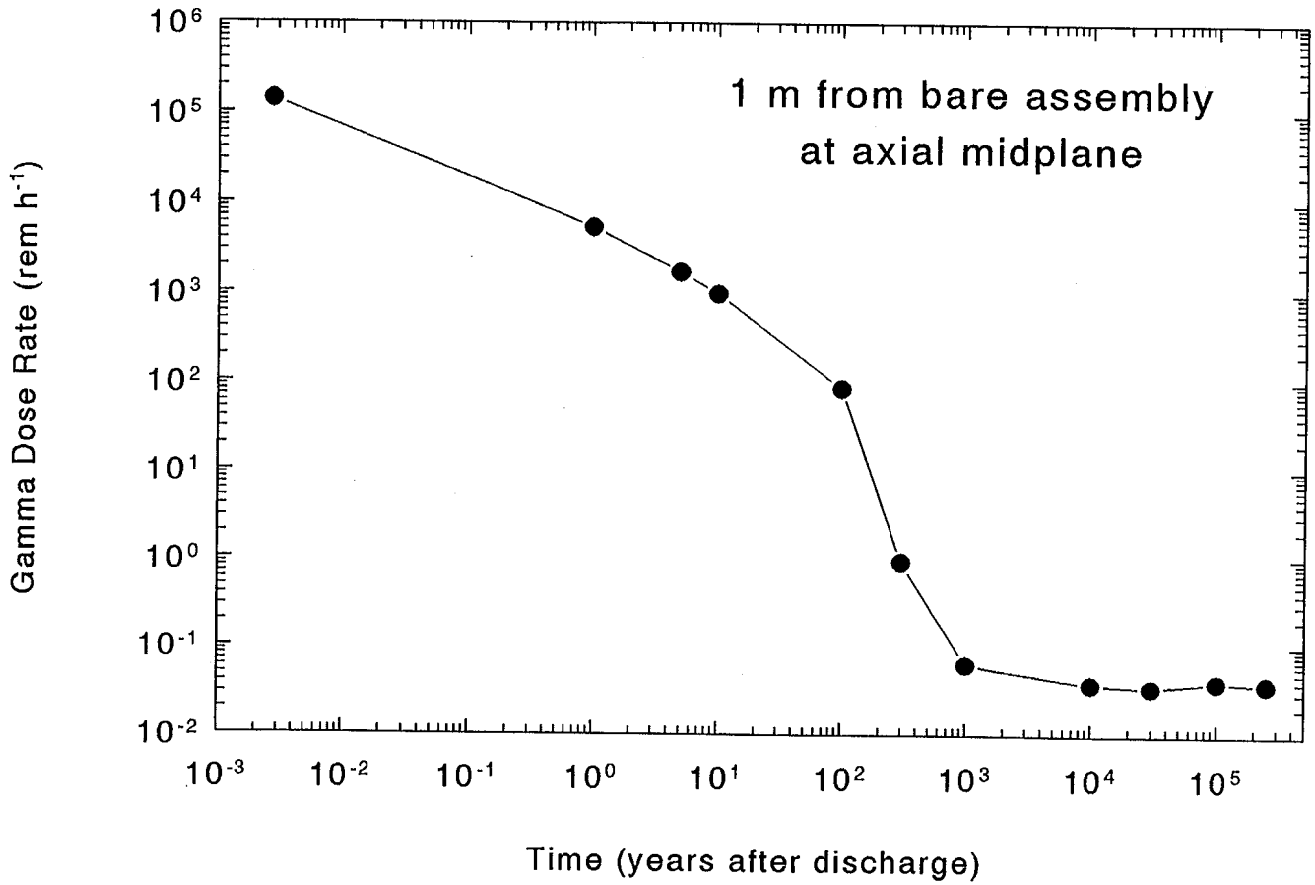


Fig. 4.7. Gamma dose rate 1 m from a bare BWR-5 MOX fuel assembly at the axial midplane.

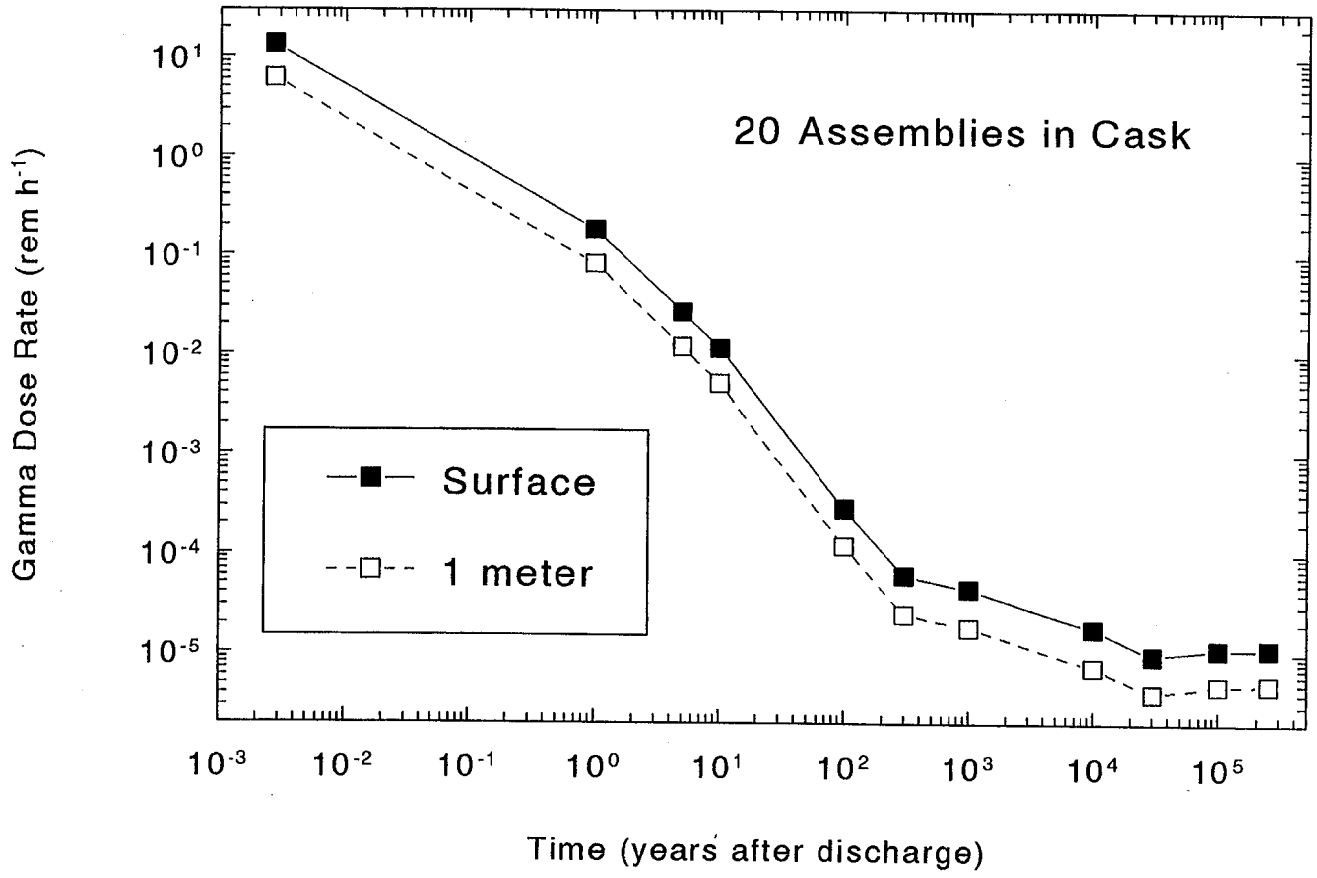


Fig. 4.8. BWR cask gamma dose rates from 20 BWR-5 MOX fuel assemblies.

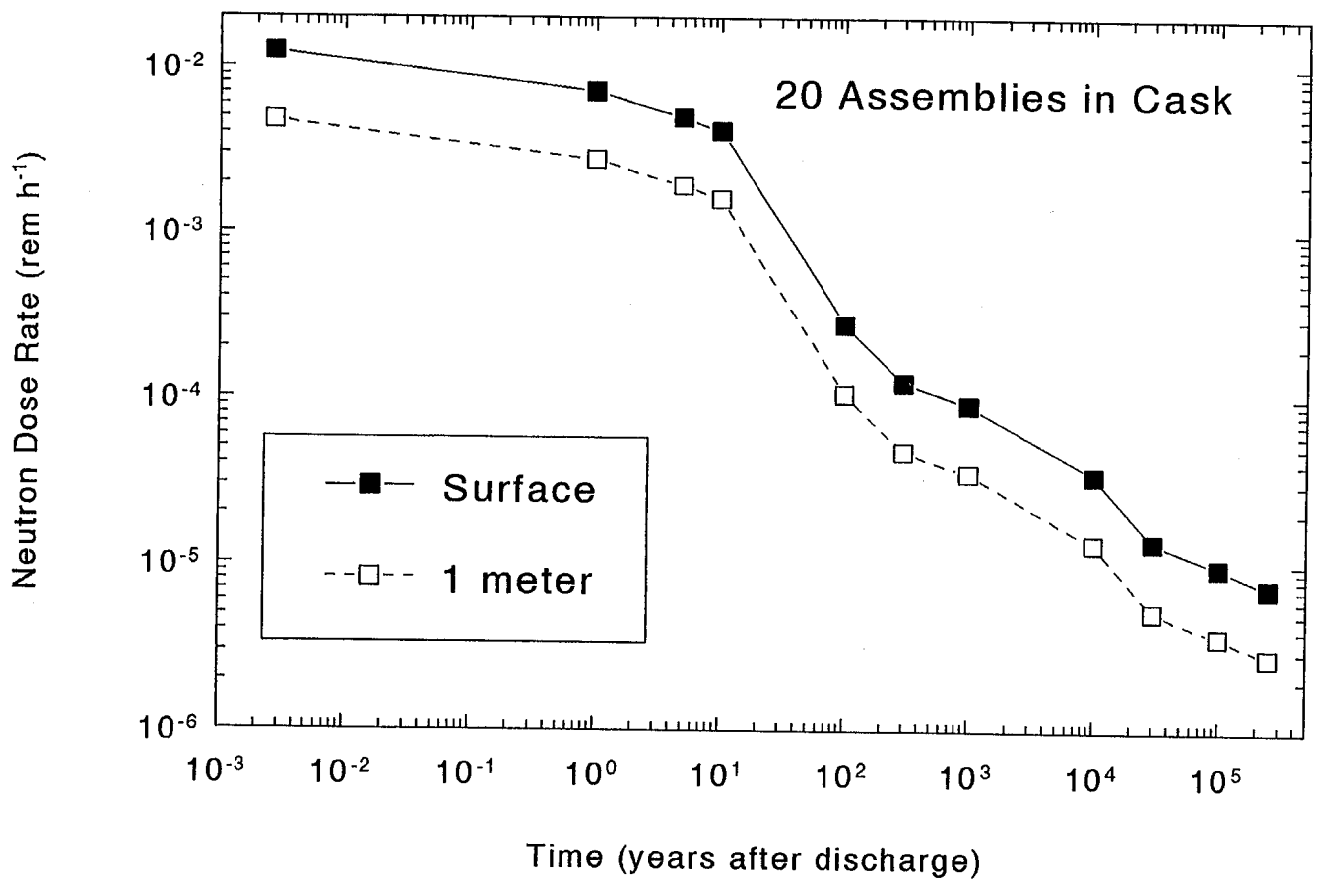


Fig. 4.9. BWR cask neutron dose rates from 20 BWR-5 MOX fuel assemblies.

Table 4.8. Nuclide inhalation hazard from a BWR-5 MOX assembly
(m^3 of air at rcg per assembly)

Time after discharge (years if not specified)	Light elements	Actinides	Fission products	Total
Discharge	2.31E+14	4.87E+16	1.03E+16	5.92E+16
1 d	8.66E+13	4.86E+16	5.09E+15	5.38E+16
1	1.26E+13	4.33E+16	8.96E+14	4.42E+16
5	1.42E+12	3.93E+16	3.26E+14	3.96E+16
10	5.51E+11	3.65E+16	2.52E+14	3.68E+16
30	5.05E+10	2.91E+16	1.46E+14	2.92E+16
100	8.17E+09	2.04E+16	2.60E+13	2.04E+16
300	1.76E+09	1.35E+16	2.11E+11	1.35E+16
1000	5.12E+07	7.65E+15	6.12E+09	7.65E+15
3000	3.86E+07	4.76E+15	6.01E+09	4.76E+15
10000	3.79E+07	2.72E+15	5.71E+09	2.72E+15
30000	3.61E+07	8.63E+14	5.00E+09	8.63E+14
100000	3.19E+07	1.14E+14	3.52E+09	1.14E+14
250000	2.73E+07	3.23E+13	2.26E+09	3.23E+13

Table 4.9. Nuclide ingestion hazard from a BWR-5 MOX assembly
(m^3 of water at rcg per assembly)

Time after discharge (years if not specified)	Light elements	Actinides	Fission products	Total
Discharge	4.09E+09	3.14E+10	3.25E+13	3.25E+13
1 d	2.98E+09	2.27E+10	3.19E+13	3.19E+13
1	3.69E+08	1.13E+09	3.87E+10	4.02E+10
5	2.26E+07	7.91E+08	2.47E+10	2.55E+10
10	6.47E+06	7.81E+08	2.10E+10	2.18E+10
30	1.19E+06	7.36E+08	1.28E+10	1.35E+10
100	4.81E+05	6.04E+08	2.28E+09	2.88E+09
300	1.11E+05	4.17E+08	1.72E+07	4.34E+08
1000	1.99E+03	1.75E+08	2.87E+05	1.75E+08
3000	1.14E+03	6.25E+07	2.84E+05	6.28E+07
10000	1.07E+03	3.47E+07	2.75E+05	3.50E+07
30000	9.16E+02	1.39E+07	2.53E+05	1.42E+07
100000	5.76E+02	9.15E+06	2.09E+05	9.36E+06
250000	2.89E+02	1.01E+07	1.75E+05	1.03E+07

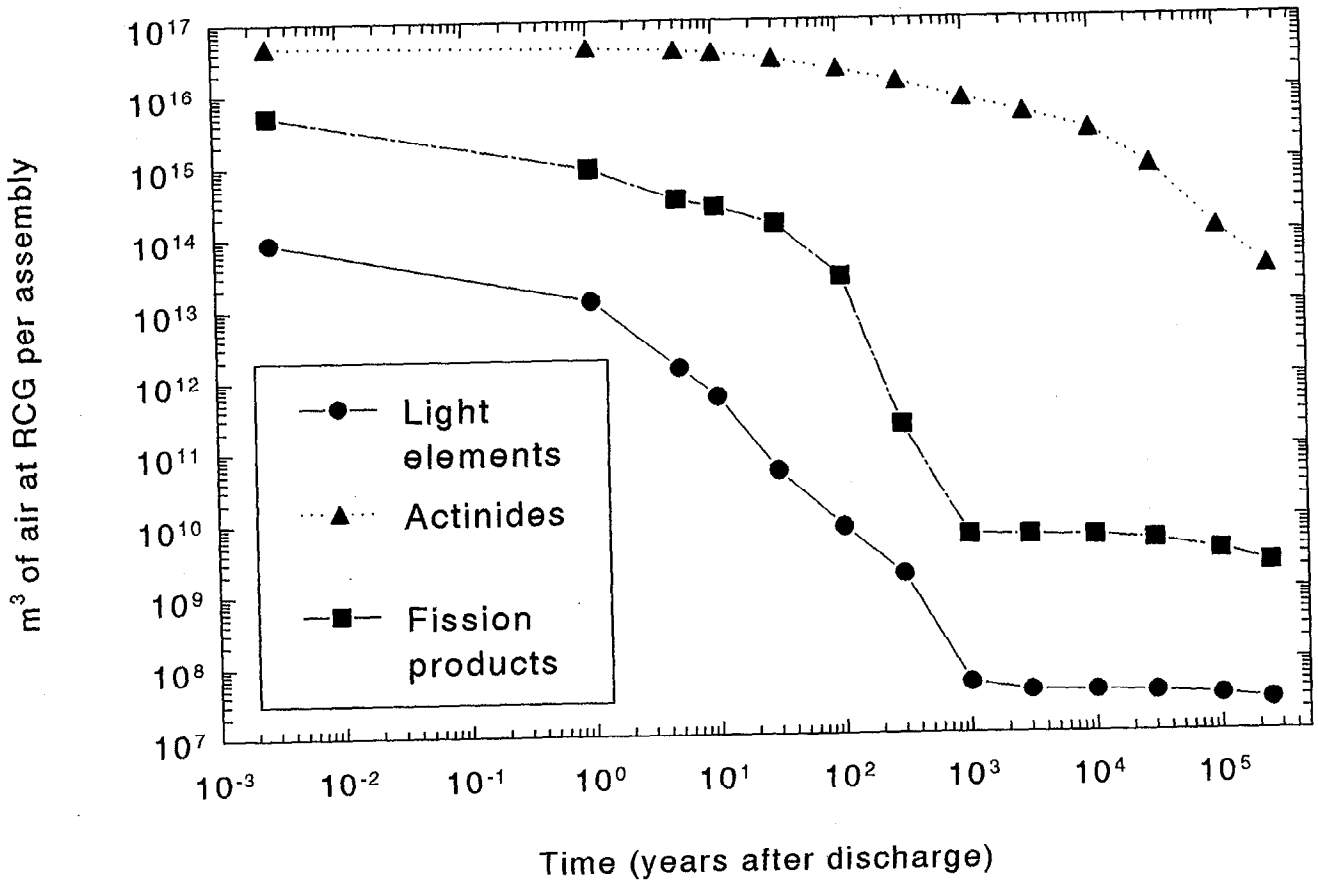


Fig. 4.10. Nuclide inhalation hazard from a BWR-5 MOX fuel assembly.

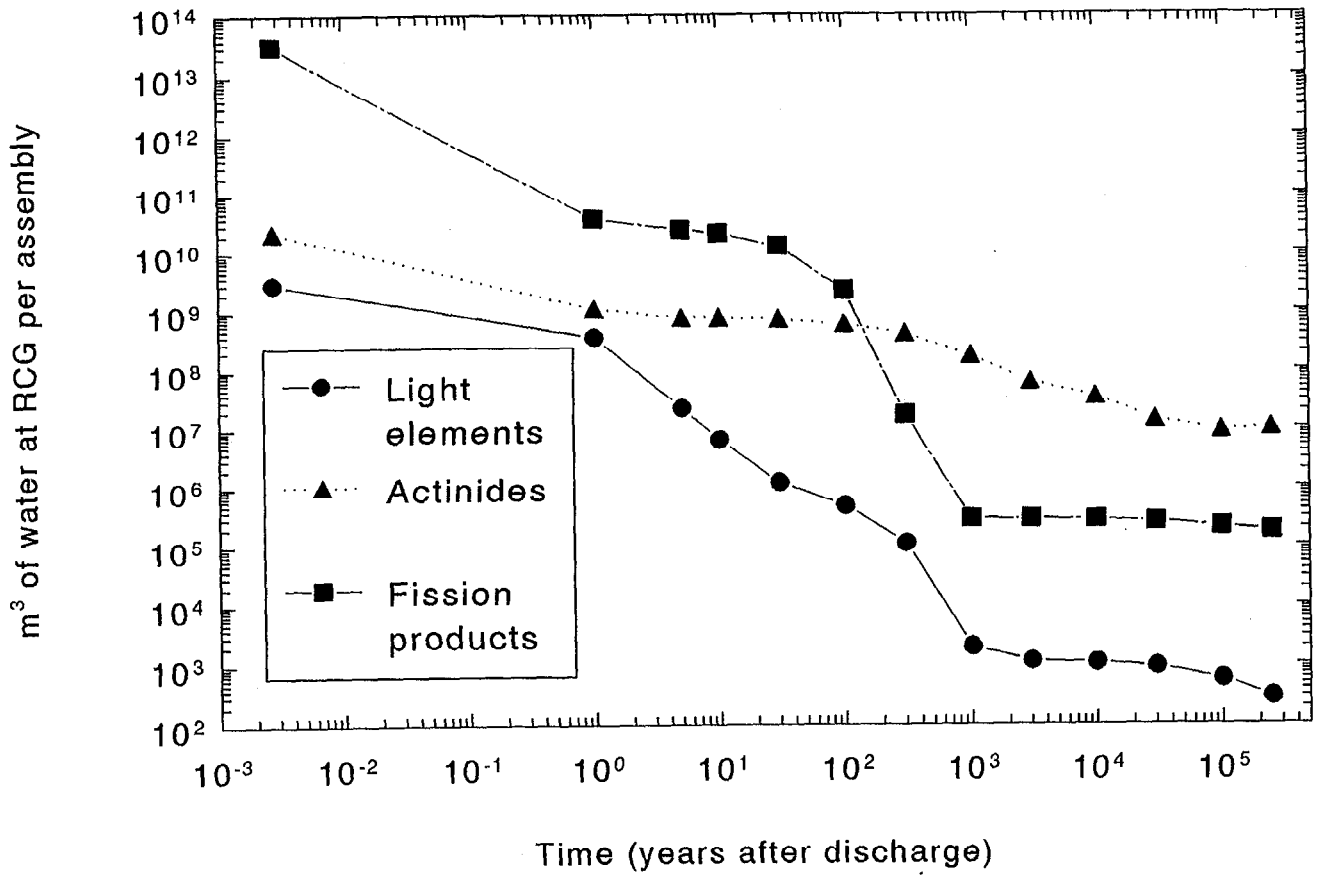


Fig. 4.11. Nuclide ingestion hazard from a BWR-5 MOX fuel assembly.

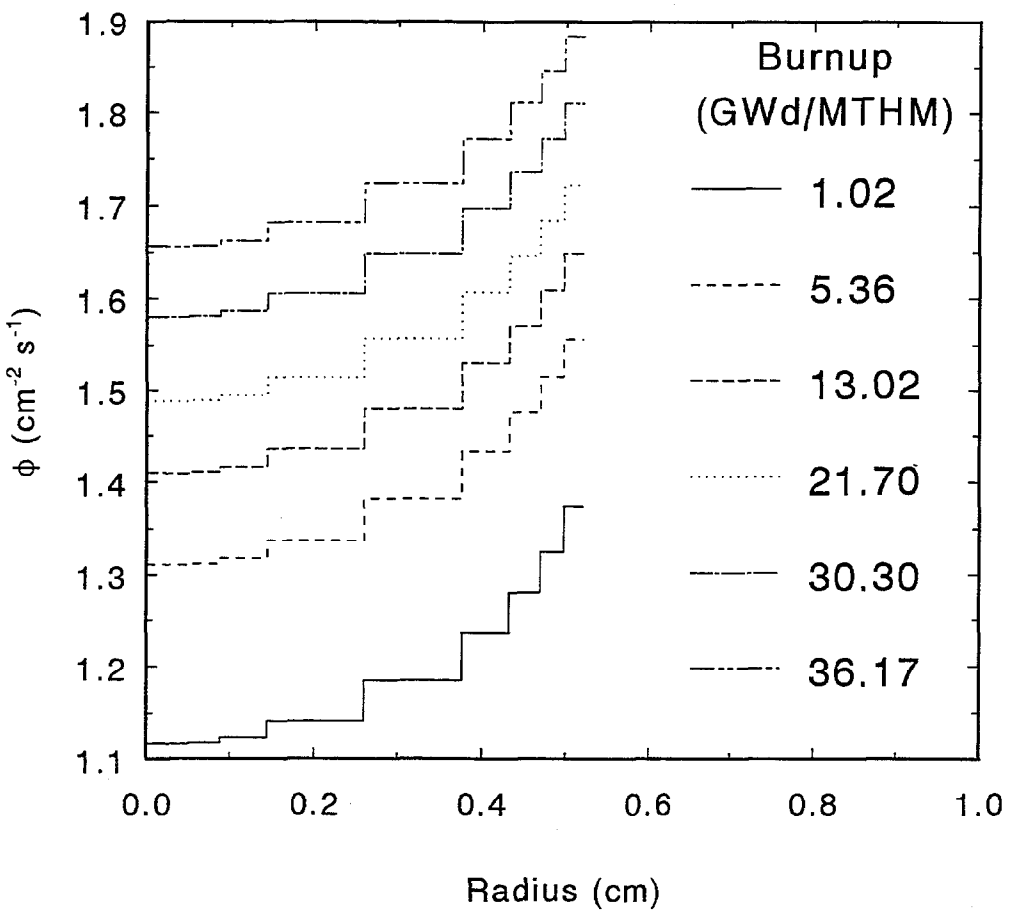


Fig. 4.12. Thermal flux in a MOX fuel pin for different assembly burnups.

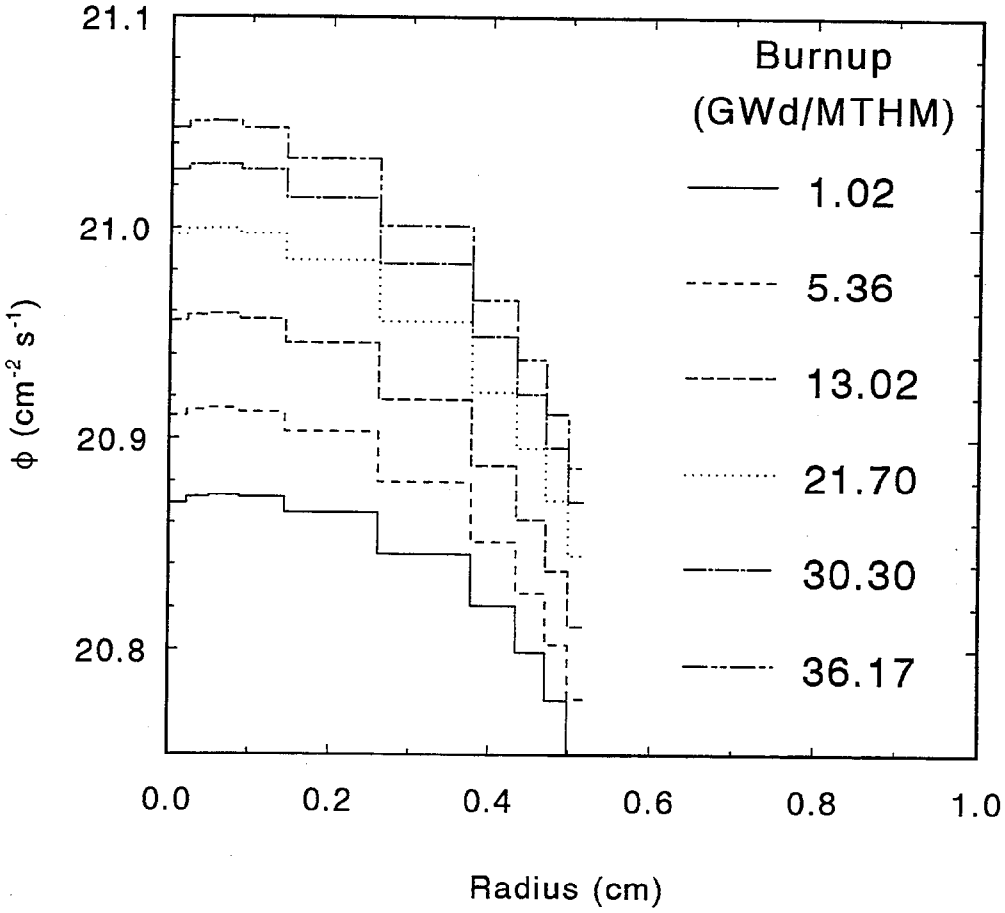


Fig. 4.13. Fast flux in a MOX fuel pin for different assembly burnups.

Table 4.10. Actinide concentrations in a BWR-5 LEU assembly (g per assembly)

Nuclide	Time after discharge						
	Discharge	1 d	1 year	5 years	10 years	30 years	100 years
^{210}Pb	2.61E-11	2.62E-11	3.12E-11	5.98E-11	1.46E-10	1.58E-09	3.55E-08
^{214}Bi	2.15E-16	2.15E-16	2.82E-16	7.10E-16	1.63E-15	1.00E-14	1.13E-13
^{210}Po	3.24E-13	3.25E-13	4.71E-13	9.20E-13	2.49E-12	2.68E-11	6.03E-10
^{225}Ra	5.59E-12	5.37E-12	7.86E-13	8.37E-13	9.17E-13	1.41E-12	5.60E-12
^{226}Ra	9.64E-09	9.65E-09	1.26E-08	3.17E-08	7.28E-08	4.47E-07	5.04E-06
^{227}Ac	9.69E-09	9.70E-09	1.36E-08	2.84E-08	4.49E-08	9.53E-08	1.87E-07
^{229}Th	1.53E-07	1.53E-07	1.56E-07	1.66E-07	1.82E-07	2.80E-07	1.11E-06
^{230}Th	2.86E-04	2.86E-04	3.66E-04	6.96E-04	1.13E-03	3.09E-03	1.21E-02
^{231}Pa	2.05E-04	2.06E-04	2.07E-04	2.12E-04	2.19E-04	2.44E-04	3.32E-04
^{233}U	4.91E-04	4.91E-04	5.27E-04	6.62E-04	8.24E-04	1.49E-03	4.17E-03
^{234}U	2.87E+01	2.87E+01	2.91E+01	3.05E+01	3.22E+01	3.83E+01	5.36E+01
^{235}U	1.31E+03	1.32E+03	1.31E+03	1.31E+03	1.31E+03	1.31E+03	1.31E+03
^{236}U	8.16E+02	8.16E+02	8.16E+02	8.16E+02	8.16E+02	8.17E+02	8.20E+02
^{238}U	1.72E+05	1.72E+05	1.72E+05	1.72E+05	1.72E+05	1.72E+05	1.72E+05
^{237}Np	9.93E+01	9.94E+01	1.01E+02	1.01E+02	1.02E+02	1.07E+02	1.34E+02
^{239}Np	1.18E+01	8.85E+00	2.31E-05	2.31E-05	2.31E-05	2.31E-05	2.29E-05
^{238}Pu	4.23E+01	4.24E+01	4.51E+01	4.45E+01	4.28E+01	3.66E+01	2.11E+01
^{239}Pu	9.81E+02	9.84E+02	9.93E+02	9.93E+02	9.93E+02	9.92E+02	9.90E+02
^{240}Pu	4.49E+02	4.49E+02	4.50E+02	4.51E+02	4.51E+02	4.53E+02	4.53E+02
^{241}Pu	2.64E+02	2.64E+02	2.52E+02	2.07E+02	1.63E+02	6.20E+01	2.11E+00
^{242}Pu	1.11E+02	1.11E+02	1.11E+02	1.11E+02	1.11E+02	1.11E+02	1.11E+02
^{241}Am	1.49E+01	1.49E+01	2.73E+01	7.12E+01	1.15E+02	2.10E+02	2.43E+02
^{242m}Am	4.14E-01	4.14E-01	4.12E-01	4.04E-01	3.94E-01	3.57E-01	2.53E-01
^{243}Am	2.68E+01	2.68E+01	2.68E+01	2.68E+01	2.68E+01	2.68E+01	2.66E+01
^{242}Cm	3.76E+00	3.76E+00	8.02E-01	2.66E-03	1.03E-03	9.31E-04	6.60E-04
^{243}Cm	1.37E-01	1.37E-01	1.33E-01	1.21E-01	1.07E-01	6.59E-02	1.20E-02
^{244}Cm	8.26E+00	8.27E+00	7.97E+00	6.83E+00	5.64E+00	2.62E+00	1.80E-01
^{245}Cm	3.49E-01	3.49E-01	3.49E-01	3.49E-01	3.48E-01	3.48E-01	3.46E-01
^{246}Cm	5.81E-02	5.81E-02	5.81E-02	5.81E-02	5.80E-02	5.78E-02	5.73E-02

Table 4.10 (continued)

Nuclide	Time after discharge (years)						
	300	1000	3000	10000	30000	100000	250000
^{210}Pb	5.70E-07	9.13E-06	6.98E-05	4.07E-04	1.41E-03	3.19E-03	3.71E-03
^{214}Bi	1.24E-12	1.58E-11	1.21E-10	7.03E-10	2.45E-09	5.52E-09	6.42E-09
^{210}Po	9.69E-09	1.55E-07	1.19E-06	6.91E-06	2.40E-05	5.43E-05	6.31E-05
^{225}Ra	4.50E-11	6.75E-10	8.07E-09	8.93E-08	5.30E-07	2.11E-06	4.33E-06
^{226}Ra	5.55E-05	7.06E-04	5.39E-03	3.14E-02	1.09E-01	2.47E-01	2.87E-01
^{227}Ac	3.56E-07	9.59E-07	2.61E-06	8.32E-06	2.34E-05	5.53E-05	6.83E-05
^{229}Th	8.90E-06	1.34E-04	1.60E-03	1.77E-02	1.05E-01	4.18E-01	8.57E-01
^{230}Th	4.75E-02	1.90E-01	5.94E-01	1.94E+00	5.20E+00	1.18E+01	1.37E+01
^{231}Pa	5.85E-04	1.47E-03	3.99E-03	1.27E-02	3.59E-02	8.47E-02	1.05E-01
^{233}U	1.49E-02	7.49E-02	3.01E-01	1.11E+00	3.30E+00	9.49E+00	1.74E+01
^{234}U	7.01E+01	7.44E+01	7.40E+01	7.28E+01	6.93E+01	5.85E+01	4.15E+01
^{235}U	1.32E+03	1.34E+03	1.39E+03	1.55E+03	1.88E+03	2.25E+03	2.31E+03
^{236}U	8.30E+02	8.61E+02	9.38E+02	1.11E+03	1.25E+03	1.26E+03	1.26E+03
^{238}U	1.72E+05	1.72E+05	1.72E+04	1.72E+05	1.72E+05	1.72E+05	1.72E+05
^{237}Np	2.00E+02	3.18E+02	3.73E+02	3.74E+02	3.72E+02	3.64E+02	3.46E+02
^{239}Np	2.24E-05	2.10E-05	1.74E-05	9.02E-06	1.37E-06	1.90E-09	4.22E-13
^{238}Pu	4.40E+00	2.10E-02	2.21E-07	2.49E-22	0.00E+00	0.00E+00	0.00E+00
^{239}Pu	9.85E+02	9.67E+02	9.17E+02	7.58E+02	4.33E+02	5.81E+01	7.79E-01
^{240}Pu	4.43E+02	4.12E+02	3.33E+02	1.59E+02	1.92E+01	1.18E-02	4.80E-09
^{241}Pu	7.00E-04	5.35E-04	4.54E-04	2.57E-04	5.02E-05	1.66E-07	8.08E-13
^{242}Pu	1.11E+02	1.11E+02	1.11E+02	1.09E+02	1.05E+02	9.23E+01	6.99E+01
^{241}Am	1.78E+02	5.80E+01	2.37E+00	7.77E-03	1.51E-03	5.02E-06	2.47E-11
^{242m}Am	9.48E-02	3.03E-03	1.63E-07	1.85E-22	0.00E+00	0.00E+00	0.00E+00
^{243}Am	2.61E+01	2.44E+01	2.02E+01	1.05E+01	1.60E+00	2.21E-03	4.90E-07
^{242}Cm	2.47E-04	7.91E-06	4.26E-10	4.84E-25	0.00E+00	0.00E+00	0.00E+00
^{243}Cm	9.27E-05	3.74E-12	2.79E-33	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{244}Cm	8.47E-05	1.93E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{245}Cm	3.40E-01	3.21E-01	2.73E-01	1.54E-01	3.02E-02	1.00E-04	4.86E-10
^{246}Cm	5.56E-02	5.02E-02	3.74E-02	1.34E-02	7.16E-04	2.52E-08	1.12E-17

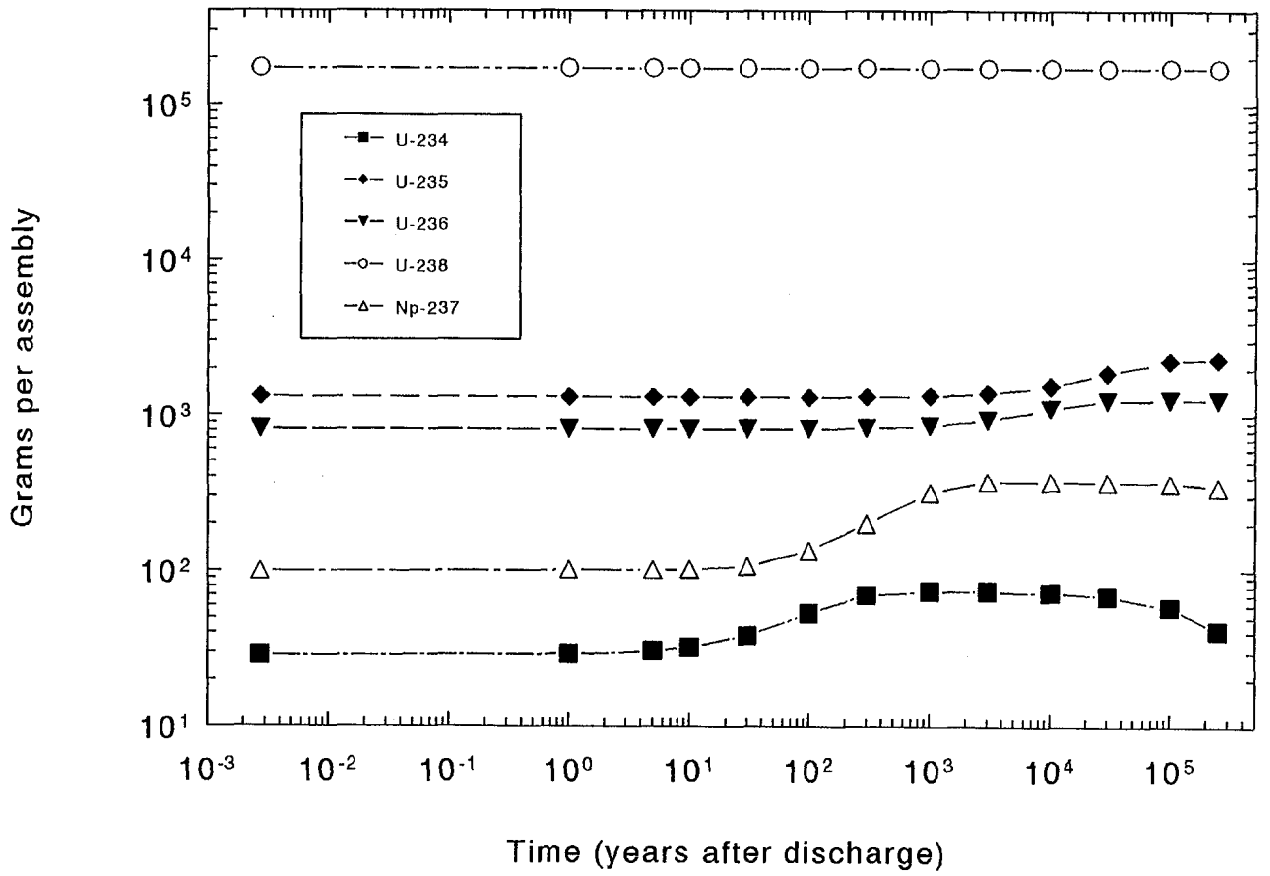


Fig. 4.14. Uranium and neptunium decay from a BWR-5 LEU fuel assembly.

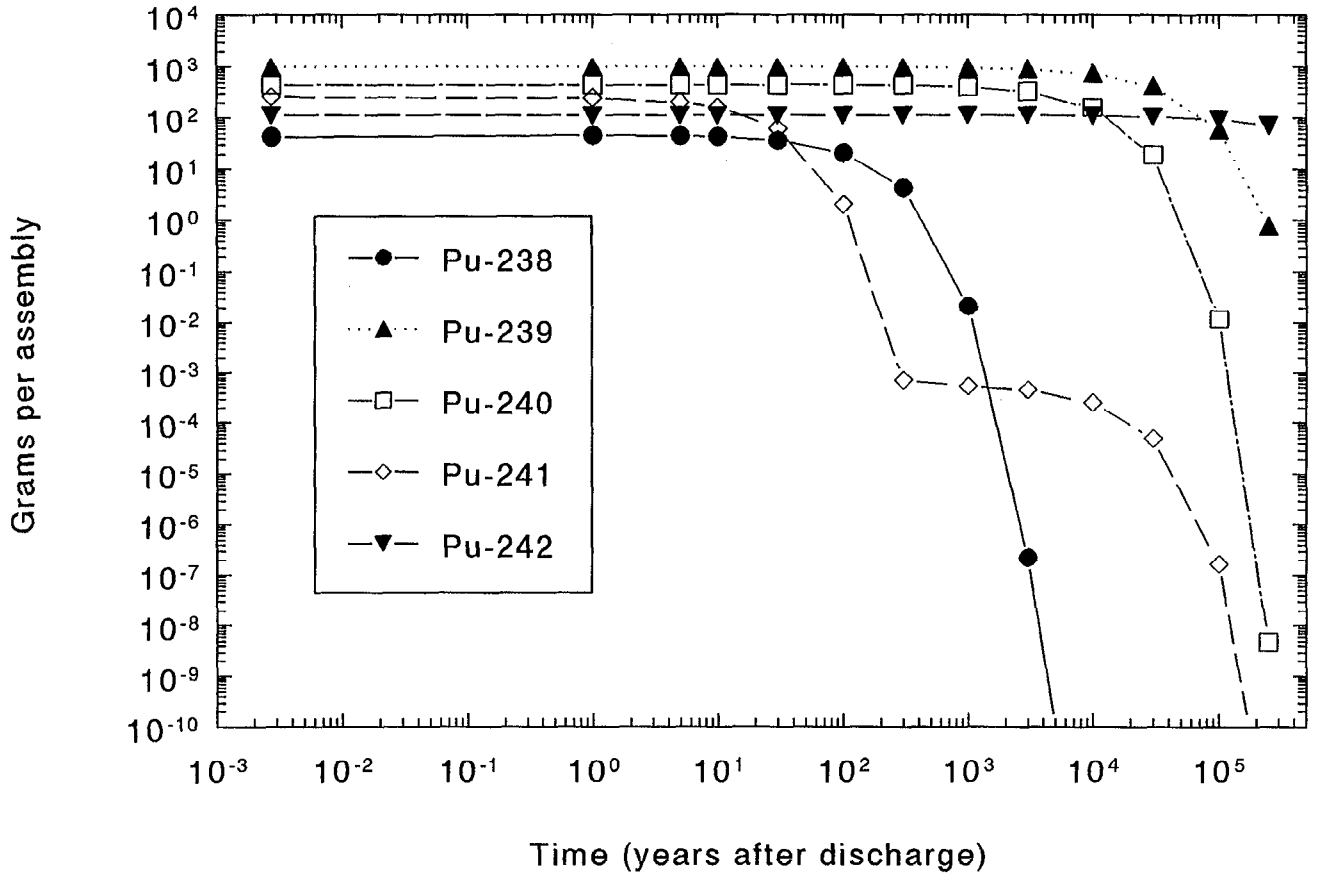


Fig. 4.15. Plutonium decay from a BWR-5 LEU fuel assembly.

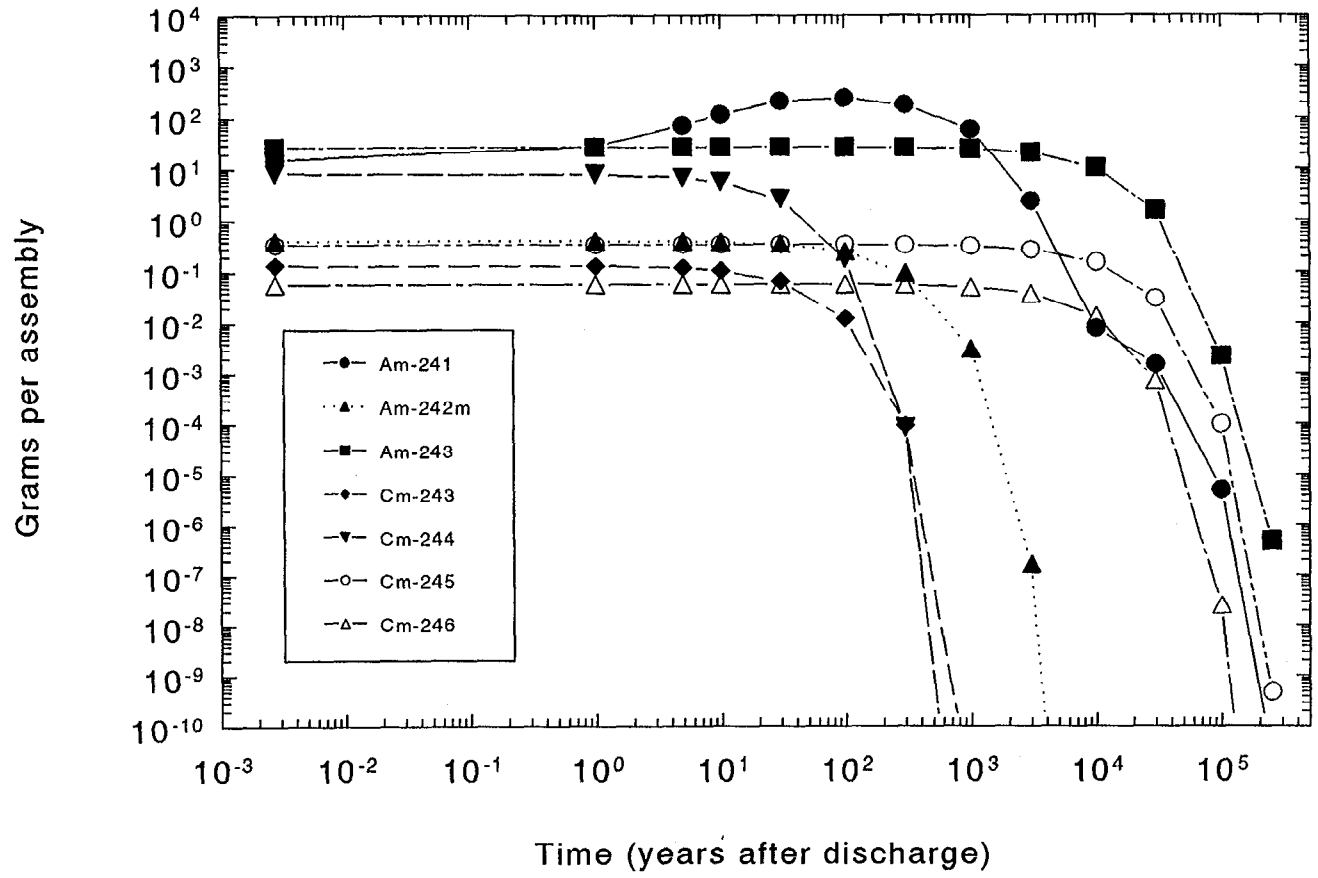


Fig. 4.16. Americium and curium decay from a BWR-5 LEU fuel assembly.

Table 4.11. Light-element and fission-product concentrations in a BWR-5 LEU assembly (g per assembly)

Nuclide	Time after discharge						
	Discharge	1 d	1 year	5 years	10 years	30 years	100 years
^{3}H	9.75E-03	9.74E-03	9.21E-03	7.36E-03	5.55E-03	1.80E-03	3.52E-05
^{14}C	9.13E-03	9.13E-03	9.13E-03	9.13E-03	9.13E-03	9.10E-03	9.02E-03
^{60}Co	4.85E-01	4.85E-01	4.25E-01	2.51E-01	1.30E-01	9.38E-03	9.40E-07
^{59}Ni	2.82E+00	2.82E+00	2.82E+00	2.82E+00	2.82E+00	2.82E+00	2.82E+00
^{63}Ni	5.58E-01	5.58E-01	5.55E-01	5.39E-01	5.21E-01	4.54E-01	2.79E-01
^{85}Kr	4.20E+00	4.20E+00	3.93E+00	3.04E+00	2.20E+00	6.03E-01	6.52E-03
^{90}Sr	1.05E+02	1.05E+02	1.02E+02	9.28E+01	8.20E+01	5.01E+01	8.94E+00
^{90}Y	2.83E-02	2.80E-02	2.66E-02	2.41E-02	2.13E-02	1.30E-02	2.32E-03
^{94}Nb	1.01E-04	1.01E-04	1.01E-04	1.01E-04	1.01E-04	1.01E-04	1.01E-04
^{95}Mo	1.52E+02	1.52E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02
^{99}Tc	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02
^{101}Ru	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02
^{106}Ru	2.27E+01	2.26E+01	1.15E+01	7.52E-01	2.50E-02	3.02E-08	5.87E-29
^{103}Rh	8.81E+01	8.82E+01	9.38E+01	9.39E+01	9.39E+01	9.39E+01	9.39E+01
^{106}Rh	2.24E-05	2.10E-05	1.07E-05	6.98E-07	2.32E-08	2.80E-14	5.44E-35
^{105}Pd	8.64E+01	8.65E+01	8.66E+01	8.66E+01	8.66E+01	8.66E+01	8.66E+01
^{109}Ag	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01
^{110m}Ag	1.33E-01	1.32E-01	4.82E-02	8.36E-04	5.26E-06	8.25E-15	0.00E+00
^{131}Xe	8.74E+01	8.75E+01	8.85E+01	8.85E+01	8.85E+01	8.85E+01	8.85E+01
^{133}Cs	2.31E+02	2.31E+02	2.32E+02	2.32E+02	2.32E+02	2.32E+02	2.32E+02
^{134}Cs	2.12E+01	2.11E+01	1.15E+01	3.94E+00	7.34E-01	8.82E-04	5.31E-14
^{137}Cs	2.50E+02	2.50E+02	2.44E+02	2.23E+02	1.98E+02	1.25E+02	2.48E+01
^{137m}Ba	3.83E-05	3.81E-05	3.73E-05	3.40E-05	3.03E-05	1.91E-05	3.78E-06
^{144}Ce	3.89E+01	3.88E+01	1.60E+01	4.57E-01	5.38E-03	1.03E-10	0.00E+00
^{141}Pr	2.23E+02	2.23E+02	2.30E+02	2.30E+02	2.30E+02	2.30E+02	2.30E+02
^{144}Pr	1.65E-03	1.63E-03	6.73E-04	1.93E-05	2.27E-07	4.34E-15	0.00E+00
^{143}Nd	1.57E+02	1.57E+02	1.60E+02	1.60E+02	1.60E+02	1.60E+02	1.60E+02
^{145}Nd	1.37E+02	1.37E+02	1.37E+02	1.37E+02	1.37E+02	1.37E+02	1.37E+02
^{147}Pm	2.71E+01	2.71E+01	2.16E+01	7.51E+00	2.00E+00	1.02E-02	9.41E-11
^{147}Sm	2.23E+01	2.23E+01	2.88E+01	4.29E+01	4.84E+01	5.04E+01	5.04E+01
^{149}Sm	3.92E-01	4.43E-01	5.81E-01	5.81E-01	5.81E-01	5.81E-01	5.81E-01
^{150}Sm	6.10E+01	6.10E+01	6.10E+01	6.10E+01	6.10E+01	6.10E+01	6.10E+01
^{151}Sm	2.83E+00	2.85E+00	2.85E+00	2.76E+00	2.65E+00	2.28E+00	1.33E+00
^{152}Sm	2.64E+01	2.64E+01	2.64E+01	2.64E+01	2.64E+01	2.64E+01	2.64E+01
^{151}Eu	4.98E-03	5.04E-03	2.70E-02	1.13E-01	2.18E-01	5.97E-01	1.54E+00
^{153}Eu	2.28E+01	2.28E+01	2.29E+01	2.29E+01	2.29E+01	2.29E+01	2.29E+01
^{154}Eu	8.71E+00	8.71E+00	8.03E+00	5.82E+00	3.89E+00	7.74E-01	2.73E-03
^{155}Gd	3.20E-02	3.33E-02	4.73E-01	1.71E+00	2.50E+00	3.19E+00	3.23E+00
^{157}Gd	1.79E-02	1.96E-02	2.05E-02	2.05E-02	2.05E-02	2.05E-02	2.05E-02

Table 4.11 (continued)

Nuclide	Time after discharge (years)						
	300	1000	3000	10000	30000	100000	250000
^{3}H	4.61E-10	3.73E-27	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{14}C	8.81E-03	8.08E-03	6.34E-03	2.72E-03	2.42E-04	5.07E-08	6.66E-16
^{60}Co	3.53E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{59}Ni	2.81E+00	2.80E+00	2.74E+00	2.57E+00	2.14E+00	1.12E+00	2.80E-01
^{63}Ni	6.99E-02	5.49E-04	5.30E-10	4.70E-31	0.00E+00	0.00E+00	0.00E+00
^{85}Kr	1.58E-08	3.46E-28	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{90}Sr	6.49E-02	2.12E-09	8.62E-31	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{90}Y	1.69E-05	5.50E-13	2.24E-34	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{94}Nb	1.00E-04	9.77E-05	9.12E-05	7.18E-05	3.63E-05	3.32E-06	1.98E-08
^{95}Mo	1.64E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02
^{99}Tc	1.59E+02	1.59E+02	1.58E+02	1.54E+02	1.44E+02	1.15E+02	7.00E+01
^{101}Ru	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02
^{106}Ru	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{103}Rh	9.39E+01	9.39E+01	9.39E+01	9.39E+01	9.39E+01	9.39E+01	9.39E+01
^{106}Rh	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{105}Pd	8.66E+01	8.66E+01	8.66E+01	8.66E+01	8.66E+01	8.66E+01	8.66E+01
^{109}Ag	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01
^{110m}Ag	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{131}Xe	8.85E+01	8.85E+01	8.85E+01	8.85E+01	8.85E+01	8.85E+01	8.85E+01
^{133}Cs	2.32E+02	2.32E+02	2.32E+02	2.32E+02	2.32E+02	2.32E+02	2.32E+02
^{134}Cs	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{137}Cs	2.44E-01	2.30E-08	1.96E-28	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{137m}Ba	3.72E-08	3.52E-15	2.99E-35	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{144}Ce	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{141}Pr	2.30E+02	2.30E+02	2.30E+02	2.30E+02	2.30E+02	2.30E+02	2.30E+02
^{144}Pr	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{143}Nd	1.60E+02	1.60E+02	1.60E+02	1.60E+02	1.60E+02	1.60E+02	1.60E+02
^{145}Nd	1.37E+02	1.37E+02	1.37E+02	1.37E+02	1.37E+02	1.37E+02	1.37E+02
^{147}Pm	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{147}Sm	5.04E+01	5.04E+01	5.04E+01	5.04E+01	5.04E+01	5.04E+01	5.04E+01
^{149}Sm	5.81E-01	5.81E-01	5.81E-01	5.81E-01	5.81E-01	5.81E-01	5.81E-01
^{150}Sm	6.10E+01	6.10E+01	6.10E+01	6.10E+01	6.10E+01	6.10E+01	6.10E+01
^{151}Sm	2.84E-01	1.29E-03	2.64E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{152}Sm	2.64E+01	2.64E+01	2.64E+01	2.64E+01	2.64E+01	2.64E+01	2.64E+01
^{151}Eu	2.59E+00	2.87E+00	2.87E+00	2.87E+00	2.87E+00	2.87E+00	2.87E+00
^{153}Eu	2.29E+01	2.29E+01	2.29E+01	2.29E+01	2.29E+01	2.29E+01	2.29E+01
^{154}Eu	2.67E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{155}Gd	3.23E+00	3.23E+00	3.23E+00	3.23E+00	3.23E+00	3.23E+00	3.23E+00
^{157}Gd	2.05E-02	2.05E-02	2.05E-02	2.05E-02	2.05E-02	2.05E-02	2.05E-02

Table 4.12. Radioactivity from a BWR-5 LEU assembly (Ci per assembly)

Time after discharge (years if not specified)	Light elements	Actinides	Fission products	Total
Discharge	1.18E+05	5.81E+06	2.16E+07	2.75E+07
1 d	6.43E+04	2.26E+06	4.28E+06	6.60E+06
1	3.21E+03	3.04E+04	3.08E+05	3.42E+05
5	6.28E+02	2.32E+04	8.80E+04	1.12E+05
10	2.57E+02	1.86E+04	6.23E+04	8.12E+04
30	3.77E+01	8.16E+03	3.58E+04	4.40E+04
100	1.64E+01	1.61E+03	6.76E+03	8.39E+03
300	4.38E+00	8.63E+02	7.08E+01	9.38E+02
1000	4.25E-01	3.64E+02	3.76E+00	3.68E+02
3000	3.81E-01	1.51E+02	3.70E+00	1.55E+02
10000	3.51E-01	8.95E+01	3.61E+00	9.35E+01
30000	3.03E-01	3.48E+01	3.36E+00	3.85E+01
100000	2.16E-01	8.35E+00	2.70E+00	1.13E+01
250000	1.40E-01	5.74E+00	1.81E+00	7.69E+00

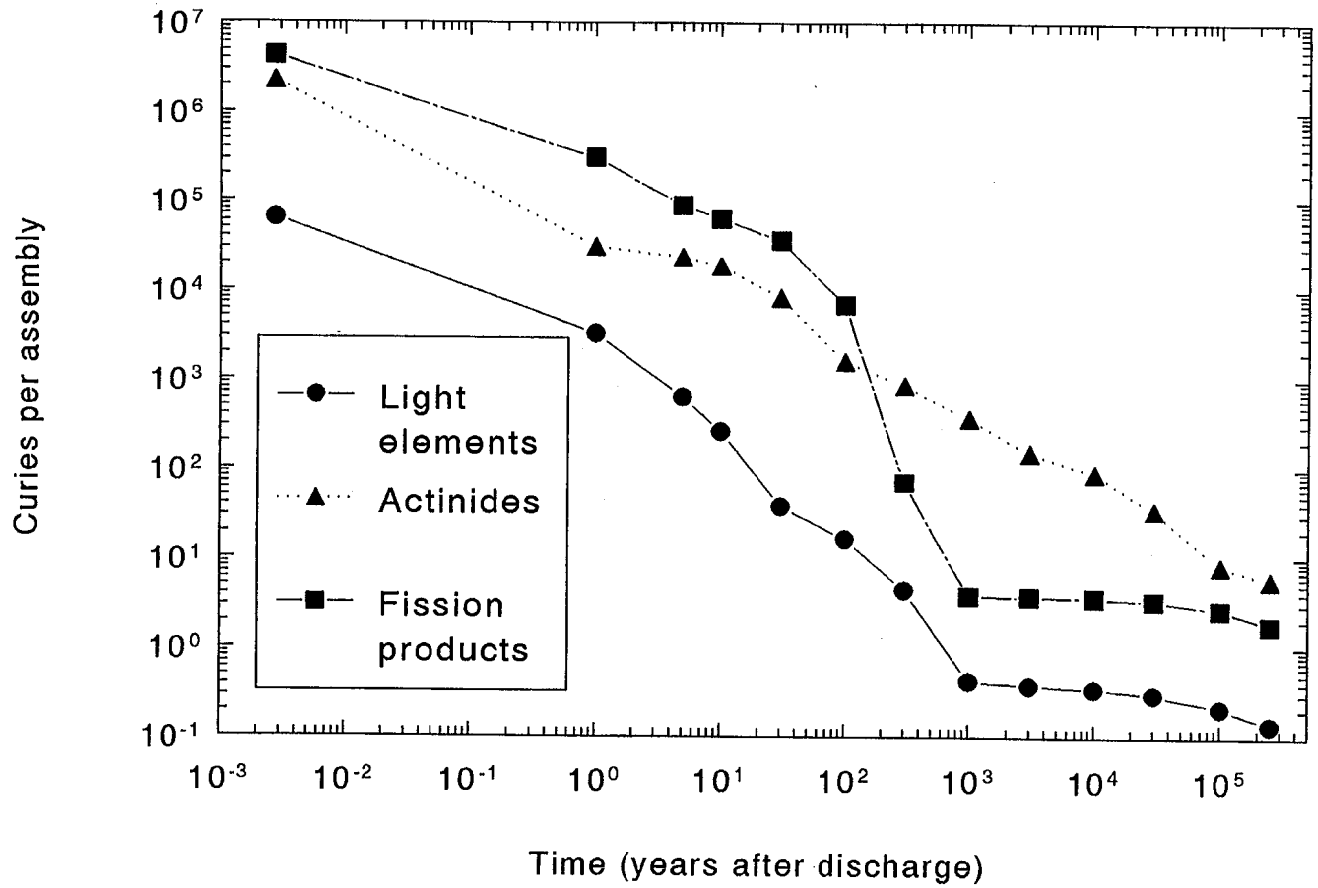


Fig. 4.17. Activity from a BWR-5 LEU fuel assembly.

Table 4.13. Discharge activities for BWR-5 MOX and LEU Assemblies

Nuclide	Activity in MOX (Ci per assembly)	Activity in LEU (Ci per assembly)	MOX/LEU
^{3}H	1.01E+02	9.41E+01	1.07E+00
^{14}C	3.17E-02	4.07E-02	7.79E-01
^{58}Co	3.94E+02	3.83E+02	1.03E+00
^{60}Co	5.51E+02	5.49E+02	1.00E+00
^{85}Kr	9.77E+02	1.65E+03	5.92E-01
^{85m}Kr	1.86E+04	2.74E+04	6.79E-01
^{87}Kr	3.63E+04	5.44E+04	6.67E-01
^{88}Kr	4.85E+04	7.49E+04	6.48E-01
^{86}Rb	1.39E+02	2.42E+02	5.74E-01
^{89}Sr	5.26E+04	8.51E+04	6.18E-01
^{90}Sr	7.55E+03	1.48E+04	5.10E-01
^{91}Sr	9.13E+04	1.33E+05	6.86E-01
^{92}Sr	1.08E+05	1.45E+05	7.45E-01
^{90}Y	8.53E+03	1.61E+04	5.30E-01
^{91}Y	7.40E+04	1.09E+05	6.79E-01
^{92}Y	1.08E+05	1.46E+05	7.40E-01
^{93}Y	9.07E+04	1.14E+05	7.96E-01
^{95}Zr	1.47E+05	1.70E+05	8.65E-01
^{97}Zr	2.04E+05	2.20E+05	9.27E-01
^{95}Nb	1.22E+05	1.41E+05	8.65E-01
^{99}Mo	2.22E+05	2.31E+05	9.61E-01
^{99m}Tc	1.95E+05	2.04E+05	9.56E-01

Table 4.13 (continued)

Nuclide	Activity in MOX (Ci per assembly)	Activity in LEU (Ci per assembly)	MOX/LEU
^{103}Ru	2.11E+05	1.87E+05	1.13E+00
^{105}Ru	1.93E+05	1.53E+05	1.26E+00
^{106}Ru	1.15E+05	7.52E+04	1.53E+00
^{105}Rh	1.86E+05	1.46E+05	1.27E+00
^{127}Sb	1.35E+04	1.18E+04	1.14E+00
^{129}Sb	4.33E+04	4.16E+04	1.04E+00
^{127}Te	1.28E+04	1.11E+04	1.15E+00
^{127m}Te	1.74E+03	1.46E+03	1.19E+00
^{129}Te	4.08E+04	3.90E+04	1.05E+00
^{129m}Te	7.74E+03	7.32E+03	1.06E+00
^{132}Te	1.76E+05	1.79E+05	9.83E-01
^{131}I	1.24E+05	1.25E+05	9.92E-01
^{132}I	1.80E+05	1.83E+05	9.84E-01
^{133}I	2.45E+05	2.55E+05	9.61E-01
^{134}I	2.63E+05	2.79E+05	9.43E-01
^{135}I	2.37E+05	2.43E+05	9.75E-01
^{131m}Xe	1.40E+03	1.40E+03	1.00E+00
^{133m}Xe	7.99E+03	8.14E+03	9.82E-01
^{133}Xe	2.36E+05	2.45E+05	9.63E-01
^{135}Xe	8.25E+04	7.61E+04	1.08E+00
^{134}Cs	2.65E+04	2.74E+04	9.67E-01
^{136}Cs	1.10E+04	9.06E+03	1.21E+00

Table 4.13 (continued)

Nuclide	Activity in MOX (Ci per assembly)	Activity in LEU (Ci per assembly)	MOX/LEU
^{137}Cs	2.14E+04	2.17E+04	9.86E-01
^{139}Ba	2.05E+05	2.21E+05	9.28E-01
^{140}Ba	2.08E+05	2.21E+05	9.41E-01
^{140}La	2.21E+05	2.36E+05	9.36E-01
^{141}La	1.86E+05	2.01E+05	9.25E-01
^{142}La	1.78E+05	1.96E+05	9.08E-01
^{141}Ce	1.72E+05	1.86E+05	9.25E-01
^{143}Ce	1.63E+05	1.85E+05	8.81E-01
^{144}Ce	1.04E+05	1.24E+05	8.39E-01
^{143}Pr	1.57E+05	1.78E+05	8.82E-01
^{147}Nd	7.85E+04	8.26E+04	9.50E-01
^{237}U	2.97E+04	1.18E+05	2.52E-01
^{239}U	2.61E+06	2.74E+06	9.53E-01
^{237}Np	2.17E-02	7.00E-02	3.10E-01
^{238}Np	1.75E+04	6.03E+04	2.90E-01
^{239}Np	2.60E+06	2.74E+06	9.49E-01
^{238}Pu	6.28E+02	7.25E+02	8.66E-01
^{239}Pu	8.72E+01	6.09E+01	1.43E+00
^{240}Pu	2.63E+02	1.02E+02	2.58E+00
^{241}Pu	5.76E+04	2.73E+04	2.11E+00
^{241}Am	1.44E+02	5.11E+01	2.82E+00
^{242}Cm	3.22E+04	1.25E+04	2.58E+00
^{244}Cm	1.63E+03	6.69E+02	2.44E+00

Table 4.14. Decay heat from a BWR-5 LEU assembly (W per assembly)

Time after discharge (years if not specified)	Light elements	Actinides	Fission products	Total
Discharge	6.06E+02	1.57E+04	2.67E+05	2.83E+05
1 d	2.93E+02	6.17E+03	1.94E+04	2.59E+04
1	1.33E+01	1.55E+02	1.27E+03	1.44E+03
5	4.77E+00	5.93E+01	2.89E+02	3.53E+02
10	2.38E+00	5.96E+01	1.84E+02	2.46E+02
30	1.67E-01	5.81E+01	1.02E+02	1.60E+02
100	1.75E-03	4.58E+01	1.89E+01	6.47E+01
300	4.54E-04	2.82E+01	1.68E-01	2.84E+01
1000	4.53E-05	1.16E+01	3.52E-03	1.16E+01
3000	3.76E-05	4.58E+00	3.48E-03	4.58E+00
10000	3.12E-05	2.71E+00	3.35E-03	2.71E+00
30000	2.63E-05	1.05E+00	3.01E-03	1.05E+00
100000	2.18E-05	2.21E-01	2.10E-03	2.23E-01
250000	1.77E-05	1.33E-01	1.06E-03	1.34E-01

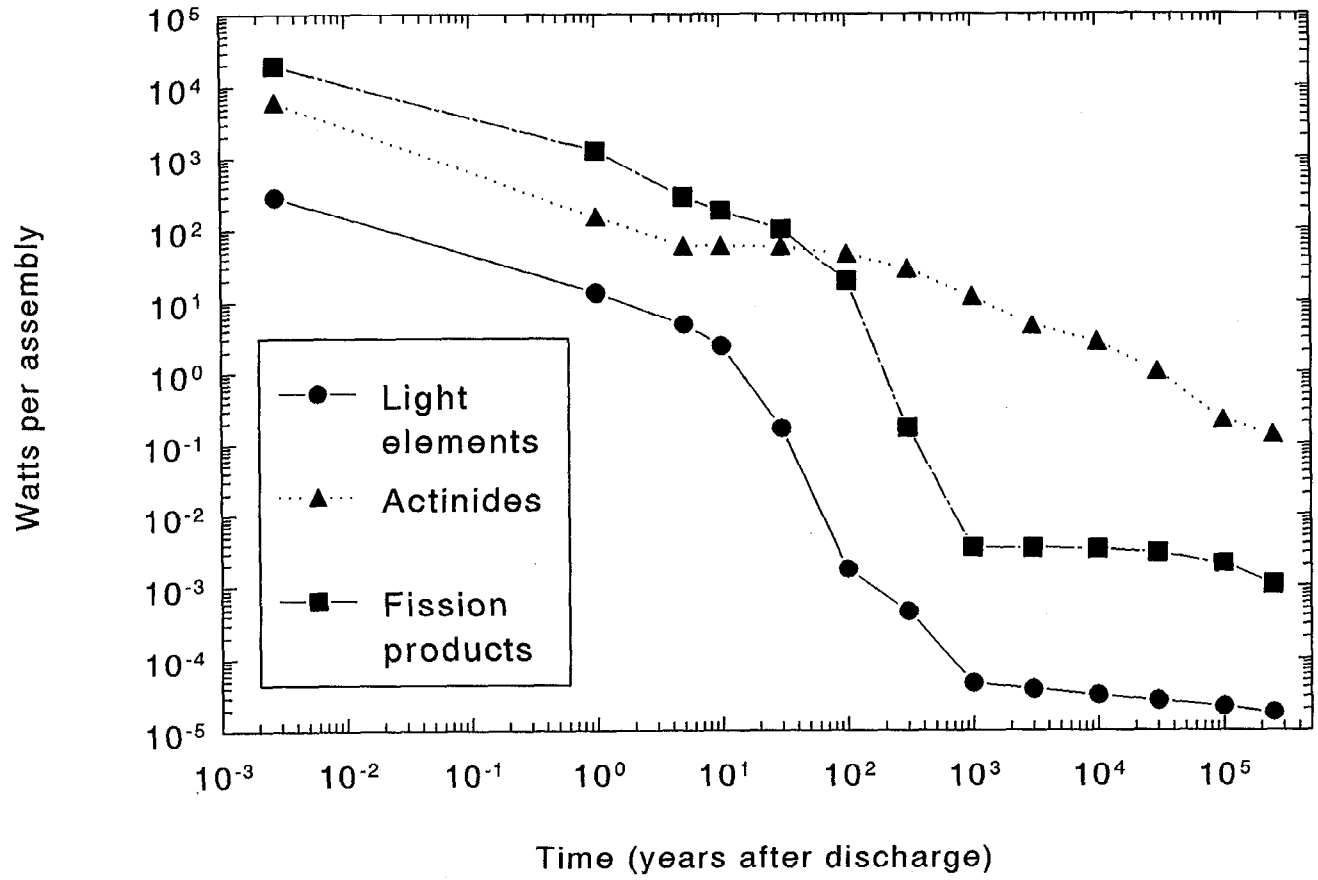


Fig. 4.18. Decay heat from a BWR-5 LEU fuel assembly.

Table 4.15. Neutron dose rate 1 m from a bare BWR-5 LEU assembly at the axial midplane

Time after discharge (years if not specified)	Dose rate (rem h^{-1})
Discharge	9.81E-02
1 d	9.82E-02
1	5.84E-02
5	4.15E-02
10	3.45E-02
100	2.35E-03
300	1.01E-03
1000	7.17E-04
10000	2.84E-04
30000	1.25E-04
100000	8.94E-05
250000	6.78E-05

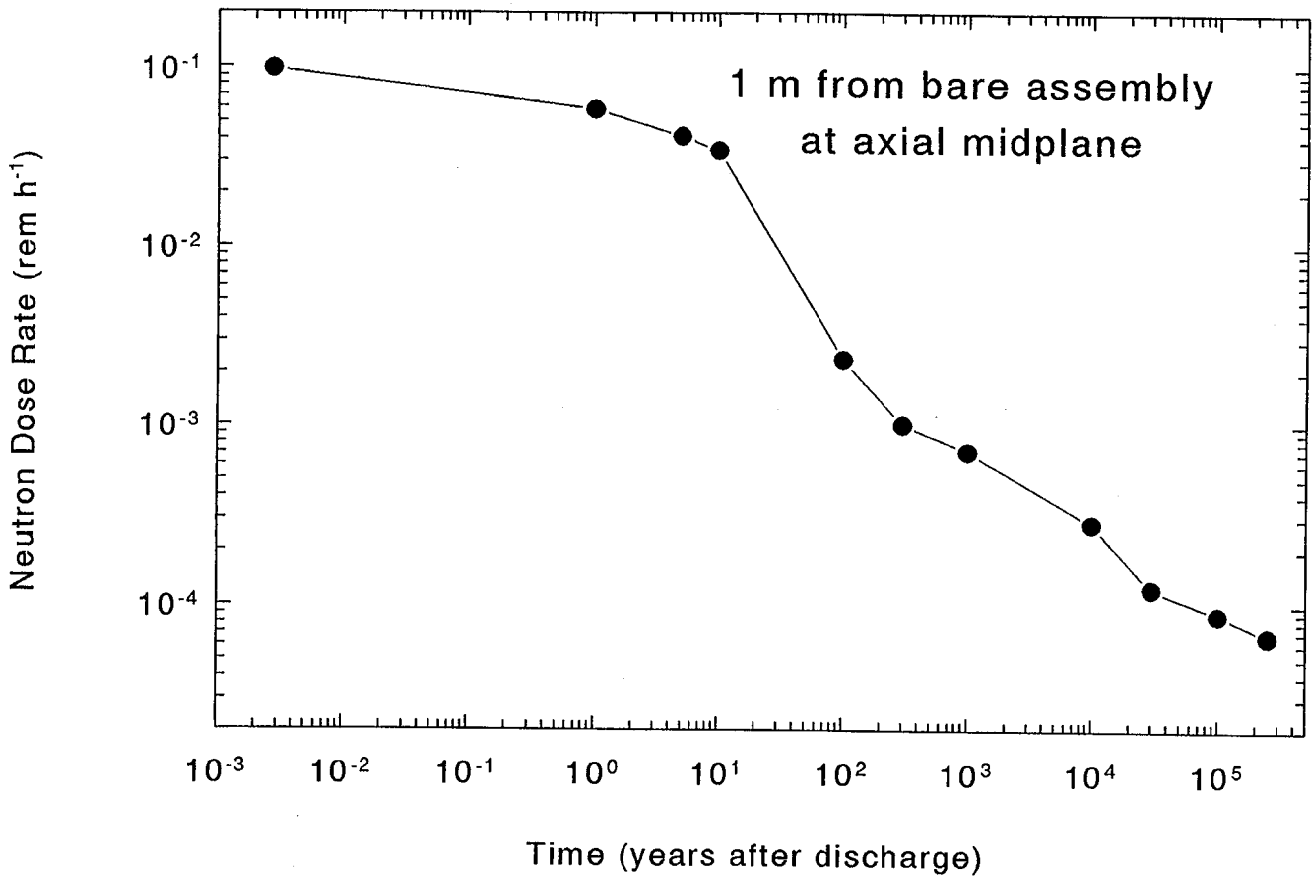


Fig. 4.19. Neutron dose rate 1 m from a BWR-5 LEU fuel assembly at the axial midplane.

Table 4.16. Gamma dose rate 1 m from a bare BWR-5 LEU assembly at the axial midplane

Time after discharge (years if not specified)	Dose rate (rem h^{-1})
Discharge	1.66E+06
1 d	1.43E+05
1	4.89E+03
5	1.61E+03
10	9.36E+02
100	8.56E+01
300	8.82E-01
1000	3.22E-02
10000	2.40E-02
30000	3.06E-02
100000	4.66E-02
250000	4.87E-02

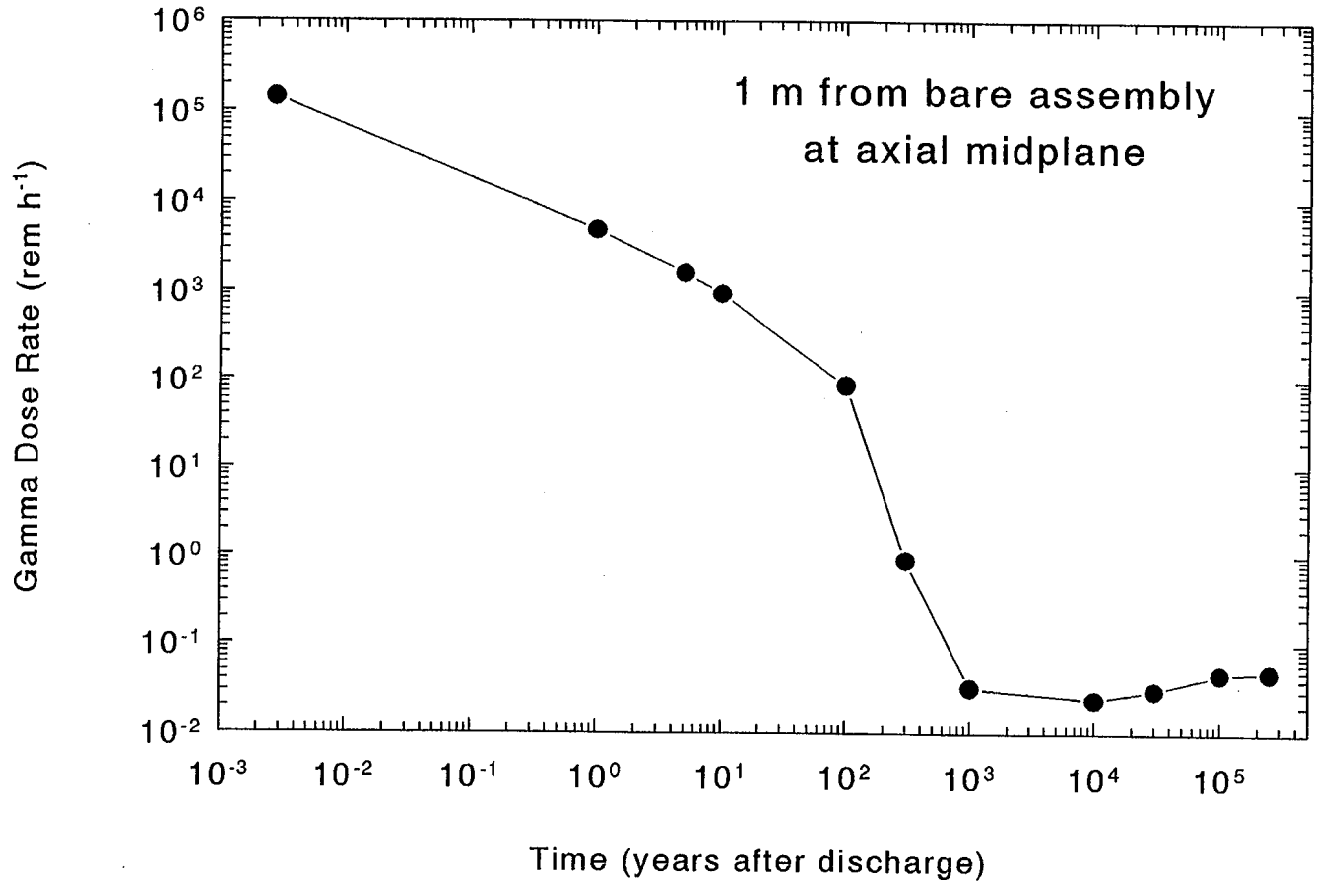


Fig. 4.20. Gamma dose rate 1 m from a bare BWR-5 LEU fuel assembly at the axial midplane.

Table 4.17. BWR cask dose rates (rem h^{-1}) from 20 BWR-5 LEU assemblies

Time after discharge (years if not specified)	Surface neutron dose rate	Surface gamma dose rate	1 meter neutron dose rate	1 meter gamma dose rate
Discharge	4.89E-03	6.34E+02	1.89E-03	2.98E+02
1 d	4.90E-03	1.35E+01	1.89E-03	6.18E+00
1	2.89E-03	1.66E-01	1.11E-03	7.61E-02
5	2.04E-03	2.26E-02	7.86E-04	1.01E-02
10	1.70E-03	9.10E-03	6.54E-04	4.04E-03
100	1.19E-04	2.20E-04	4.59E-05	9.48E-05
300	5.16E-05	2.61E-05	1.99E-05	1.07E-05
1000	3.61E-05	1.75E-05	1.39E-05	7.17E-06
10000	1.41E-05	8.17E-06	5.44E-06	3.41E-06
30000	6.20E-06	7.31E-06	2.39E-06	3.21E-06
100000	4.40E-06	1.18E-05	1.70E-06	5.30E-06
250000	3.34E-06	1.28E-05	1.29E-06	5.81E-06

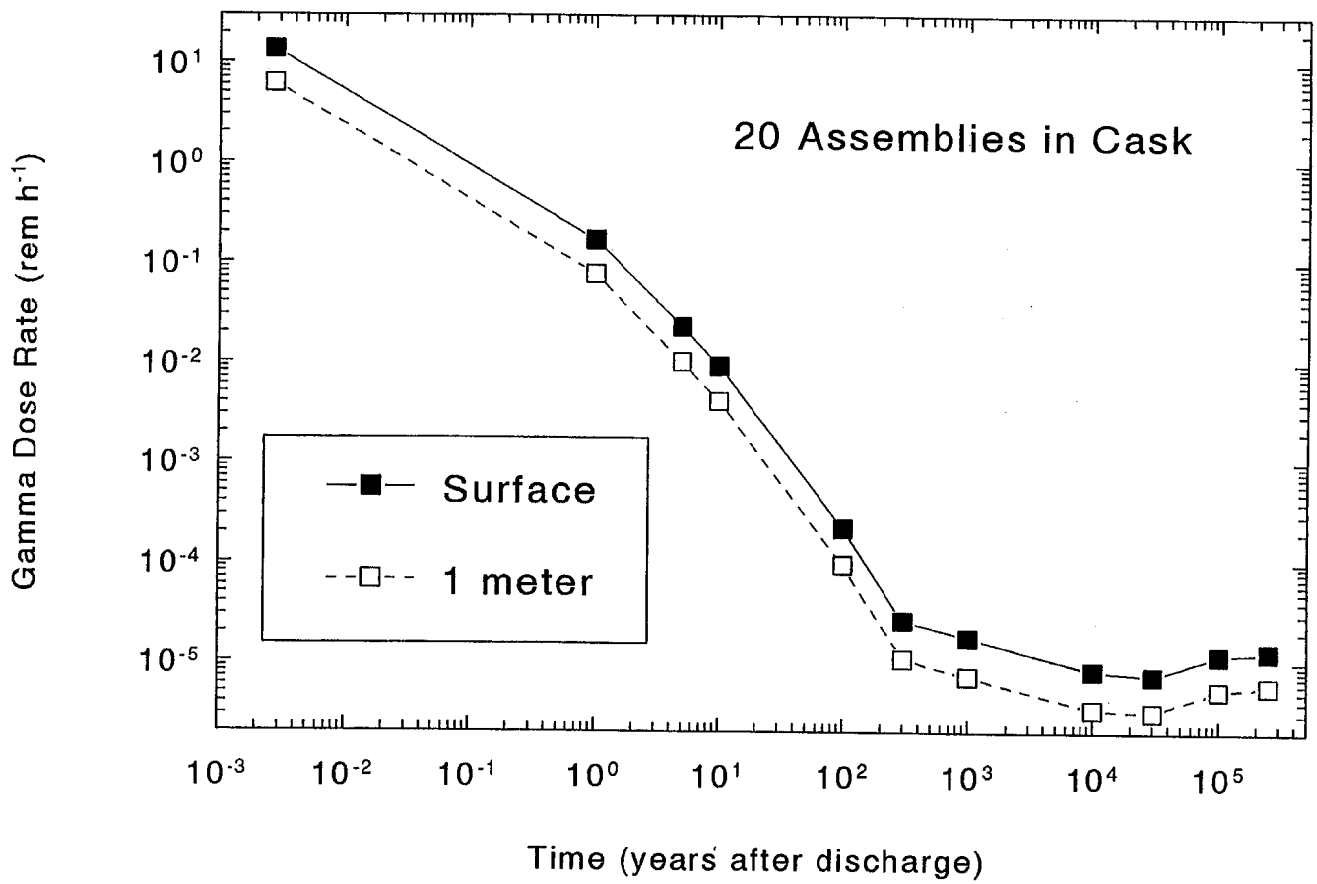


Fig. 4.21. BWR cask gamma dose rates from 20 BWR-5 LEU fuel assemblies.

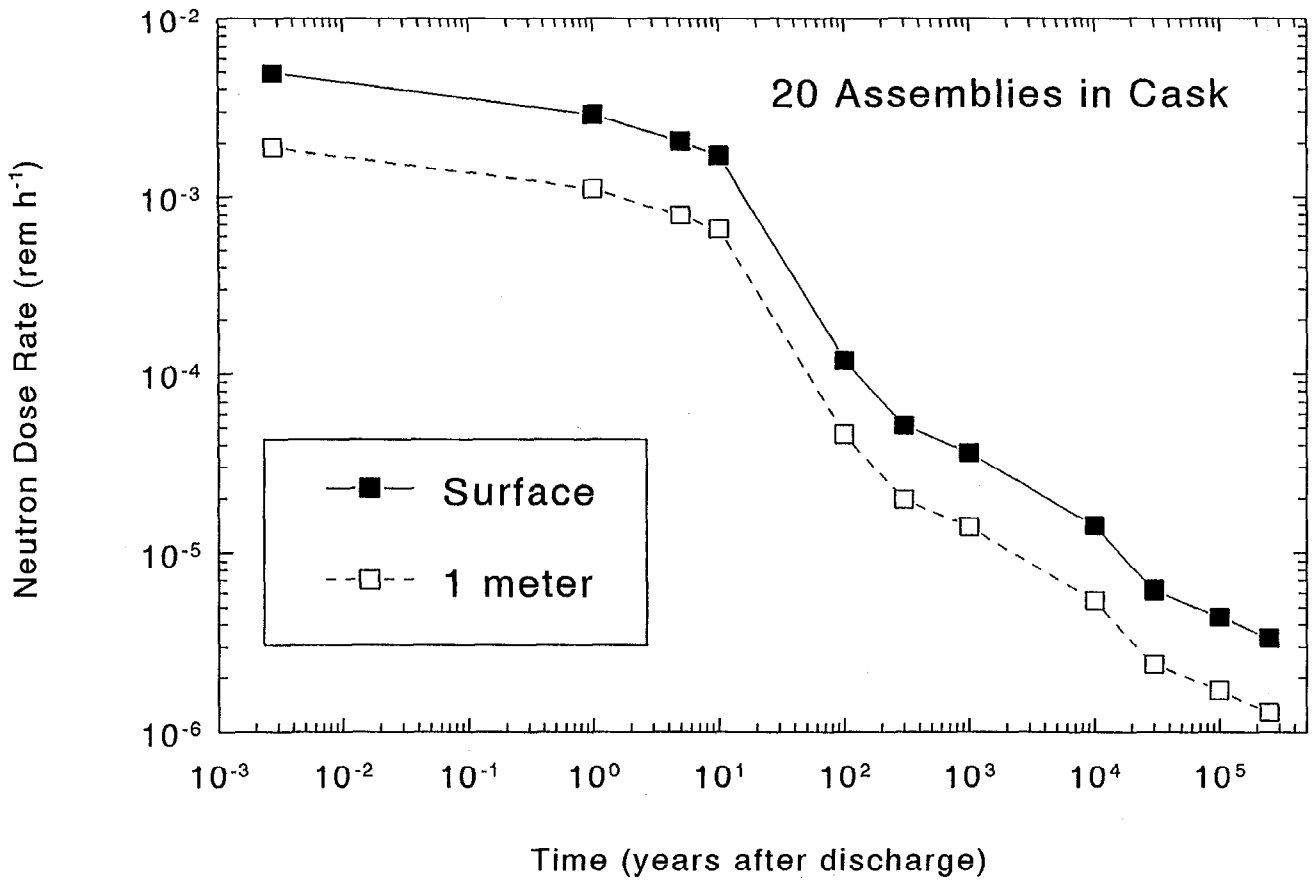


Fig. 4.22. BWR cask neutron dose rates from 20 BWR-5 LEU fuel assemblies.

Table 4.18. Nuclide inhalation hazard from a BWR-5 LEU assembly
(m^3 of air at rcg per assembly)

Time after discharge (years if not specified)	Light elements	Actinides	Fission products	Total
Discharge	2.56E+14	2.81E+16	1.08E+16	3.92E+16
1 d	9.54E+13	2.80E+16	5.31E+15	3.34E+16
1	1.24E+13	2.58E+16	1.07E+15	2.69E+16
5	1.41E+12	2.39E+16	5.33E+14	2.44E+16
10	5.52E+11	2.24E+16	4.40E+14	2.28E+16
30	5.32E+10	1.82E+16	2.62E+14	1.85E+16
100	9.91E+09	1.22E+16	4.69E+13	1.22E+16
300	2.19E+09	6.87E+15	3.61E+11	6.87E+15
1000	6.29E+07	3.59E+15	5.09E+09	3.59E+15
3000	4.51E+07	2.28E+15	4.99E+09	2.28E+15
10000	4.32E+07	1.41E+15	4.72E+09	1.41E+15
30000	4.10E+07	5.35E+14	4.09E+09	5.35E+14
100000	3.59E+07	7.68E+13	2.84E+09	7.68E+13
250000	3.40E+07	2.07E+13	1.85E+09	2.07E+13

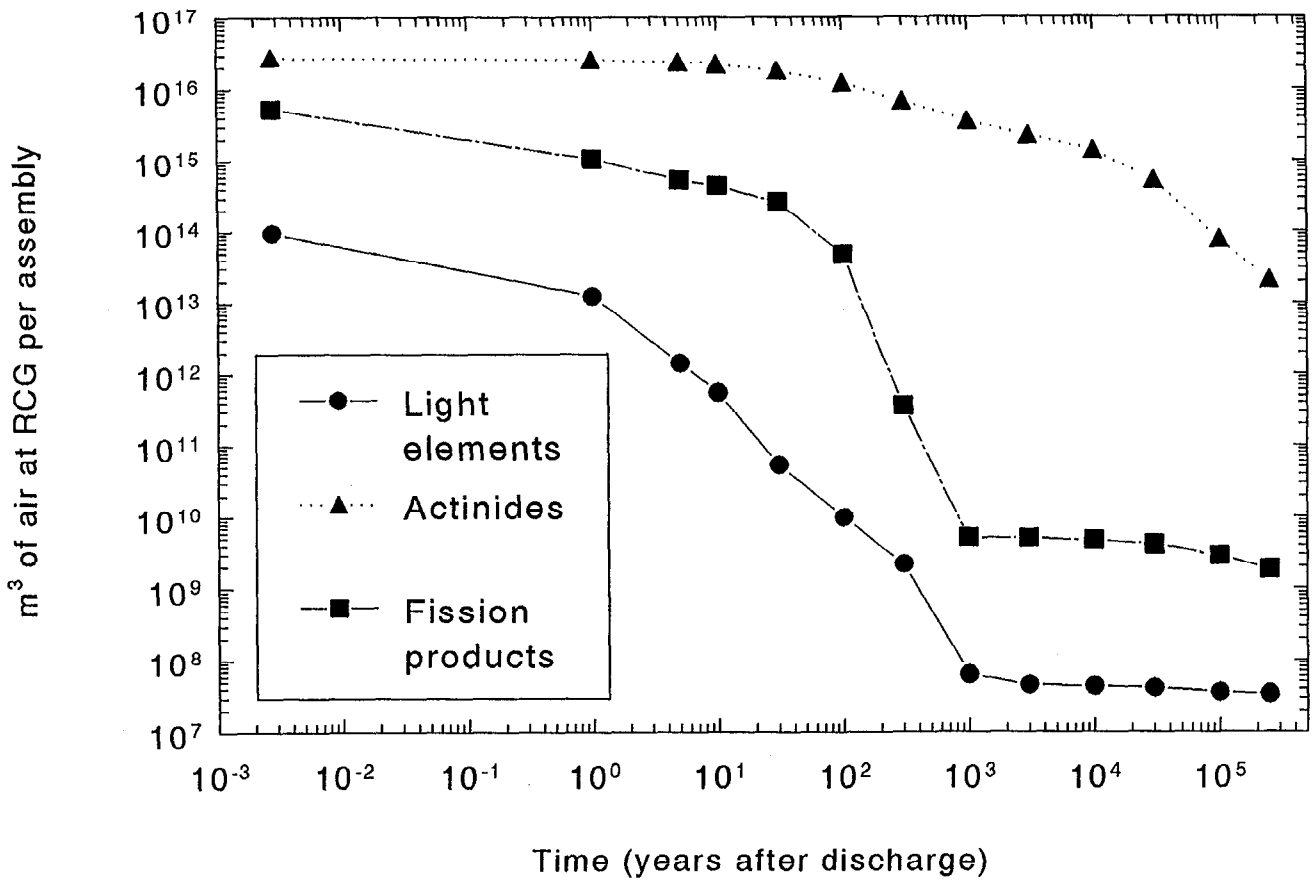


Fig. 4.23. Nuclide inhalation hazard from a BWR-5 LEU fuel assembly.

Table 4.19. Nuclide ingestion hazard from a BWR-5 LEU assembly
(m^3 of water at rcg per assembly)

Time after discharge (years if not specified)	Light elements	Actinides	Fission products	Total
Discharge	4.48E+09	3.27E+10	3.68E+13	3.68E+13
1 d	3.29E+09	2.36E+10	3.61E+13	3.61E+13
1	3.62E+08	5.70E+08	6.17E+10	6.26E+10
5	2.25E+07	4.37E+08	4.64E+10	4.69E+10
10	6.66E+06	4.31E+08	4.02E+10	4.06E+10
30	1.37E+06	4.08E+08	2.45E+10	2.49E+10
100	5.94E+05	3.19E+08	4.38E+09	4.70E+09
300	1.39E+05	2.02E+08	3.23E+07	2.34E+08
1000	2.56E+03	8.20E+07	2.25E+05	8.22E+07
3000	1.42E+03	3.01E+07	2.23E+05	3.03E+07
10000	1.30E+03	1.88E+07	2.14E+05	1.90E+07
30000	1.11E+03	1.16E+07	1.95E+05	1.18E+07
100000	6.88E+02	1.25E+07	1.59E+05	1.27E+07
250000	3.33E+02	1.40E+07	1.35E+05	1.41E+07

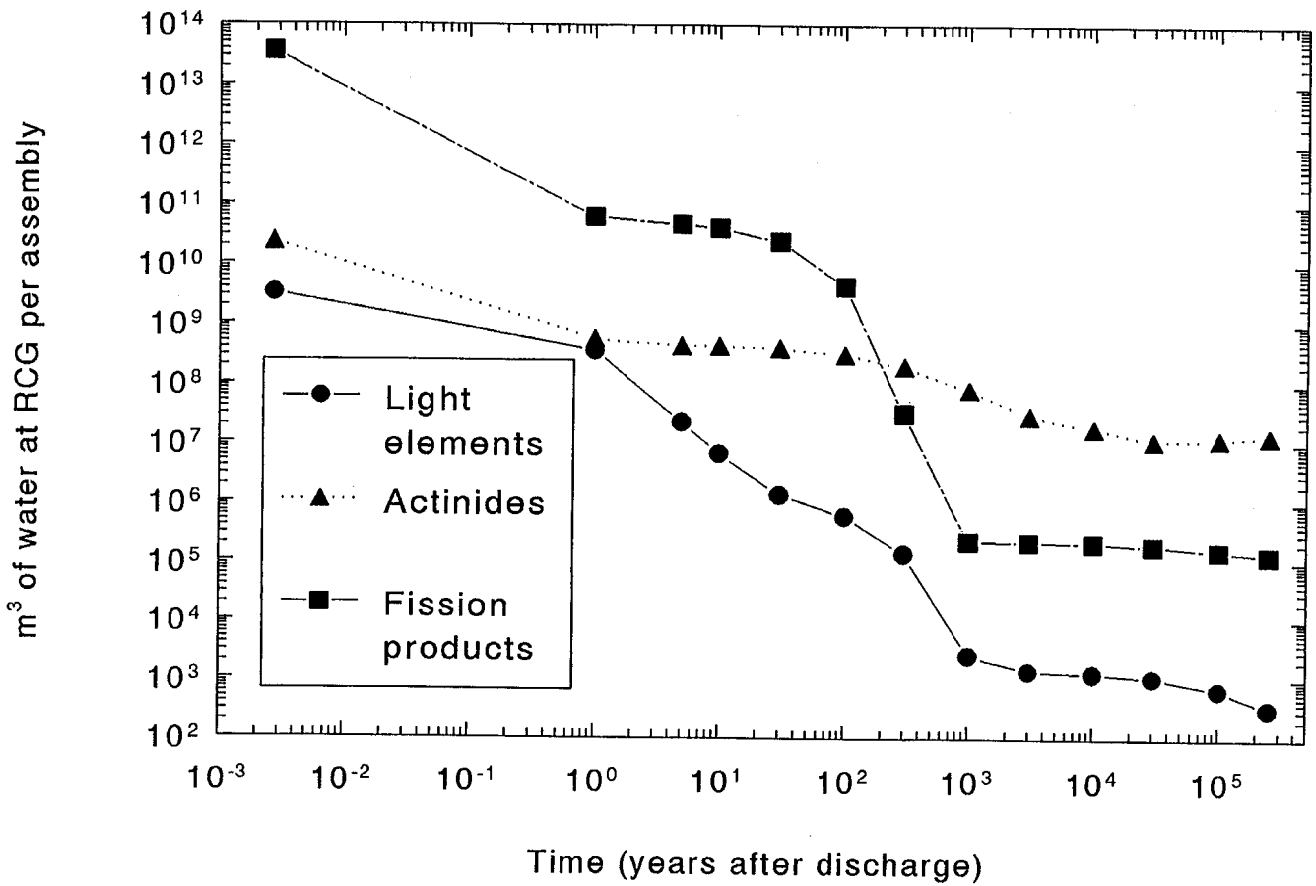


Fig. 4.24. Nuclide ingestion hazard from a BWR-5 LEU fuel assembly.

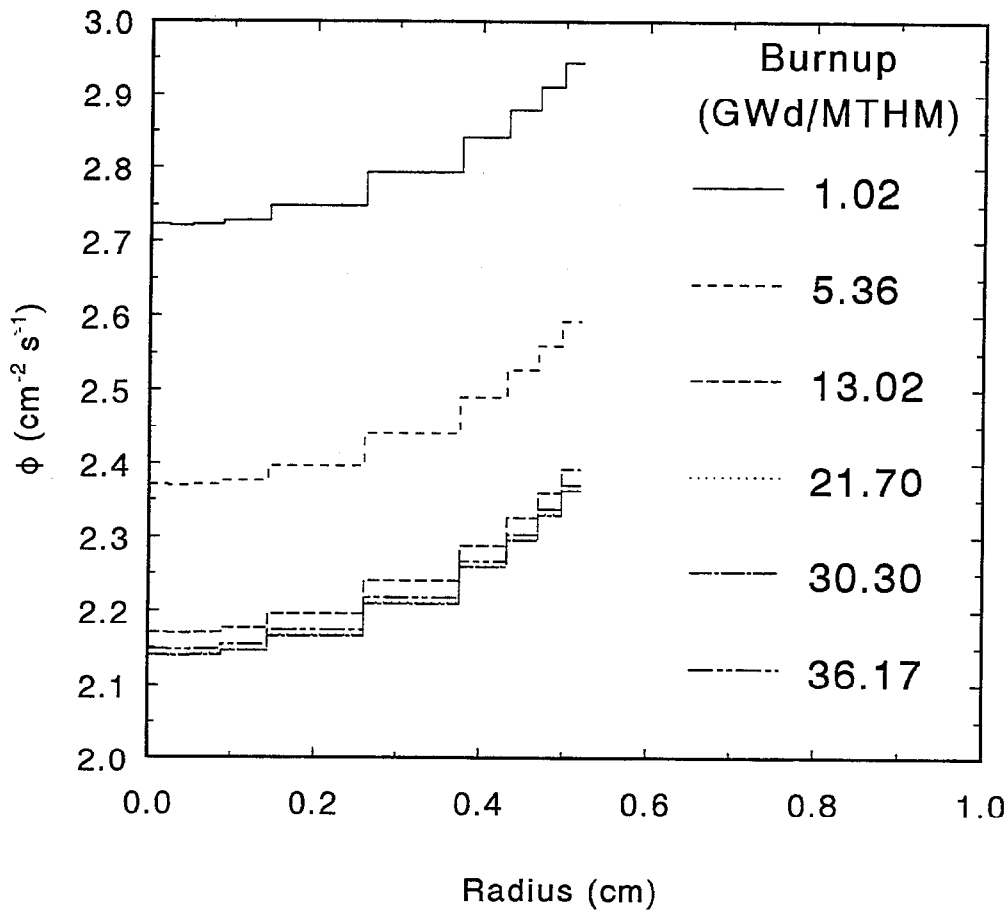


Fig. 4.25. Thermal flux in a LEU fuel pin for different assembly burnups.

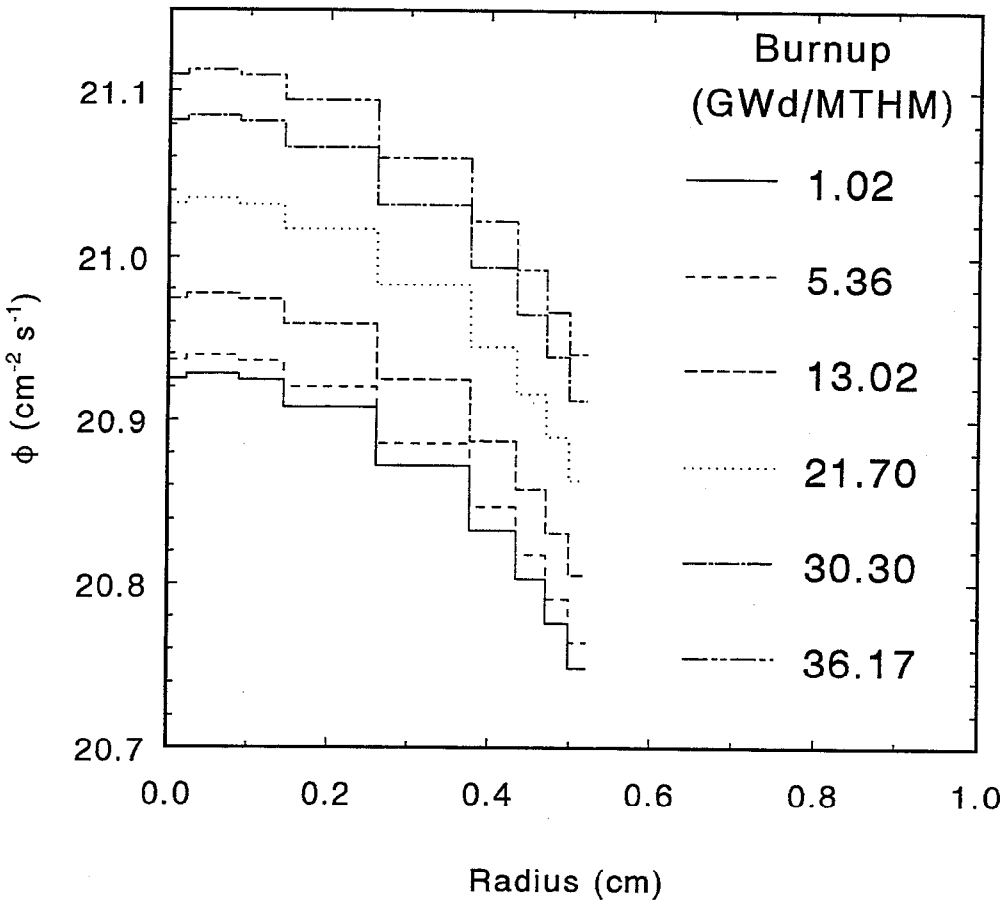


Fig. 4.26. Fast flux in a LEU fuel pin for different assembly burnups.

Table 4.20. Nuclides of interest for criticality safety in a BWR-5 MOX assembly (g per assembly)

Nuclide	Time after discharge						
	Discharge	1 d	1 year	5 years	10 years	30 years	100 years
^{16}O	4.72E+04	4.72E+04	4.72E+04	4.72E+04	4.72E+04	4.72E+04	4.72E+04
^{95}Mo	1.25E+02	1.25E+02	1.33E+02	1.34E+02	1.34E+02	1.34E+02	1.34E+02
^{99}Tc	1.52E+02	1.52E+02	1.52E+02	1.52E+02	1.52E+02	1.52E+02	1.52E+02
^{101}Ru	1.63E+02	1.63E+02	1.63E+02	1.63E+02	1.63E+02	1.63E+02	1.63E+02
^{103}Rh	1.23E+02	1.23E+02	1.29E+02	1.29E+02	1.29E+02	1.29E+02	1.29E+02
^{105}Pd	1.45E+02	1.46E+02	1.46E+02	1.46E+02	1.46E+02	1.46E+02	1.46E+02
^{109}Ag	3.71E+01	3.71E+01	3.71E+01	3.71E+01	3.71E+01	3.71E+01	3.71E+01
^{131}Xe	9.42E+01	9.43E+01	9.53E+01	9.53E+01	9.53E+01	9.53E+01	9.53E+01
^{133}Cs	2.24E+02	2.24E+02	2.25E+02	2.25E+02	2.25E+02	2.25E+02	2.25E+02
^{141}Pr	2.01E+02	2.02E+02	2.08E+02	2.08E+02	2.08E+02	2.08E+02	2.08E+02
^{143}Nd	1.38E+02	1.38E+02	1.40E+02	1.40E+02	1.40E+02	1.40E+02	1.40E+02
^{145}Nd	1.16E+02	1.16E+02	1.16E+02	1.16E+02	1.16E+02	1.16E+02	1.16E+02
^{147}Sm	2.08E+01	2.08E+01	2.69E+01	4.02E+01	4.54E+01	4.73E+01	4.73E+01
^{149}Sm	4.44E-01	4.93E-01	6.28E-01	6.28E-01	6.28E-01	6.28E-01	6.28E-01
^{150}Sm	6.17E+01	6.17E+01	6.17E+01	6.17E+01	6.17E+01	6.17E+01	6.17E+01
^{151}Sm	3.39E+00	3.40E+00	3.40E+00	3.30E+00	3.17E+00	2.72E+00	1.59E+00
^{152}Sm	3.11E+01	3.11E+01	3.11E+01	3.11E+01	3.11E+01	3.11E+01	3.11E+01
^{151}Eu	6.90E-03	6.97E-03	3.32E-02	1.36E-01	2.61E-01	7.14E-01	1.85E+00
^{153}Eu	2.87E+01	2.87E+01	2.89E+01	2.89E+01	2.89E+01	2.89E+01	2.89E+01
^{154}Eu	1.15E+01	1.15E+01	1.06E+01	7.68E+00	5.13E+00	1.02E+00	2.60E-03
^{155}Gd	1.39E-01	1.41E-01	7.45E-01	2.44E+00	3.54E+00	4.49E+00	4.54E+00
^{233}U	3.64E-05	3.64E-05	4.74E-05	9.02E-05	1.43E-04	3.93E-04	2.04E-03
^{234}U	4.57E-01	4.58E-01	7.78E-01	2.18E+00	3.88E+00	1.00E+01	2.54E+01
^{235}U	1.04E+02	1.04E+02	1.04E+02	1.04E+02	1.04E+02	1.05E+02	1.08E+02
^{236}U	4.39E+01	4.39E+01	4.40E+01	4.45E+01	4.51E+01	4.75E+01	5.60E+01
^{238}U	1.68E+05	1.68E+05	1.68E+05	1.68E+05	1.68E+05	1.68E+05	1.68E+05
^{237}Np	3.08E+01	3.09E+01	3.13E+01	3.20E+01	3.36E+01	4.53E+01	1.03E+02
^{238}Pu	3.66E+01	3.67E+01	4.40E+01	4.46E+01	4.29E+01	3.67E+01	2.13E+01
^{239}Pu	1.40E+03	1.41E+03	1.42E+03	1.42E+03	1.42E+03	1.41E+03	1.41E+03
^{240}Pu	1.16E+03	1.16E+03	1.16E+03	1.16E+03	1.16E+03	1.17E+03	1.16E+03
^{241}Pu	5.57E+02	5.57E+02	5.31E+02	4.37E+02	3.44E+02	1.31E+02	4.44E+00
^{242}Pu	2.43E+02	2.43E+02	2.43E+02	2.43E+02	2.43E+02	2.43E+02	2.43E+02
^{241}Pu	4.20E+01	4.21E+01	6.82E+01	1.61E+02	2.53E+02	4.54E+02	5.22E+02
^{242m}Am	1.17E+00	1.17E+00	1.16E+00	1.14E+00	1.11E+00	1.01E+00	7.14E-01
^{243}Am	5.97E+01	5.97E+01	5.97E+01	5.97E+01	5.96E+01	5.95E+01	5.91E+01
^{245}Cm	9.74E-01	9.74E-01	9.74E-01	9.74E-01	9.73E-01	9.72E-01	9.66E-01

Table 4.20 (continued)

Nuclide	Time after discharge (years)						
	300	1000	3000	10000	30000	100000	250000
^{16}O	4.72E+04	4.72E+04	4.72E+04	4.72E+04	4.72E+04	4.72E+04	4.72E+04
^{95}Mo	1.34E+02	1.34E+02	1.34E+02	1.34E+02	1.34E+02	1.34E+02	1.34E+02
^{99}Tc	1.52E+02	1.52E+02	1.51E+02	1.47E+02	1.38E+02	1.10E+02	6.70E+01
^{101}Ru	1.63E+02	1.63E+02	1.63E+02	1.63E+02	1.63E+02	1.63E+02	1.63E+02
^{103}Rh	1.29E+02	1.29E+02	1.29E+02	1.29E+02	1.29E+02	1.29E+02	1.29E+02
^{105}Pd	1.46E+02	1.46E+02	1.46E+02	1.46E+02	1.46E+02	1.46E+02	1.46E+02
^{109}Ag	3.71E+01	3.71E+01	3.71E+01	3.71E+01	3.71E+01	3.71E+01	3.71E+01
^{131}Xe	9.53E+01	9.53E+01	9.53E+01	9.53E+01	9.53E+01	9.53E+01	9.53E+01
^{133}Cs	2.25E+02	2.25E+02	2.25E+02	2.25E+02	2.25E+02	2.25E+02	2.25E+02
^{141}Pr	2.08E+02	2.08E+02	2.08E+02	2.08E+02	2.08E+02	2.08E+02	2.08E+02
^{143}Nd	1.40E+02	1.40E+02	1.40E+02	1.40E+02	1.40E+02	1.40E+02	1.40E+02
^{145}Nd	1.16E+02	1.16E+02	1.16E+02	1.16E+02	1.16E+02	1.16E+02	1.16E+02
^{147}Sm	4.73E+01	4.73E+01	4.73E+01	4.73E+01	4.73E+01	4.73E+01	4.73E+01
^{149}Sm	6.28E-01	6.28E-01	6.28E-01	6.28E-01	6.28E-01	6.28E-01	6.28E-01
^{150}Sm	6.17E+01	6.17E+01	6.17E+01	6.17E+01	6.17E+01	6.17E+01	6.17E+01
^{151}Sm	3.40E-01	1.55E-03	3.16E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{152}Sm	3.11E+01	3.11E+01	3.11E+01	3.11E+01	3.11E+01	3.11E+01	3.11E+01
^{151}Eu	3.09E+00	3.43E+00	3.43E+00	3.43E+00	3.43E+00	3.43E+00	3.43E+00
^{153}Eu	2.89E+01	2.89E+01	2.89E+01	2.89E+01	2.89E+01	2.89E+01	2.89E+01
^{154}Eu	3.53E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{155}Gd	4.54E+00	4.54E+00	4.54E+00	4.54E+00	4.54E+00	4.54E+00	4.54E+00
^{233}U	1.34E-02	1.02E-01	4.73E-01	1.83E+00	5.46E+00	1.58E+01	2.89E+01
^{234}U	4.22E+01	4.68E+01	4.66E+01	4.59E+01	4.39E+01	3.77E+01	2.78E+01
^{235}U	1.16E+02	1.43E+02	2.20E+02	4.57E+02	9.33E+02	1.47E+03	1.55E+03
^{236}U	8.00E+01	1.60E+02	3.58E+02	7.99E+02	1.15E+03	1.20E+03	1.19E+03
^{238}U	1.68E+05	1.68E+05	1.68E+05	1.68E+05	1.68E+05	1.68E+05	1.68E+05
^{237}Np	2.45E+02	4.98E+02	6.15E+02	6.19E+02	6.15E+02	6.02E+02	5.73E+02
^{238}Pu	4.54E+00	2.80E-02	6.20E-07	7.01E-22	0.00E+00	0.00E+00	0.00E+00
^{239}Pu	1.41E+03	1.38E+03	1.31E+03	1.09E+03	6.28E+02	8.46E+01	1.13E+00
^{240}Pu	1.14E+03	1.06E+03	8.57E+02	4.09E+02	4.95E+01	3.04E-02	1.52E-08
^{241}Pu	1.86E-03	1.49E-03	1.27E-03	7.17E-04	1.40E-04	4.65E-07	2.26E-12
^{242}Pu	2.43E+02	2.43E+02	2.42E+02	2.39E+02	2.30E+02	2.02E+02	1.53E+02
^{241}Am	3.82E+02	1.24E+02	5.09E+00	2.17E-02	4.23E-03	1.40E-05	7.17E-11
^{242m}Am	2.67E-01	8.55E-03	4.59E-07	5.22E-22	0.00E+00	0.00E+00	0.00E+00
^{243}Am	5.80E+01	5.43E+01	4.50E+01	2.33E+01	3.55E+00	4.92E-03	1.53E-06
^{245}Cm	9.51E-01	8.98E-01	7.63E-01	4.31E-01	8.43E-02	2.79E-04	1.36E-09

Table 4.21. Nuclides of interest for criticality safety in a BWR-5 LEU assembly (g per assembly)

Nuclide	Time after discharge						
	Discharge	1 d	1 year	5 years	10 years	30 years	100 years
^{16}O	4.84E+04	4.84E+04	4.84E+04	4.84E+04	4.84E+04	4.84E+04	4.84E+04
^{95}Mo	1.52E+02	1.52E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02
^{99}Tc	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02
^{101}Ru	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02
^{103}Rh	8.81E+01	8.82E+01	9.38E+01	9.39E+01	9.39E+01	9.39E+01	9.39E+01
^{105}Pd	8.64E+01	8.65E+01	8.66E+01	8.66E+01	8.66E+01	8.66E+01	8.66E+01
^{109}Ag	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01
^{131}Xe	8.74E+01	8.75E+01	8.85E+01	8.85E+01	8.85E+01	8.85E+01	8.85E+01
^{133}Cs	2.31E+02	2.31E+02	2.32E+02	2.32E+02	2.32E+02	2.32E+02	2.32E+02
^{141}Pr	2.23E+02	2.23E+02	2.30E+02	2.30E+02	2.30E+02	2.30E+02	2.30E+02
^{143}Nd	1.57E+02	1.57E+02	1.60E+02	1.60E+02	1.60E+02	1.60E+02	1.60E+02
^{145}Nd	1.37E+02	1.37E+02	1.37E+02	1.37E+02	1.37E+02	1.37E+02	1.37E+02
^{147}Sm	2.23E+01	2.23E+01	2.88E+01	4.29E+01	4.84E+01	5.04E+01	5.04E+01
^{149}Sm	3.92E-01	4.43E-01	5.81E-01	5.81E-01	5.81E-01	5.81E-01	5.81E-01
^{150}Sm	6.10E+01	6.10E+01	6.10E+01	6.10E+01	6.10E+01	6.10E+01	6.10E+01
^{151}Sm	2.83E+00	2.85E+00	2.85E+00	2.76E+00	2.65E+00	2.28E+00	1.33E+00
^{152}Sm	2.64E+01	2.64E+01	2.64E+01	2.64E+01	2.64E+01	2.64E+01	2.64E+01
^{151}Eu	4.98E-03	5.04E-03	2.70E-02	1.13E-01	2.18E-01	5.97E-01	1.54E+00
^{153}Eu	2.28E+01	2.28E+01	2.29E+01	2.29E+01	2.29E+01	2.29E+01	2.29E+01
^{154}Eu	8.71E+00	8.71E+00	8.03E+00	5.82E+00	3.89E+00	7.74E-01	2.73E-03
^{155}Gd	3.20E-02	3.33E-02	4.73E-01	1.71E+00	2.50E+00	3.19E+00	3.23E+00
^{233}U	4.91E-04	4.91E-04	5.27E-04	6.62E-04	8.24E-04	1.49E-03	4.17E-03
^{234}U	2.87E+01	2.87E+01	2.91E+01	3.05E+01	3.22E+01	3.83E+01	5.36E+01
^{235}U	1.31E+03	1.32E+03	1.31E+03	1.31E+03	1.31E+03	1.31E+03	1.31E+03
^{236}U	8.16E+02	8.16E+02	8.16E+02	8.16E+02	8.16E+02	8.17E+02	8.20E+02
^{238}U	1.72E+05	1.72E+05	1.72E+05	1.72E+05	1.72E+05	1.72E+05	1.72E+05
^{237}Np	9.93E+01	9.94E+01	1.01E+02	1.01E+02	1.02E+02	1.07E+02	1.34E+02
^{238}Pu	4.23E+01	4.24E+01	4.51E+01	4.45E+01	4.28E+01	3.66E+01	2.11E+01
^{239}Pu	9.81E+02	9.84E+02	9.93E+02	9.93E+02	9.93E+02	9.92E+02	9.90E+02
^{240}Pu	4.49E+02	4.49E+02	4.50E+02	4.51E+02	4.51E+02	4.53E+02	4.53E+02
^{241}Pu	2.64E+02	2.64E+02	2.52E+02	2.07E+02	1.63E+02	6.20E+01	2.11E+00
^{242}Pu	1.11E+02	1.11E+02	1.11E+02	1.11E+02	1.11E+02	1.11E+02	1.11E+02
^{241}Am	1.49E+01	1.49E+01	2.73E+01	7.12E+01	1.15E+02	2.10E+02	2.43E+02
^{242m}Am	4.14E-01	4.14E-01	4.12E-01	4.04E-01	3.94E-01	3.57E-01	2.53E-01
^{243}Am	2.68E+01	2.68E+01	2.68E+01	2.68E+01	2.68E+01	2.68E+01	2.66E+01
^{245}Cm	3.76E+00	3.76E+00	3.49E-01	3.49E-01	3.48E-01	3.48E-01	6.60E-04

Table 4.21 (continued)

Nuclide	Time after discharge (years)						
	300	1000	3000	10000	30000	100000	250000
^{16}O	4.84E+04	4.84E+04	4.84E+04	4.84E+04	4.84E+04	4.84E+04	4.84E+04
^{95}Mo	1.64E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02
^{99}Tc	1.59E+02	1.59E+02	1.58E+02	1.54E+02	1.44E+02	1.15E+02	7.00E+01
^{101}Ru	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02	1.59E+02
^{103}Rh	9.39E+01	9.39E+01	9.39E+01	9.39E+01	9.39E+01	9.39E+01	9.39E+01
^{105}Pd	8.66E+01	8.66E+01	8.66E+01	8.66E+01	8.66E+01	8.66E+01	8.66E+01
^{109}Ag	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01
^{131}Xe	8.85E+01	8.85E+01	8.85E+01	8.85E+01	8.85E+01	8.85E+01	8.85E+01
^{133}Cs	2.32E+02	2.32E+02	2.32E+02	2.32E+02	2.32E+02	2.32E+02	2.32E+02
^{141}Pr	2.30E+02	2.30E+02	2.30E+02	2.30E+02	2.30E+02	2.30E+02	2.30E+02
^{143}Nd	1.60E+02	1.60E+02	1.60E+02	1.60E+02	1.60E+02	1.60E+02	1.60E+02
^{145}Nd	1.37E+02	1.37E+02	1.37E+02	1.37E+02	1.37E+02	1.37E+02	1.37E+02
^{147}Sm	5.04E+01	5.04E+01	5.04E+01	5.04E+01	5.04E+01	5.04E+01	5.04E+01
^{149}Sm	5.81E-01	5.81E-01	5.81E-01	5.81E-01	5.81E-01	5.81E-01	5.81E-01
^{150}Sm	6.10E+01	6.10E+01	6.10E+01	6.10E+01	6.10E+01	6.10E+01	6.10E+01
^{151}Sm	2.84E-01	1.29E-03	2.64E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{152}Sm	2.64E+01	2.64E+01	2.64E+01	2.64E+01	2.64E+01	2.64E+01	2.64E+01
^{151}Eu	2.87E+00	2.87E+00	2.87E+00	2.87E+00	2.87E+00	2.87E+00	2.87E+00
^{153}Eu	2.29E+01	2.29E+01	2.29E+01	2.29E+01	2.29E+01	2.29E+01	2.29E+01
^{154}Eu	2.67E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{155}Gd	3.23E+00	3.23E+00	3.23E+00	3.23E+00	3.23E+00	3.23E+00	3.23E+00
^{233}U	1.49E-02	7.49E-02	3.01E-01	1.11E+00	3.30E+00	9.49E+00	1.74E+01
^{234}U	7.01E+01	7.44E+01	7.40E+01	7.28E+01	6.93E+01	5.85E+01	4.15E+01
^{235}U	1.32E+03	1.34E+03	1.39E+03	1.55E+03	1.88E+03	2.25E+03	2.31E+03
^{236}U	8.30E+02	8.61E+02	9.38E+02	1.11E+03	1.25E+03	1.26E+03	1.26E+03
^{238}U	1.72E+05	1.72E+05	1.72E+05	1.72E+05	1.72E+05	1.72E+05	1.72E+05
^{237}Np	2.00E+02	3.18E+02	3.73E+02	3.74E+02	3.72E+02	3.64E+02	3.46E+02
^{238}Pu	4.40E+00	2.10E-02	2.21E-07	2.49E-22	0.00E+00	0.00E+00	0.00E+00
^{239}Pu	9.85E+02	9.67E+02	9.17E+02	7.58E+02	4.33E+02	5.81E+01	7.79E-01
^{240}Pu	4.43E+02	4.12E+02	3.33E+02	1.59E+02	1.92E+01	1.18E-02	4.80E-09
^{241}Pu	7.00E-04	5.35E-04	4.54E-04	2.57E-04	5.02E-05	1.66E-07	8.08E-13
^{242}Pu	1.11E+02	1.11E+02	1.11E+02	1.09E+02	1.05E+02	9.23E+01	6.99E+01
^{241}Am	1.78E+02	5.80E+01	2.37E+00	7.77E-03	1.51E-03	5.02E-06	2.47E-11
^{242m}Am	9.48E-02	3.03E-03	1.63E-07	1.85E-22	0.00E+00	0.00E+00	0.00E+00
^{243}Am	2.61E+01	2.44E+01	2.02E+01	1.05E+01	1.60E+00	2.21E-03	4.90E-07
^{245}Cm	3.40E-01	3.21E-01	2.73E-01	1.54E-01	3.02E-02	1.00E-04	4.86E-10

Table 4.22. Comparison of plutonium discharged from a BWR-5 MOX assembly

Isotope	Quantity disharged (g per assembly)		Percentage Difference[a]
	ORNL (SAS2H/ORIGEN-S)	GE (PANACEA/TGBLA)	
^{238}Pu	3.66E+01	3.354E+01	9.1
^{239}Pu	1.42E+03[b]	1.426E+03	-0.4
^{240}Pu	1.16E+03	1.196E+03	-3.0
^{241}Pu	5.57E+02	5.109E+02	9.0
^{242}Pu	2.43E+02	2.223E+02	9.3
Total Pu	3.42E+03	3.388E+03	0.9
Pu/total heavy metal	1.99E-02	1.968E-02	1.1

[a] (ORNL/GE - 1) × 100%

[b] Includes ^{239}Np and ^{239}U at shutdown, which decays to ^{239}Pu by discharge.

Table 4.23. Comparison of decay heat from a BWR-5 MOX assembly

Decay time after shutdown (years)	Decay heat (W per assembly)		Percentage Difference[a]
	ORNL (SAS2H/ORIGEN-S)	GE (PANACEA/TGBLA)	
0	2.672E+05	2.615E+05	2.2
1	1.705E+03	1.898E+03	-10.2
10	2.554E+02	2.410E+02	6.0
100	9.940E+01	9.400E+01	5.7

[a] (ORNL/GE - 1) × 100%.

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APPENDIX A

IMPORTANT SCALE SYSTEM DATA FILES

As indicated earlier, some important files associated with the operation of the SCALE system codes are being included with this report. They are on the two attached diskettes.

The SAS2H code sequence was used to simulate the burnup of the fuel assemblies studied. Using nuclide concentrations from the SAS2H output file as input, ORIGEN-S was then used to calculate the isotopic composition and associated characteristics of an assembly following discharge. ORIGEN-S also provided source terms that were used by the SAS1X code sequence to determine neutron and gamma dose rates for the spent-fuel assemblies.

The files contained on the first diskette attached to this report are all for BWR MOX fuel. They are:

File Name	Description
ORCIHAZ.INP	ORIGEN-S input to generate "Ci" and hazard tables
ORCIHAZ.OUT	ORIGEN-S output with "Ci" and hazard tables
ORGRAM.INP	ORIGEN-S input to generate "gram" tables
ORGRAM.OUT	ORIGEN-S output with "gram" tables
ORHEATSO.INP	ORIGEN-S input to generate decay heat tables and source terms for SAS1X
ORHEATSO.OUT	ORIGEN-S output with decay heat and source tables
ORIGEN.BAT	Script file to run SCALE 4.2p version of ORIGEN-S
README.MOX	File describing all files on diskette
SAS1BARE.INP	SAS1X input for bare fuel assembly
SAS1CASK.INP	SAS1X input for cask of 20 fuel assemblies
SAS1X.BAT	Script file to run SCALE 4.2p version of SAS1X
SAS2H.BAT	Script file to run SCALE 4.2p version of SAS2H
SAS2H.INP	SAS2H input for BWR fuel assembly

The files contained on the second diskette attached to this report are all for BWR LEU fuel. They are named identically to those on the first diskette except that the description file is called README.LEU.

The output files from the SAS2H and SAS1X cases are too long to include on the diskettes.

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