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Characteristics of Spent Fuel from Plutonium Disposition Reactors Vol. 3: A Westinghouse Pressurized-Water Reactor Design

B. D. Murphy

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CHARACTERISTICS OF SPENT FUEL FROM PLUTONIUM DISPOSITION REACTORS
VOL. 3: A WESTINGHOUSE PRESSURIZED-WATER REACTOR DESIGN

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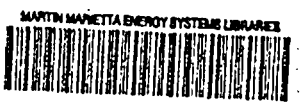

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ABSTRACT

This report discusses the results of a simulation study involving the burnup of mixed-oxide (MOX) fuel in a Westinghouse pressurized-water reactor (PWR). The MOX was composed of uranium and plutonium oxides, where the plutonium was of weapons-grade composition. The study was part of the Fissile Materials Disposition Program and considered the possibility of fueling commercial reactors with weapons plutonium. The isotopic composition, the activities, and the decay heat, together with gamma and neutron dose rates are discussed for the spent fuel. For the steady-state situation involving this PWR burning MOX fuel, two burn histories are reported. In one case, an assembly is burned in the reactor for two cycles, and in the second case an assembly is burned for three cycles. Furthermore, assemblies containing wet annular burnable absorbers (WABAs) and assemblies that do not contain WABAs are considered in all cases. The two-cycle cases have a burnup of 35 GWd/t, and the three-cycle cases have a burnup of 52.5 GWd/t.

1. INTRODUCTION

The Fissile Materials Disposition Program (FMDP) has conducted studies on the burning of fuel containing weapons-grade plutonium. The plutonium is contained in a mixed oxide (MOX) of uranium and plutonium. These studies have been conducted for a variety of existing reactor designs. The characteristics of spent fuel in such instances are important for decision-making purposes related to the disposition of fissile material. Various reports have been compiled detailing such characteristics and comparing spent fuel originating from admixtures of weapons-grade plutonium with regular uranium fuel. In particular, a detailed report has been issued on the studies conducted for a Combustion Engineering System 80+ reactor,[1] and the current report on studies conducted on a Westinghouse pressurized-water reactor (PWR) is intended to provide more details of this ongoing effort. Many aspects of this report are similar to the report on the Combustion Engineering reactor. As a result, some portions of this report will repeat material already presented so the current report will be self-contained and will minimize repeated references to other reports.

This report discusses the characteristics of spent MOX fuel from a Westinghouse PWR.[2] These studies are an extension of the work reported in Ref. 2. We report details concerning the characteristics of the spent fuel in a typical assembly following burnup in the reactor core. Note that the fresh fuel in an assembly contains an integral burnable absorber (a coating of ZrB_2 on the pellets). Various codes from the Oak Ridge National Laboratory (ORNL) SCALE system[3] were used in the calculations that will be described. The burnup of the fuel in the core was simulated using the SAS2H module[4] of SCALE. SAS2H starts with a definition of the contents and the physical arrangement of the materials comprising the assembly, develops neutron spectra and cross-section libraries for the burn cycles, and tracks the inventory of fuel and reaction products to discharge. Using the inventory of nuclides at discharge, the ORIGEN-S code[5] was employed to follow decay and ingrowth out to 250,000 years. For each desired time point during this decay process, ORIGEN-S makes available a file that is then used for dose calculations using the SAS1 code.[6] The important input and output data sets are stored on diskettes that are on the inside back cover of this report.

2. CHARACTERISTICS OF THE WESTINGHOUSE PRESSURIZED-WATER REACTOR ASSEMBLY

The Westinghouse PWR core contains 193 assemblies with the fuel rods arranged on a 17 × 17 grid. A diagram of the Westinghouse assembly is shown in Fig. 1. Of the 289 locations in the 17 × 17 grid, 264 contain fuel rods; the remaining 25 locations contain guide-tube thimbles. The usual arrangement is to designate the central guide tube as an instrument tube and the other 24 guide tubes as possible locations for burnable poison absorbers. In the configuration studied here and suggested for the disposition of weapons-grade plutonium, two types of assemblies exist in the core: assemblies containing wet annular burnable absorber (WABA) elements and assemblies that do not contain WABAs. When WABAs are present, they are contained in these 24 burnable absorber locations. As the term implies, the WABA contains absorber material that is encapsulated in an annular structure that is then inserted in a guide-tube thimble. As Fig. 1 also illustrates, the plutonium in the outer regions of the assembly is of a lower enrichment. In the calculations described below, the plutonium enrichment will be an average value for the assembly.

A number of core-loading designs were proposed by Westinghouse in earlier studies. A study of all assemblies associated with all core-loading designs was not possible with the available resources. Rather, this report discusses results that are typical for what is known as a full weapons-grade mixed-oxide (WG-MOX) core design. The majority of assemblies in that type of core do not have WABAs. However, assemblies both with and without WABAs will be discussed.

The fuel in each assembly is a mixture of UO_2 and PuO_2. The plutonium content is 4.56 wt % of the heavy metal. This is an assembly-averaged value. The weight percentage plutonium isotopic distribution is as follows: ^{239}Pu, 93.6%; ^{240}Pu, 5.9%; ^{241}Pu, 0.4%; and ^{242}Pu, 0.1%. The uranium in the MOX consists of tails uranium, and the weight percentage isotopic composition is ^{234}U, 0.002%; ^{235}U, 0.2%; ^{236}U, 0.001%; and ^{238}U, 99.797%. Each of the fuel rods contains an integral fuel burnable absorber (IFBA), which is a coating of ZrB_2 on the fuel pellets such that there is 1.57 mg of ^{10}B per inch (some of the values used in these calculations were obtained from design specifications that unfortunately employ mixed units). For the 193 assemblies in the PWR core, the total amount of heavy metal was 89.1 metric tons (MT). Each of the two assembly configurations (with and without WABAs) contained the same amount of fuel. The central location in the assembly is an instrument hole, and thus the other 24 nonfuel locations are either empty or contain burnable absorber rods. Table 1 summarizes the details of the fuel assembly.

The burnable absorber is composed of a mixture of B_4H and Al_2O_3. This material is in a rod with an annular cross section and with a ^{10}B content such that there is 6.03 mg/cm in the axial direction. The length of the burnable absorber material is 240.4 cm, whereas the active length of the fuel rods is 365.8 cm. However, in modeling the assembly using SAS2H (i.e., a two-dimensional cylindrical symmetry calculation), the burnable poison absorbers were treated as if they were constant throughout the total length of the assembly.

A drawing of the burnable absorber rods is shown in Fig. 2.1.1.1–16 of Ref. 2. Checks on the drawing seemed to indicate that it was not accurately to scale. However, knowing the linear density of the ^{10}B and the composition of the absorber material, the amount of B_4H-Al_2O_3 in each rod could be determined. The amount of Zircaloy in each rod was not critical to the calculations. The SAS2H model of the assembly shown in Fig. 1 is a series of concentric cylinders. In constructing a SAS2H model of the WABA-containing assembly, the 24 WABA locations could be conveniently divided between an inner zone corresponding to 8 locations and an outer zone corresponding to 16 locations. Because none of the WABA locations contains fuel, and because the burnup is the same for WABA and non-WABA assemblies, the only differences expected would arise from differences between the neutron flux spectra.

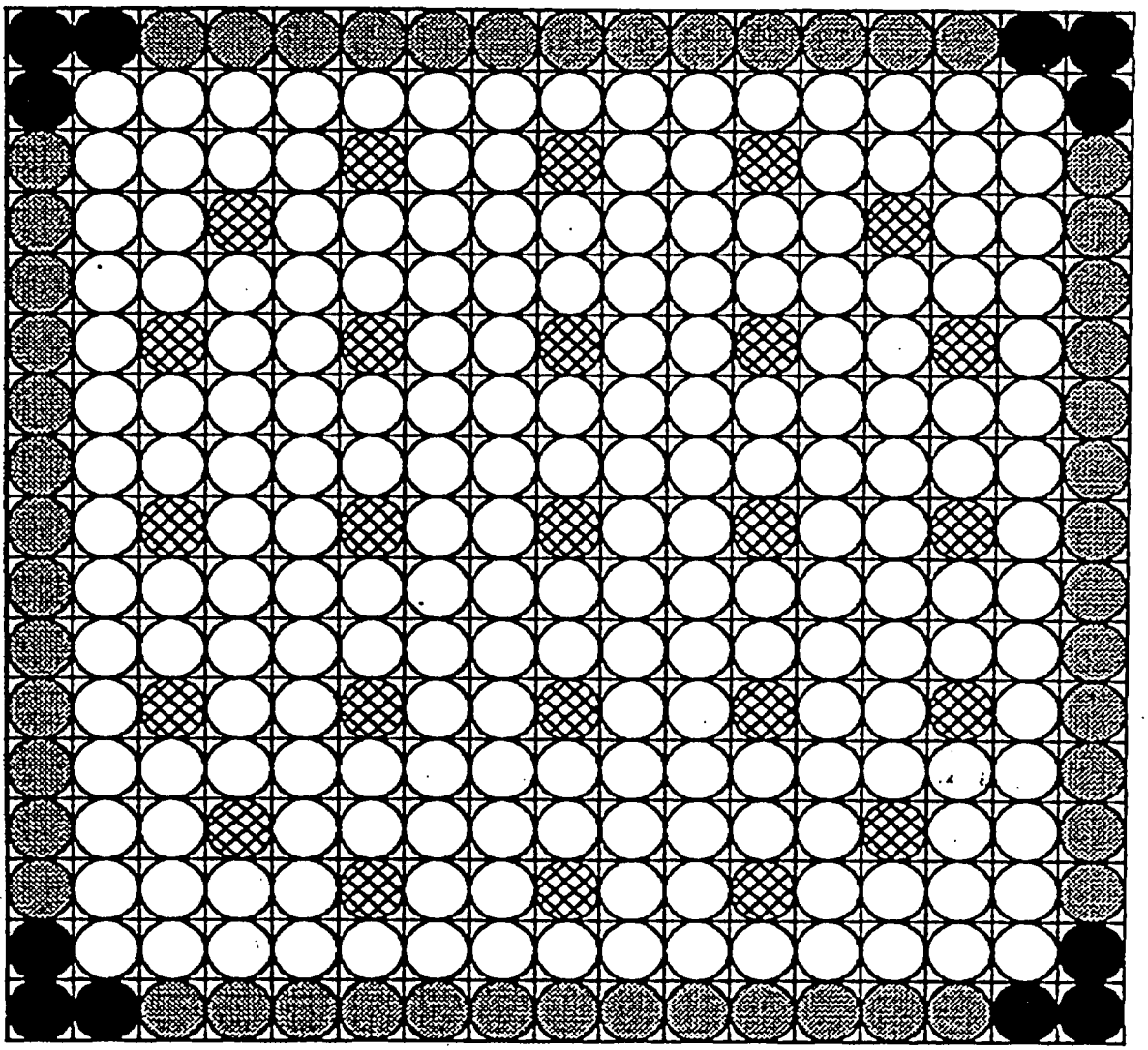

Fig. 1. The Westinghouse 17 × 17 PWR assembly

Table 1. Design basis fuel and operating parameters

Pellet diameter	0.819 cm
Rod pitch	1.26 cm
Active fuel length	365.8 cm
Clad outside diameter (OD)	0.95 cm
Clad inside diameter (ID)	0.836 cm
Clad material	Zircaloy-4
Integral fuel burnable absorber (ZrB_2)	1.57 mg/in. of ^{10}B
Boron in moderator (cycle average)	750 ppm
Plutonium fuel characteristics	
Total plutonium	4.56 wt % HM
^{239}Pu	93.6%
^{240}Pu	5.9%
^{241}Pu	0.4%
^{242}Pu	0.1%
Uranium fuel characteristics	
^{234}U	0.002%
^{235}U	0.2%
^{236}U	0.001%
^{238}U	99.797%

In the early stages of this study, a burnup of 44.4 GWd/ton was assumed. This value was obtained from Table 2.1.1.1-3 of the Westinghouse report.[2] It is proposed to burn some assemblies for two cycles and some others for three cycles, and this 44.4 GWd/ton was considered representative of the average assembly. Reactivity worth of spent assemblies will be greater after two cycles of burnup rather than three. However, shielding considerations in the design of facilities must accommodate the maximum neutron source, which is achieved with the maximum higher actinide content and the maximum gamma source, which asymptotically increases with burnup. Therefore, it was decided to study assemblies following both two and three burnup cycles. The two-cycle case corresponds to a burnup of 35 GWd/ton, and for the three-cycle case the burnup is 52.5 GWd/ton.

In summary, this report discusses four distinct spent fuel assemblies:

1. two-cycle (35 GWd/ton) burnup of an assembly without WABAs,

2. two-cycle burnup of an assembly containing WABAs,

3. three-cycle (52.5 GWd/ton) burnup of an assembly without WABAs, and

4. three-cycle burnup of an assembly containing WABAs.

In reporting results, these four distinct fuel assembly arrangements will be referred to as cases 1, 2, 3, and 4, respectively.

3. SPENT FUEL CHARACTERISTICS AND COMPARISONS

If plutonium generated by the weapons program is to be disposed of by means of commercial reactors, an understanding of the characteristics of the spent fuel following the expected burnup is important, particularly for assurance that the plutonium in the spent fuel will be of nonweapons grade. In fact, it must be demonstrated that the spent fuel is isotopically similar to plutonium in discharged low-enriched uranium (LEU) assemblies. It should also be shown that the radiation field surrounding a spent MOX assembly is comparable to that surrounding a spent LEU assembly. In this section, the results of calculations to determine the isotopic composition of the discharged MOX fuel from the typical assembly will be discussed. Apart from the isotopic composition of the discharged plutonium, a detailed characterization of the spent fuel is needed to assess the hazards associated with handling, transportation, and storage. Results for a number of different quantities that define hazards associated with spent fuel will be presented below. However, before discussing these results, some details of the computational methods will be outlined.

3.1 COMPUTATIONAL METHODS

The results to be discussed were obtained by exercising the SAS2H, ORIGEN-S, and SAS1 codes. The SAS2H and SAS1 codes are actually control modules that invoke a sequence of codes in the SCALE system.[3] The SAS2H module estimates the burnup of a fuel assembly by determining assembly averaged neutron spectra based on the fuel and structural composition of the assembly and the assembly power level. It then determines depletion, ingrowth, and decay of nuclear material, and it does so for each reactor cycle. The ORIGEN-S code[5] is employed to follow the inventory of nuclear material in the assembly after discharge from the reactor (ORIGEN-S is also one of the codes employed in the SAS2H sequence). In the analyses following discharge, ORIGEN-S provides estimates of activities, decay heat, and hazards associated with the nuclear material. Using source terms calculated by ORIGEN-S, the SAS1[6] system was used to determine the dose rates from the assemblies (both shielded and unshielded) at various times following discharge. More specific details of the calculations are explained in the following paragraphs.

In the estimation of multigroup cross sections, SAS2H executes a two-step calculational process. The first step is the calculation of cell-weighted cross sections for a unit fuel-pin cell. In the second step, SAS2H uses these cell-weighted cross sections in a larger unit cell representative of assembly-weighted conditions. This larger cell is defined to represent the desired assembly design containing water holes, burnable poison rods, cladding material, fuel rods containing absorbers, etc. It models the actual assembly by defining an equivalent cylindrical representation. The time-dependent neutron flux spectra obtained from this larger (cylindrical) unit cell are used to determine the appropriate time-dependent cross sections for the burnup-dependent fuel compositions.

Two points are noteworthy concerning the two-step nature of these calculations. First, if the fuel itself were "smeared out" over the larger unit cell, resonance self-shielding effects could not be taken into account. Hence, the effective cross sections for the fuel rod are calculated in the first step, thereby accounting for resonance self-shielding. Second, the discrete-ordinates transport calculation assumes an azimuthal isotropy that is achieved via the spatial averaging. Thus, when XSDRNPM (see below) is referred to as being "one-dimensional" (1-D) this means that there is a variation in just one dimension (i.e., the radial direction).

SAS2H[4] is one of the Shielding Analysis Sequences (SAS) that was developed for the SCALE system. SAS2H calls three separate programs for the calculation of neutron cross sections:

- BONAMI, a program to perform resonance-shielding calculations through the application of the Bondarenko shielding factor method,
- NITAWL-II, a program that applies the Nordheim Integral Technique to perform neutron cross-section processing in the resonance energy range, and
- XSDRNPM, a discrete-ordinates transport code that is used in this sequence to produce cell-weighted cross sections.

BONAMI, NITAWL-II, and XSDRNPM are further explained in Ref. 3.

SAS2H calls ORIGEN-S so that the depletion and decay calculations can be carried out following library preparation. ORIGEN-S calculates the time evolution of the nuclide species in a typical assembly during burnup and following discharge. ORIGEN refers to the Oak Ridge Isotope GENeration and depletion code, and the S in ORIGEN-S indicates the version that is part of the SCALE system. ORIGEN-S calculates the time evolution of nuclide species at a typical point in an assembly, and it makes use of neutron-flux data, cross sections, and other neutronic data (e.g., fission product yields, decay rates, branching fractions) that are representative of the assembly and its nuclide inventory as a function of time (results from the XSDRNPM portion of the sequence). Thus, it simulates the depletion of fissionable species, the ingrowth of actinides and fission products, and the subsequent decay and ingrowth of daughter nuclides over time. ORIGEN-S can also determine decay-heat source strengths as well as radiation source spectra and strengths.

SAS2H iteratively calls the three cross-section modules and the ORIGEN module. This is necessary because the makeup of the assembly will vary with time, which in turn affects the neutron flux and thus the cross sections. Thus, cross-section libraries must be adjusted over the course of the burnup history. Following discharge from the reactor, the time evolution of the nuclide species (decay of spent fuel) is calculated by using ORIGEN-S in its stand-alone version.

SAS1 was employed to perform 1-D discrete-ordinates shielding calculations. SAS1 is similar to SAS2H in that it replaces the actual assembly with a cylindrically symmetric equivalent. The radiation source strength of the assembly at various times following discharge (which had been calculated with ORIGEN-S) was then dispersed throughout this equivalent assembly, and radiation dose rates in the vicinity of an assembly at various times following discharge were calculated.

The cross-section library used in the SAS2H calculations was the latest version of what is known as the 44-group ENDF/B-V library (distributed as part of the SCALE system). This 44-group library is specifically designed for the analysis of fresh and spent fuel and radioactive waste systems. For the shielding analysis performed with SAS1, the 27n-18g coupled library (based on ENDF/B-IV) was used (also a standard SCALE library). This library consists of 27 neutron groups and 18 gamma-ray groups. Both of these libraries are discussed in Sect. M4.2 of the SCALE documentation.

3.2 ISOTOPIC COMPOSITION OF SPENT FUEL

Characteristics of the spent fuel following discharge were calculated for a range of decay times; we have chosen a graphical presentation of results because that mode emphasizes trends and shows differences that exist between cases. In the plots that follow, the calculated data points are shown. The lines through the data points are included purely as an aid to visualization. The more detailed numerical results are presented in the tables in Appendix A.

Figures 2 and 3 show the plutonium content of an assembly vs time after discharge for cases 1 and 3. Both of these are non-WABA cases. With the absorbers present, the results are not very different; however, see the following discussions for more detailed comments. The plutonium results for all four cases are shown in detail in Tables A.1–A.4. The assembly was charged with plutonium that was high in ^{239}Pu

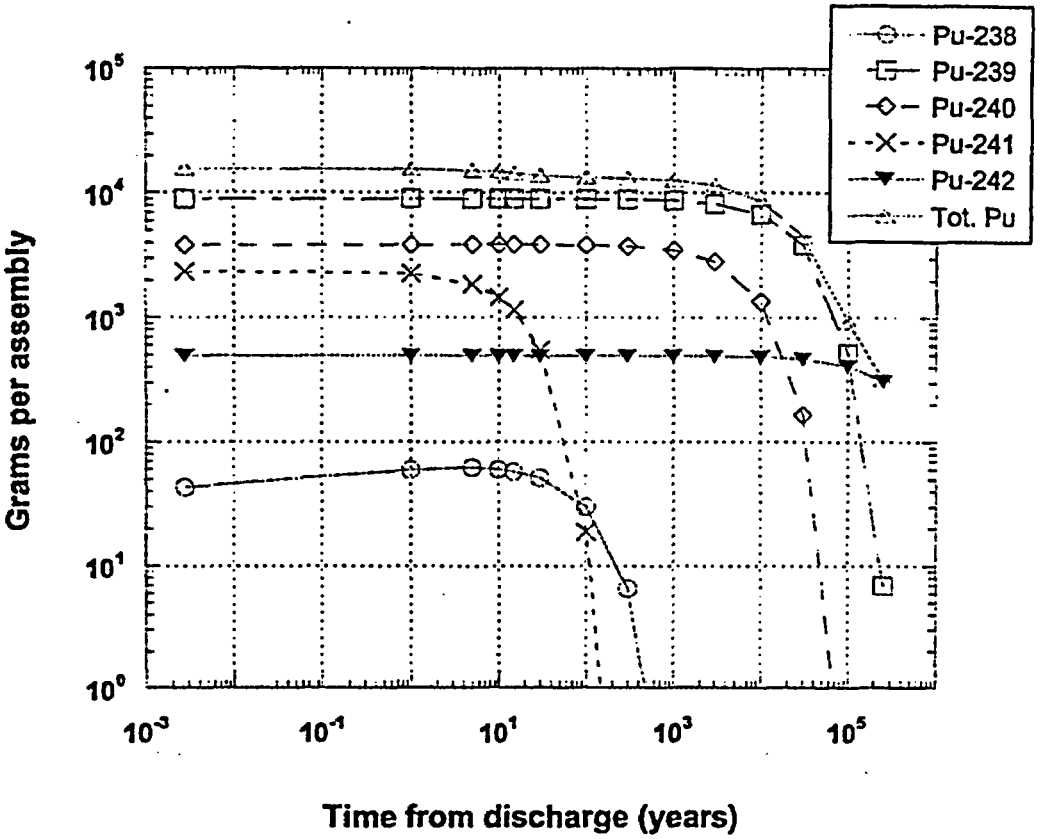

Fig. 2. Plutonium content of an assembly without wet annular bunable absorbers (WABAs) and burned for two reactor cycles.

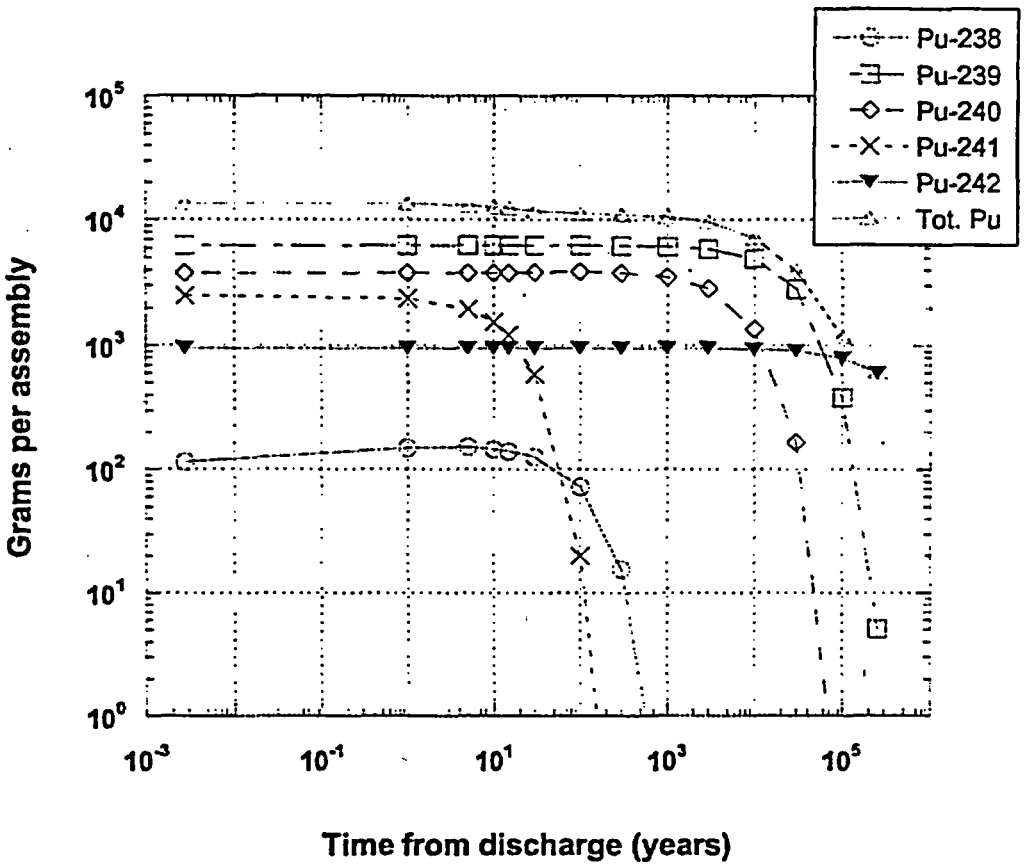

Fig. 3. Plutonium content of an assembly without WABAs and following three cycles.

(93.6%) with small amounts of ^{240}Pu, ^{241}Pu, and ^{242}Pu. Plutonium-239 is the principal fissile nuclide. It is depleted during burnup, and the concentrations grow for ^{240}Pu, ^{241}Pu, and ^{242}Pu. Some ^{238}Pu is also in evidence at discharge. Plutonium-238 was not present in the fresh fuel. It can arise by a succession of captures on ^{235}U and ^{236}U via ^{237}Np, but some of it comes from the decay of ^{242}Cm. The ^{239}Pu content at discharge is 57%, 61%, 46%, and 49% for cases 1, 2, 3, and 4, respectively. These compositions compare favorably with some values for LEU fuel. For instance, for LEU fuel following a burnup of 47.8 GWd/ton (Ref. 1), the ^{239}Pu is calculated to be 53.7 wt % of total plutonium. However, for the current calculations, note that the spectral impact of the WABAs is apparent in the higher ^{239}Pu content.

Concentration values for other actinides of interest are illustrated in Figs. 4 and 5 and again for cases 1 and 3, respectively. The detailed results for all cases are also contained in Tables A.1–A.4. There is significant burnup of the ^{235}U (although relative to the ^{239}Pu burnup it is small), and about 3 to 4% of the ^{238}U is also depleted. There is noticeable production of ^{236}U and ^{237}Np. The concentrations of these two nuclides and ^{235}U grow over time because of the decay of ^{240}Pu, ^{241}Am, and ^{239}Pu, respectively. Note that the ^{241}Am concentration increases following discharge because of the decay of the ^{241}Pu. Plutonium-241 has a half-life of 14.4 years. However, inspection of the calculated numbers shows that ^{241}Pu is present out to some thousands of years. This is because of the presence of ^{245}Cm.

In comparing concentrations following two- and three-cycle burnup, it can be seen that ^{235}U, ^{239}Pu, and ^{238}U are depleted more as burnup increases, and most of the other actinides have concentrations that are enhanced with greater burnup. We note, however, that the concentration of ^{244}Cm builds at a rate that far exceeds the increase in burnup. Because ^{244}Cm is a significant neutron source, this has important implications. In the burning of MOX fuel, a series of neutron-capture reactions on the plutonium isotopes eventually leads to ^{243}Pu, which beta decays to ^{243}Am, leading to ^{244}Am via neutron capture and then to ^{244}Cm via beta decay. In turn, neutron captures lead to the higher curium isotopes. The relative values of the capture cross sections in this chain will determine the final concentration distribution among the various species. However, not all of these cross sections may be well known. There is, of course, the issue of the availability of experimental cross-section data in the energy range of interest. But there is also the issue of being able to model the time-dependent neutron spectra during burnup and then using these to determine realistic one-group cross sections with the available data. Much more experience exists with uranium fuels on this latter point than is the case with MOX fuels.

Regarding the effect of the burnable absorbers, the results obtained with the WABAs are not greatly different from those without WABAs. However, with the absorbers present, a kilogram or more extra ^{239}Pu in the assembly is predicted at discharge (out of a total amount in the range of 6 to 10 kg). Inspection of the ^{238}U content indicates, however, that the extra ^{239}Pu comes from the conversion of more of the ^{238}U. Because there will always be a large amount of ^{238}U in MOX fuel, even small uncertainties in the ^{238}U capture cross section can affect the precision with which ^{239}Pu discharge concentrations can be calculated.

We note that some fuel cycles have been proposed in which the WABAs are removed from the assemblies after one or two cycles. Modeling of such a sequence is possible with the SCALE system, but has not been attempted.

3.3 ACTIVITY AND DECAY HEAT

Activity and the decay heat of the assembly are shown in Figs. 6–8. Specifically, activities are shown for the two- and three-cycle non-WABA cases in Figs. 6 and 7 (the WABA cases show only negligible differences). The decay heat results for the three-cycle non-WABA case are shown in Fig. 8. In many ways, these two quantities (activity and decay heat) behave in a similar manner. In the early stages following discharge, the fission products are the biggest contributors to activity and decay heat. At later

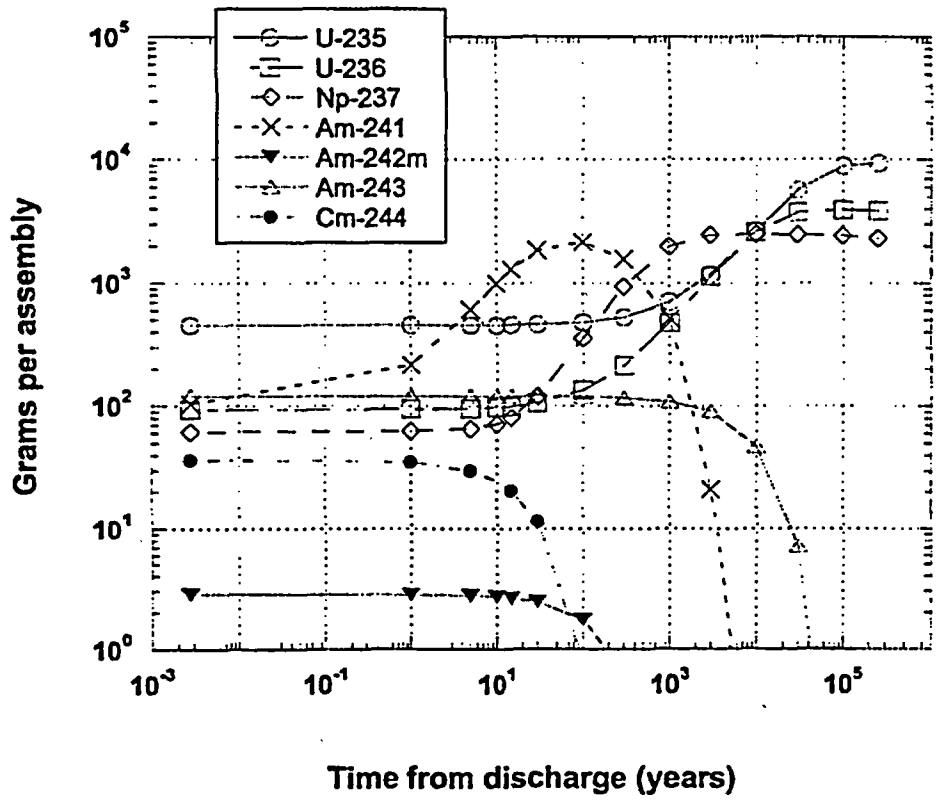

Fig. 4. Assembly inventories for some nonplutonium actinides following two cycles without WABAs.

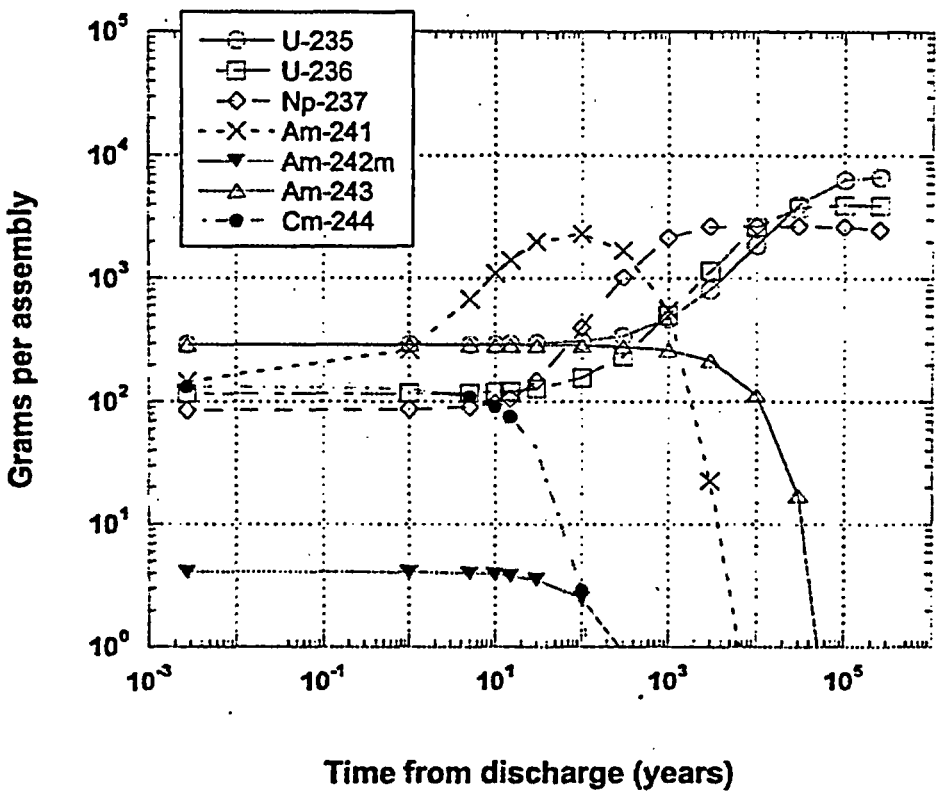

Fig. 5. Assembly inventories for some nonplutonium actinides following three cycles without WABAs.

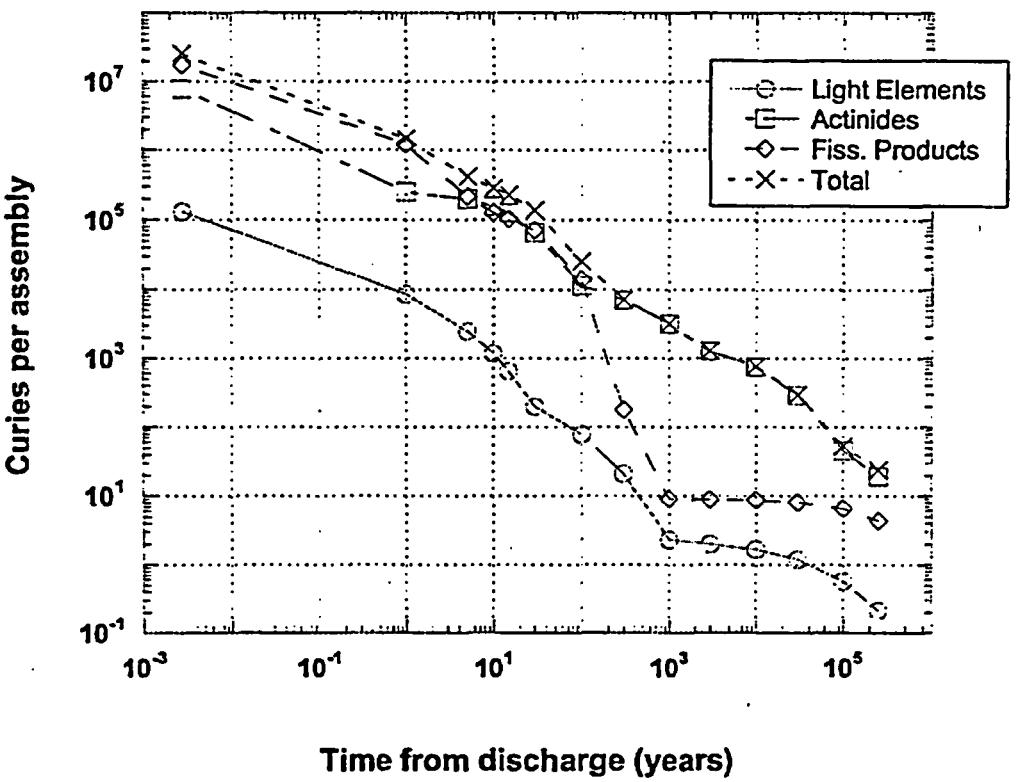

Fig. 6. Assembly activities following two cycles without WABAs.

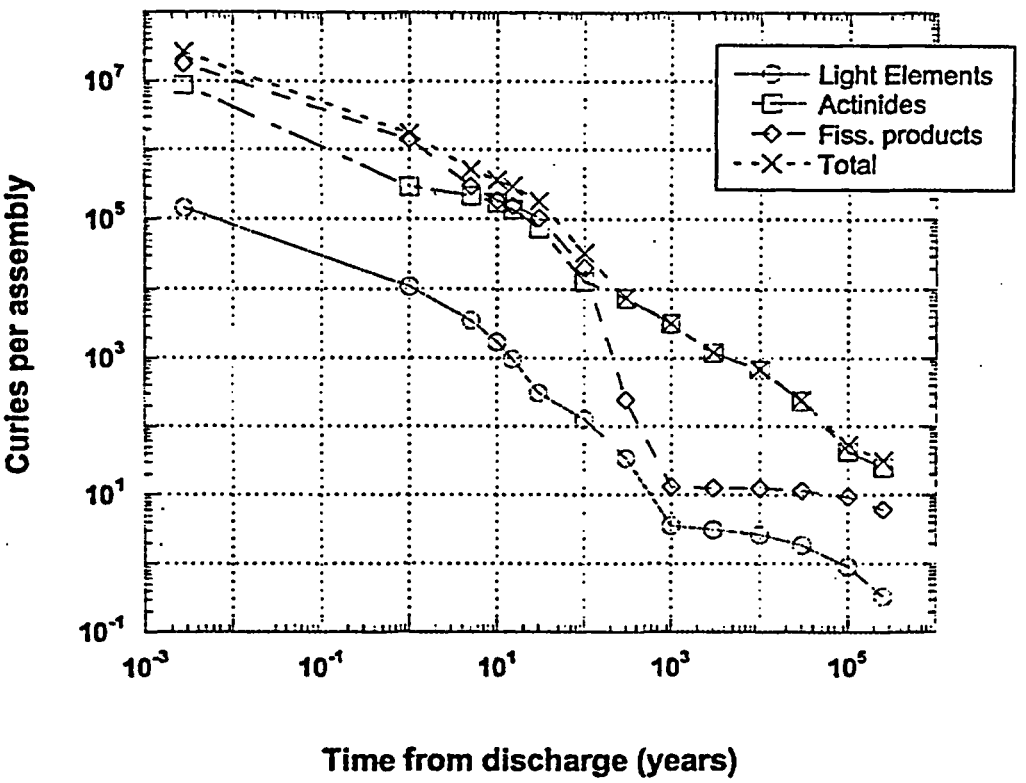

Fig. 7. Assembly activities following three cycles without WABAs.

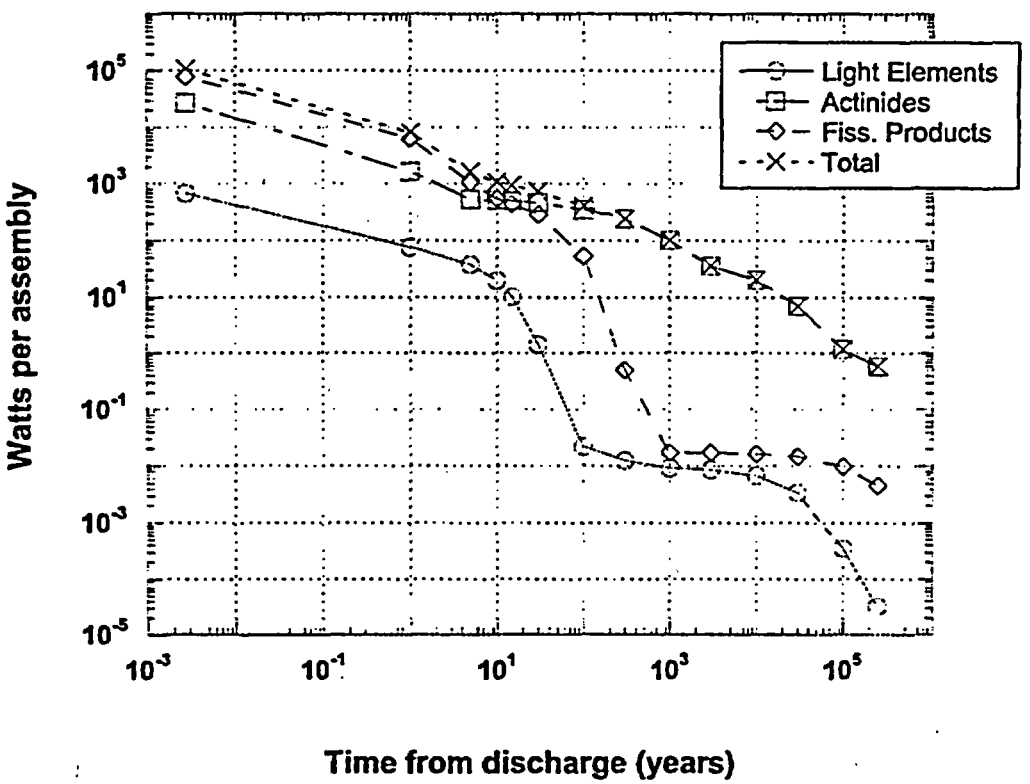

Fig. 8. Assembly decay-heat values folowing three cycles without WABAs.

times, the actinides, because of their generally greater half-lives, contribute most to both the activity and the decay heat. Activity, as shown, includes alpha, beta, and gamma decays. Therefore, it should be considered as a gross indicator of spent fuel radiological characteristics.

The contribution of the light elements to the activity (and decay heat) is rather small. The light elements are defined to be those elements contained in the structural materials of the assembly. In this study, their share of the overall material composition was chosen to be representative of a typical PWR. The most important contributor to the light-element activity is ^{60}Co, resulting from the irradiation of natural cobalt (100% ^{59}Co). Cobalt is a contaminant of nickel and is therefore present in structural materials containing alloys of nickel. Among different structures that nominally contain the same materials, the amount of cobalt can vary considerably. Cobalt-60 is a gamma emitter (1.17 and 1.33 MeV with a 5.3-year half-life), and it contributes significantly to the gamma source strength.

The fission-product activities are due to beta and gamma decays. Examples of important contributors are ^{90}Sr, ^{134}Cs, and ^{137}Cs. The actinides contribute to the alpha-decay activity. There are also some actinides that undergo spontaneous fission with subsequent neutron emission. Actual numerical values for calculated activities are shown in Tables A.5 and A.6.

3.4 GAMMA AND NEUTRON DOSE RATES

Gamma and neutron dose rates are of interest both for a bare assembly and for the exterior of a transportation cask containing spent fuel assemblies. Figures 9 and 10 show the gamma dose rates (at the surface and at 1 m from the surface) for an unshielded assembly following two and three reactor cycles, respectively. Figures 11 and 12 show the corresponding neutron dose rates (all of these dose-rate plots are for non-WABA cases). Comparing Figs. 9 and 10, one can see that the gamma dose rates scale roughly with burnup. This is as might be expected. However, comparisons between Figs. 11 and 12 do not show the same thing. The majority of the neutron flux comes from the spontaneous fissioning of ^{244}Cm (and also from ^{242}Cm in the early stages following discharge). The concentration of ^{244}Cm (as well as that of ^{242}Cm) tends to build with burnup in a manner that is not proportional to the amount of the burnup. Compare, for instance, the concentration of ^{244}Cm in Figs. 4 and 5 (or compare the values in Tables A.1 and A.3).

Gamma and neutron dose rates have also been calculated in the vicinity of a spent fuel storage cask. Specifically, for this work, the TN-24P spent fuel storage cask[7] was studied. This cask can store 24 PWR spent fuel assemblies. It consists of a forged steel body for structural integrity and gamma shielding that is surrounded by a resin layer for neutron shielding. Figure 13 shows the gamma and neutron dose rates at 1 m from the surface of a TN-24P cask that contained 24 assemblies following three reactor cycles. Dose rates for these cases are shown out to 10 years beyond discharge.

The detailed numerical results from the dose-rate calculations are shown in Tables A.7 through A.11. Tables A.7 and A.8 show gamma dose rates for an assembly that was burned for two and three cycles, respectively (with and without WABAs). Tables A.9 and A.10 show corresponding neutron dose rates. Table A.11 shows the external gamma and neutron dose rates for a transportation cask containing 24 spent fuel assemblies. For the case of a transportation cask, the assemblies have been burned for three cycles so that the maximum radiation levels can be taken into account.

3.5 SEVERE ACCIDENT ANALYSES

Design-basis accidents usually involve destruction of the fuel by some mechanism with consequent release to the environment of various fission products and actinides. One question of concern to the

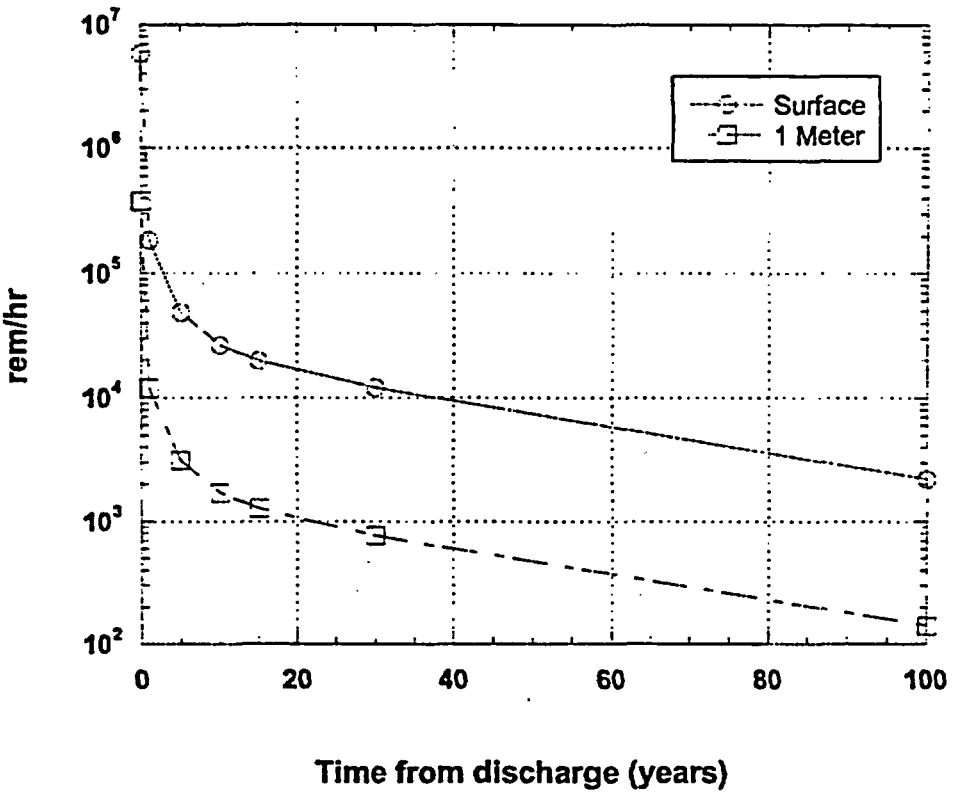

Fig. 9. Assembly gamma dose rates following two cycles without WABAs.

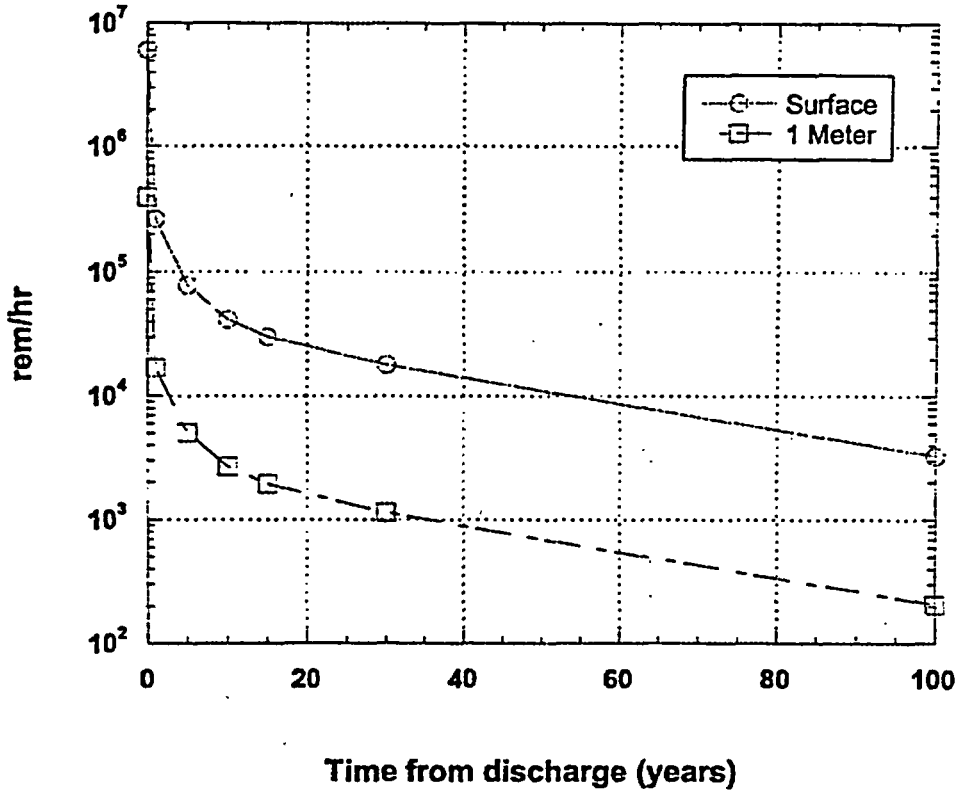

Fig. 10. Assembly gamma dose rates following three cycles without WABAs.

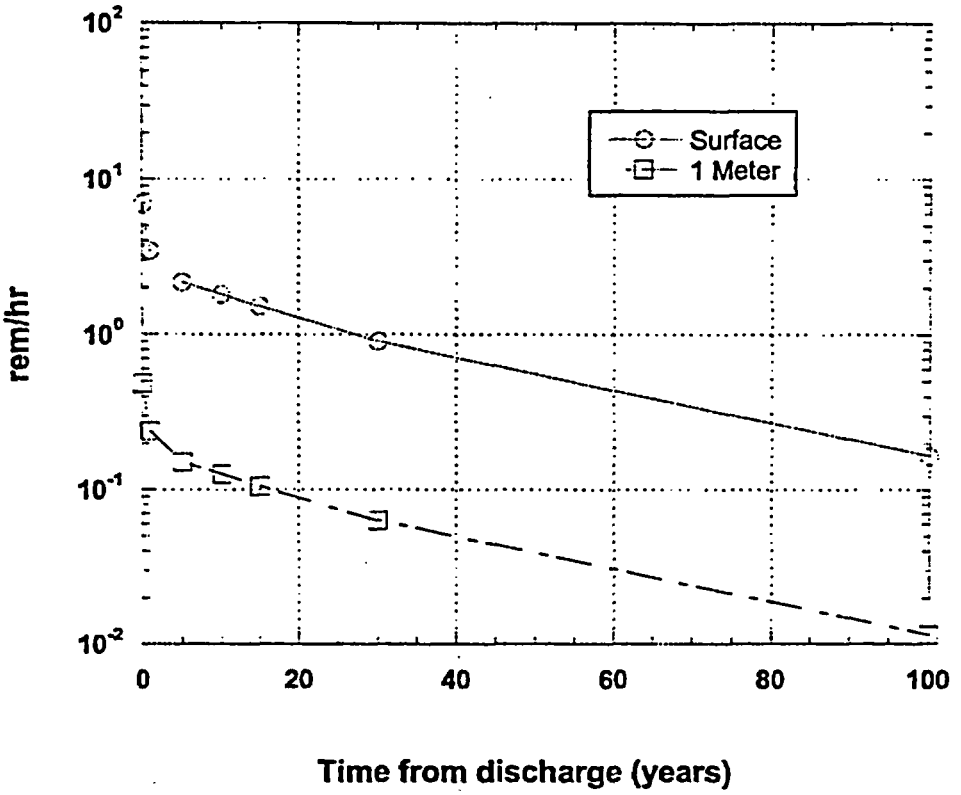

Fig. 11. Assembly neutron dose rates following two cycles.

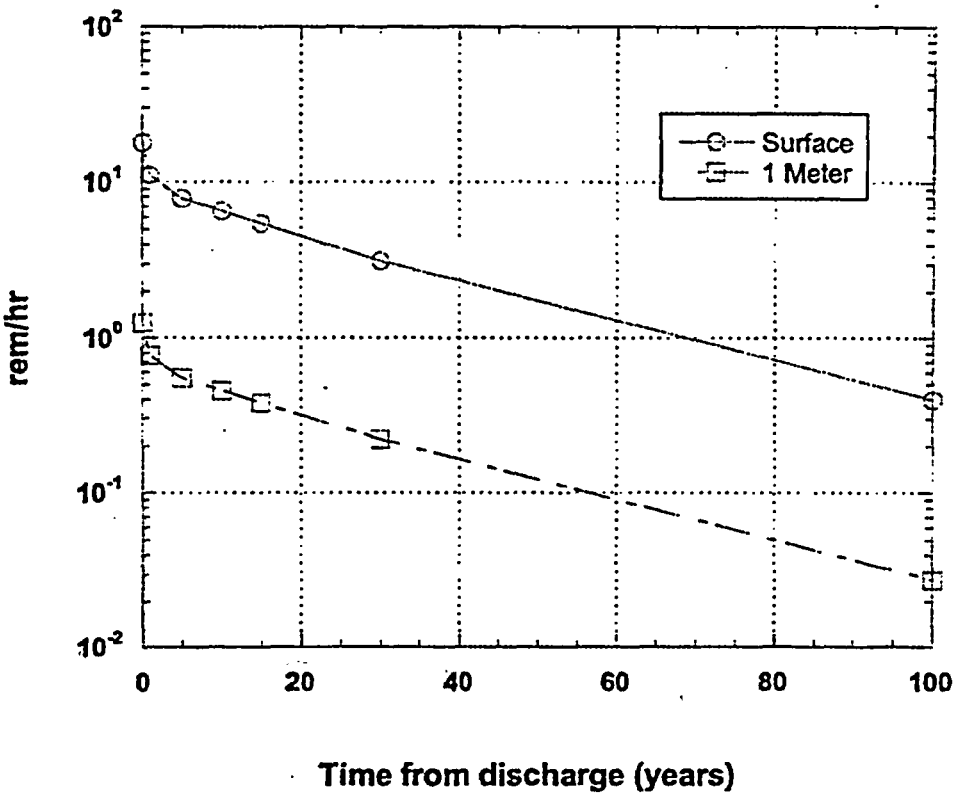

Fig. 12. Assembly neutron dose rates following three cycles.

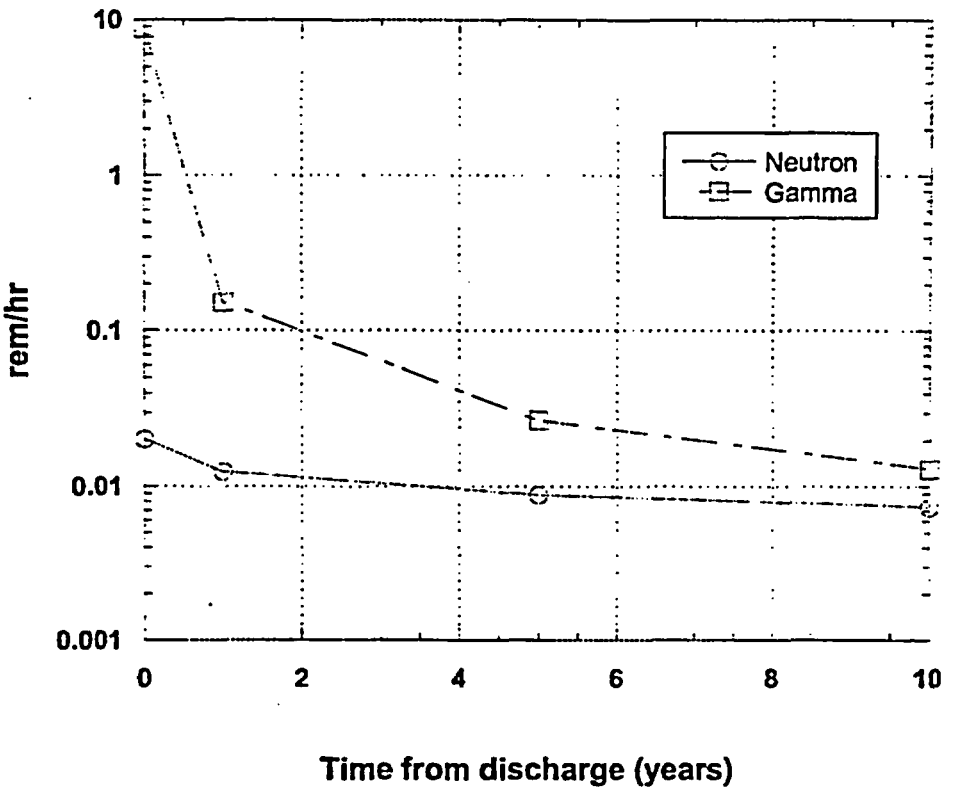

Fig. 13. Dose rates at 1 m from the surface of a transportation cask containing 24 spent fuel assemblies following three reactor cycles.

plutonium disposition program is whether the consequences of a severe accident with MOX fuel exceed the consequences of a similar accident with LEU fuel. While the analysis of such an accident is beyond the scope of these studies, input to severe accident studies can be obtained from ORIGEN analyses.

Selected nuclides commonly input to accident analyses are shown in Tables 2 and 3. Table 2 gives nuclide activities at discharge, and Table 3 contains activities at 10 years following discharge. In all cases, results are shown for an assembly following two reactor cycles both with and without burnable absorbers, and similarly for an assembly following three reactor cycles. The ratios of activities with and without WABAs are shown for the two- and three-cycle cases in Tables 2 and 3. Also shown are the ratios of three- to two-cycle concentrations (averaged between the WABA and non-WABA cases).

The nuclides in Tables 2 and 3 will generally tend to have short half-lives, and many of them have insignificant activities after 10 years. In the tables, the half-life values are listed. Since many of these nuclides result from the decay of other species, the half-life is not necessarily a good indication of activity at a future date, and some species with quite short half-lives are still in evidence after 10 years.

3.6 CRITICALITY SAFETY FOR GEOLOGIC REPOSITORY

Assessment of repository safety involves analyses covering time periods thousands of years into the future. While actinide concentrations have been discussed above, criticality analyses for a repository include selected nuclides that have significant absorption cross sections (i.e., they are good neutron poisons). Estimates of the quantities of these nuclides were obtained from ORIGEN calculations, and they are listed in Tables 4 and 5 for the time of discharge and for 10 years following discharge, respectively. In all cases, results are shown for an assembly following two reactor cycles with and without burnable absorbers, and similarly for an assembly following three reactor cycles. Ratios between WABA and non-WABA cases and three- to two-cycle ratios are again shown as they were in Tables 2 and 3.

3.7 MOX FUEL PIN RADIAL POWER PROFILE

A calculation was performed to determine the power (fission activity) density as a function of radial distance in a typical MOX fuel pin in the Westinghouse PWR. The calculation considers the situation with fresh fuel. The discrete-ordinates code XSDRNPM (that is part of the SCALE system) was employed to analyze a fuel pin in a cylindrically symmetric arrangement that simulated part of the PWR assembly. The calculated radial power profile is shown in Fig. 14.

The model used in exercising XSDRNPM consisted of a MOX fuel pin, coated with integral fuel burnable absorber, that was surrounded by clad and, in turn, surrounded by moderator. This represents one fuel-pin cell in the 17x17 lattice. Continuing with circular symmetry, this one fuel-pin cell was surrounded by an outer zone in which fuel, clad, and moderator were evenly mixed to represent their relative concentrations in the assembly as a whole. The radius of this outer zone was varied by XSDRNPM so as to produce a k_{eff} of unity. The ensuing neutron spectra and one-group cross sections were then used to determine the fission activity as a function of radial position in the MOX fuel rod.

XSDRNPM determined that a cylinder of 74.5 cm, which is equivalent to 47 fuel rods, would product a k_{eff} of unity. The Westinghouse assembly in question consists of 264 fuel rods; however, it also contains control rods, and this was not the case with the XSDRNPM assembly. The intent of the XSDRNPM simulation was to mimic a multirod part of the actual assembly. The fission activity radial profile was calculated by XSDRNPM and was normalized to the power level in the actual PWR assembly.

Normalization was effected by adjusting the linear power density resulting from the calculated radial profile so that it corresponded to the average value quoted in the Westinghouse specifications, i.e., 186.35 W/cm. An alternative normalization was also used, which relied on the fact that a specific power value of 39.77 MW/t applied to the MOX assembly as a whole. The results obtained via the two methods of normalization are within 2% of one another. This is to be expected since both approaches depend on values obtained from Westinghouse design specifications.

Table 2. Comparison of principal activity sources at discharge (Ci/assembly)

		Two cycles			Three cycles			
Nuclide	Half-life	No WABAs	With WABAs	With/ non-WABAs	No WABAs	With WABAs	With/ non-WABAs	Average ratio of three to two cycles
^{241}Am	432.7 years	3.57×10^{2}	3.85×10^{2}	1.08	4.96×10^{2}	5.50×10^{2}	1.11	1.41
^{140}Ba	12.75 d	8.28×10^{5}	8.29×10^{5}	1.00	8.27×10^{5}	8.27×10^{5}	1.00	1.00
^{14}C	5715 years	3.38×10^{-1}	3.35×10^{-1}	0.99	5.56×10^{-1}	5.37×10^{-1}	0.97	1.62
^{141}Ce	32.5 d	7.58×10^{5}	7.59×10^{5}	1.00	7.48×10^{5}	7.49×10^{5}	1.00	0.99
^{143}Ce	1.377 d	6.49×10^{5}	6.50×10^{5}	1.00	6.46×10^{5}	6.46×10^{5}	1.00	0.99
^{144}Ce	284.6 d	4.80×10^{5}	4.79×10^{5}	1.00	5.08×10^{5}	5.07×10^{5}	1.00	1.06
^{242}Cm	162.8 d	7.26×10^{4}	7.40×10^{4}	1.02	1.44×10^{5}	1.46×10^{5}	1.01	1.98
^{244}Cm	18.1 years	2.94×10^{3}	3.18×10^{3}	1.08	1.08×10^{4}	1.08×10^{4}	1.00	3.53
^{58}Co	70.88 d	6.53×10^{3}	6.80×10^{3}	1.04	6.59×10^{3}	6.64×10^{3}	1.01	0.99
^{60}Co	5.271 years	3.24×10^{3}	3.25×10^{3}	1.00	4.70×10^{3}	4.59×10^{3}	0.98	1.43
^{134}Cs	2.065 years	7.17×10^{4}	7.55×10^{4}	1.05	1.34×10^{5}	1.35×10^{5}	1.01	1.83
^{134m}Cs	2.90 h	1.96×10^{4}	2.02×10^{4}	1.03	2.92×10^{4}	2.88×10^{4}	0.99	1.46
^{136}Cs	13.16 d	4.12×10^{4}	4.76×10^{4}	1.16	5.80×10^{4}	6.34×10^{4}	1.09	1.37
^{137}Cs	30.07 years	5.35×10^{4}	5.32×10^{4}	0.99	7.86×10^{4}	7.84×10^{4}	1.00	1.47
^{3}H	12.32 years	2.65×10^{2}	2.64×10^{2}	1.00	3.80×10^{2}	3.79×10^{2}	1.00	1.43
^{131}I	8.02 d	5.10×10^{5}	5.13×10^{5}	1.01	4.96×10^{5}	4.99×10^{5}	1.01	0.97
^{132}I	2.28 h	7.32×10^{5}	7.37×10^{5}	1.01	7.15×10^{5}	7.19×10^{5}	1.01	0.98
^{133}I	20.8 h	9.82×10^{5}	9.85×10^{5}	1.00	9.71×10^{5}	9.73×10^{5}	1.00	0.99
^{135}I	6.57 h	9.45×10^{5}	9.47×10^{5}	1.00	9.43×10^{5}	9.44×10^{5}	1.00	1.00
^{85}Kr	10.76 years	2.60×10^{3}	2.59×10^{3}	1.00	3.63×10^{3}	3.63×10^{3}	1.00	1.40
^{85m}Kr	4.48 h	7.59×10^{4}	7.64×10^{4}	1.01	7.31×10^{4}	7.36×10^{4}	1.01	0.96
^{88}Kr	2.84 h	1.96×10^{5}	1.97×10^{5}	1.01	1.91×10^{5}	1.92×10^{5}	1.01	0.97
^{140}La	1.678 d	8.44×10^{5}	8.44×10^{5}	1.00	8.53×10^{5}	8.52×10^{5}	1.00	1.01
^{141}La	3.90 h	7.48×10^{5}	7.51×10^{5}	1.00	7.38×10^{5}	7.40×10^{5}	1.00	0.99
^{99}Mo	2.75 d	8.87×10^{5}	8.90×10^{5}	1.00	8.84×10^{5}	8.86×10^{5}	1.00	1.00
^{95}Nb	34.97 d	6.94×10^{5}	6.94×10^{5}	1.00	6.77×10^{5}	6.79×10^{5}	1.00	0.98
^{95m}Nb	3.61 d	7.87×10^{3}	7.87×10^{3}	1.00	7.68×10^{3}	7.70×10^{3}	1.00	0.98
^{147}Nd	10.98 d	3.09×10^{5}	3.09×10^{5}	1.00	3.11×10^{5}	3.11×10^{5}	1.00	1.01
^{238}Np	2.117 d	4.10×10^{4}	4.56×10^{4}	1.11	6.35×10^{4}	6.42×10^{4}	1.01	1.47
^{143}Pr	13.57 d	6.41×10^{5}	6.41×10^{5}	1.00	6.37×10^{5}	6.37×10^{5}	1.00	0.99
^{238}Pu	87.7 years	7.26×10^{2}	8.12×10^{2}	1.12	1.96×10^{3}	2.07×10^{3}	1.06	2.62

Table 2 (continued)

Nuclide	Half-life	Two cycles			Three cycles			Average ratio of three to two cycles
		No WABAs	With WABAs	With/ non-WABAs	No WABAs	With WABAs	With/ non-WABAs	
^{239}Pu	2.41 × 10^{4} years	5.56 × 10^{2}	6.52 × 10^{2}	1.17	3.88 × 10^{2}	4.55 × 10^{2}	1.17	0.70
^{240}Pu								
^{241}Pu	14.4 years	2.43 × 10^{5}	2.58 × 10^{5}	1.06	2.57 × 10^{5}	2.77 × 10^{5}	1.08	1.07
^{86}Rb	18.65 d	4.77 × 10^{2}	5.09 × 10^{2}	1.07	7.68 × 10^{2}	7.79 × 10^{2}	1.01	1.57
^{105}Rh	35.4 h	7.32 × 10^{5}	7.37 × 10^{5}	1.01	7.31 × 10^{5}	7.36 × 10^{5}	1.01	1.00
^{105m}Rh	40 s	2.18 × 10^{5}	2.19 × 10^{5}	1.00	2.20 × 10^{5}	2.20 × 10^{5}	1.00	1.01
^{103}Ru	39.27 d	9.70 × 10^{5}	9.71 × 10^{5}	1.00	9.57 × 10^{5}	9.59 × 10^{5}	1.00	0.99
^{105}Ru	4.44 h	7.69 × 10^{5}	7.70 × 10^{5}	1.00	7.75 × 10^{5}	7.75 × 10^{5}	1.00	1.01
^{106}Ru	1.02 years	4.85 × 10^{5}	4.80 × 10^{5}	0.99	5.45 × 10^{5}	5.42 × 10^{5}	0.99	1.13
^{127}Sb	3.84 d	5.62 × 10^{4}	5.67 × 10^{4}	1.01	5.31 × 10^{4}	5.37 × 10^{4}	1.01	0.95
^{129}Sb	4.4 h	1.80 × 10^{5}	1.81 × 10^{5}	1.01	1.71 × 10^{5}	1.73 × 10^{5}	1.01	0.95
^{89}Sr	50.52 d	2.59 × 10^{5}	2.60 × 10^{5}	1.00	2.51 × 10^{5}	2.51 × 10^{5}	1.00	0.97
^{90}Sr	28.78 years	1.88 × 10^{4}	1.89 × 10^{4}	1.01	2.73 × 10^{4}	2.73 × 10^{4}	1.00	1.45
^{91}Sr	9.5 h	3.70 × 10^{5}	3.73 × 10^{5}	1.01	3.61 × 10^{5}	3.62 × 10^{5}	1.00	0.97
^{92}Sr	2.71 h	4.37 × 10^{5}	4.40 × 10^{5}	1.01	4.27 × 10^{5}	4.28 × 10^{5}	1.00	0.97
^{99m}Tc	6.01 h	7.86 × 10^{5}	7.89 × 10^{5}	1.00	7.86 × 10^{5}	7.88 × 10^{5}	1.00	1.00
^{127}Te	9.4 h	5.60 × 10^{4}	5.64 × 10^{4}	1.01	5.29 × 10^{4}	5.35 × 10^{4}	1.01	0.94
^{127m}Te	109 d	9.85 × 10^{3}	9.83 × 10^{3}	100	9.23 × 10^{3}	9.31 × 10^{3}	1.01	0.94
^{129}Te	1.16 h	1.72 × 10^{5}	1.73 × 10^{5}	1.01	1.64 × 10^{5}	1.65 × 10^{5}	1.01	0.95
^{129m}Te	33.6 d	3.58 × 10^{4}	3.60 × 10^{4}	1.01	3.39 × 10^{4}	3.42 × 10^{4}	1.01	0.95
^{132}Te	3.2 d	7.13 × 10^{5}	7.18 × 10^{5}	1.01	6.98 × 10^{5}	7.01 × 10^{5}	1.00	0.98
^{131m}Xe	11.9 d	6.50 × 10^{3}	6.60 × 10^{3}	1.02	6.94 × 10^{3}	6.97 × 10^{3}	1.00	1.07
^{133}Xe	5.243 d	9.85 × 10^{5}	9.89 × 10^{5}	1.00	9.76 × 10^{5}	9.78 × 10^{5}	1.00	0.99
^{133m}Xe	2.19 d	3.23 × 10^{4}	3.25 × 10^{4}	1.01	3.18 × 10^{4}	3.19 × 10^{4}	1.00	0.98
^{135}Xe	9.10 h	4.68 × 10^{5}	5.39 × 10^{5}	1.15	3.94 × 10^{5}	4.43 × 10^{5}	1.12	0.84
^{135m}Xe	15.3 min	2.26 × 10^{5}	2.27 × 10^{5}	1.00	2.20 × 10^{5}	2.21 × 10^{5}	1.00	0.97
^{90}Y	2.67 d	1.93 × 10^{4}	1.93 × 10^{4}	1.00	2.81 × 10^{4}	2.81 × 10^{4}	1.00	1.46
^{91}Y	58.5 d	3.77 × 10^{5}	3.78 × 10^{5}	1.00	3.66 × 10^{5}	3.67 × 10^{5}	1.00	0.97
^{91m}Y	49.7 min	2.15 × 10^{5}	2.16 × 10^{5}	1.00	2.09 × 10^{5}	2.10 × 10^{5}	1.00	0.97
^{92}Y	3.54 h	4.39 × 10^{5}	4.42 × 10^{5}	1.01	4.29 × 10^{5}	4.30 × 10^{5}	1.00	0.98
^{93}Y	10.2 h	3.68 × 10^{5}	3.71 × 10^{5}	1.01	3.59 × 10^{5}	3.61 × 10^{5}	1.01	0.98
^{95}Zr	64.02 d	6.96 × 10^{5}	6.96 × 10^{5}	1.00	6.79 × 10^{5}	6.81 × 10^{5}	1.00	0.98
^{97}Zr	16.8 h	7.29 × 10^{5}	7.33 × 10^{5}	1.01	7.19 × 10^{5}	7.21 × 10^{5}	1.00	0.99

Table 3. Comparison of principal activity sources after 10 years (Ci/assembly)

Nuclide	Half-life	Two cycles			Three cycles			Average ratio of three to two cycles
		No WABAs	With WABAs	With/ non-WABAs	No WABAs	With WABAs	With/ non-WABAs	
^{241}Am	432.7 years	3.41×10^{3}	3.63×10^{3}	1.06	3.73×10^{3}	4.03×10^{3}	1.08	1.10
^{140}Ba	12.75 d	–	–	–	–	–	–	–
^{14}C	5715 years	3.37×10^{-1}	3.35×10^{-1}	0.99	5.55×10^{-1}	5.37×10^{-1}	0.97	1.63
^{141}Ce	32.5 d	–	–	–	–	–	–	–
^{143}Ce	1.377 d	–	–	–	–	–	–	–
^{144}Ce	284.6 d	6.65×10^{1}	6.63×10^{1}	1.00	7.04×10^{1}	7.02×10^{1}	1.00	1.06
^{242}Cm	162.8 d	2.32×10^{1}	2.75×10^{1}	1.19	3.36×10^{1}	3.96×10^{1}	1.18	1.44
^{244}Cm	18.1 years	2.01×10^{3}	2.17×10^{3}	1.08	7.35×10^{3}	7.37×10^{3}	1.00	3.52
^{58}Co	70.88 d	–	–	–	–	–	–	–
^{60}Co	5.271 years	8.71×10^{2}	8.73×10^{2}	1.00	1.26×10^{3}	1.23×10^{3}	0.98	1.43
^{134}Cs	2.065 years	2.49×10^{3}	2.62×10^{3}	1.05	4.63×10^{3}	4.68×10^{3}	1.01	1.82
^{134m}Cs	2.90 h	–	–	–	–	–	–	–
^{136}Cs	13.16 d	–	–	–	–	–	–	–
^{137}Cs	30.07 years	4.25×10^{4}	4.22×10^{4}	0.99	6.24×10^{4}	6.22×10^{4}	1.00	1.47
^{3}H	12.32 years	1.51×10^{2}	1.50×10^{2}	0.99	2.17×10^{2}	2.16×10^{2}	1.00	1.44
^{131}I	8.02 d	–	–	–	–	–	–	–
^{132}I	2.28 h	–	–	–	–	–	–	–
^{133}I	20.8 h	–	–	–	–	–	–	–
^{135}I	6.57 h	–	–	–	–	–	–	–
^{85}Kr	10.76 years	1.36×10^{3}	1.36×10^{3}	1.00	1.90×10^{3}	1.90×10^{3}	1.00	1.40
^{85m}Kr	4.48 h	–	–	–	–	–	–	–
^{88}Kr	2.84 h	–	–	–	–	–	–	–
^{140}La	1.678 d	–	–	–	–	–	–	–
^{141}La	3.90 h	–	–	–	–	–	–	–
^{99}Mo	2.75 d	–	–	–	–	–	–	–
^{95}Nb	34.97 d	–	–	–	–	–	–	–
^{95m}Nb	3.61 d	–	–	–	–	–	–	–
^{147}Nd	10.98 d	–	–	–	–	–	–	–
^{238}Np	2.117 d	1.27×10^{-1}	1.50×10^{-1}	1.18	1.84×10^{-1}	2.16×10^{-1}	1.17	1.44
^{143}Pr	13.57 d	–	–	–	–	–	–	–
^{238}Pu	87.7	1.02×10^{3}	1.11×10^{3}	1.09	2.50×10^{3}	2.61×10^{3}	1.04	2.40
^{239}Pu	2.41×10^{4} years	5.58×10^{2}	6.55×10^{2}	1.17	3.91×10^{2}	4.57×10^{2}	1.17	0.7

Table 3 (continued)

Nuclide	Half-life	Two cycles			Three cycles			Average ratio of three to two cycles
		No WABAs	With WABAs	With/ non-WABAs	No WABAs	With WABAs	With/ non-WABAs	
^{240}Pu	6.56×10^3 years	8.72×10^2	8.57×10^2	0.98	8.60×10^2	8.80×10^2	1.02	1.01
^{241}Pu	14.4 years	1.50×10^5	1.59×10^5	1.06	1.59×10^5	1.71×10^5	1.08	1.07
^{86}Rb	18.65 d	–	–	–	–	–	–	–
^{105}Rh	35.4 h	–	–	–	–	–	–	–
^{105m}Rh	40 s	–	–	–	–	–	–	–
^{103}Ru	39.27 d	–	–	–	–	–	–	–
^{105}Ru	4.44 h	–	–	–	–	–	–	–
^{106}Ru	1.02 years	5.34×10^2	5.28×10^2	0.99	6.00×10^2	5.96×10^2	0.99	1.13
^{127}Sb	3.84 d	–	–	–	–	–	–	–
^{129}Sb	4.4 h	–	–	–	–	–	–	–
^{89}Sr	50.52 d	–	–	–	–	–	–	–
^{90}Sr	28.78 years	1.47×10^4	1.47×10^4	1.00	2.13×10^4	2.13×10^4	1.00	1.45
^{91}Sr	9.5 h	–	–	–	–	–	–	–
^{92}Sr	2.71 h	–	–	–	–	–	–	–
^{99m}Tc	6.01 h	–	–	–	–	–	–	–
^{127}Te	9.4 h	8.19×10^{-7}	8.18×10^{-7}	1.00	7.68×10^{-7}	7.75×10^{-7}	1.01	0.94
^{127m}Te	109 d	8.36×10^{-7}	8.35×10^{-7}	1.00	7.84×10^{-7}	7.91×10^{-7}	1.01	0.94
^{129}Te	1.16 h	–	–	–	–	–	–	–
^{129m}Te	33.6 d	–	–	–	–	–	–	–
^{132}Te	3.2 d	–	–	–	–	–	–	–
^{131m}Xe	11.9 d	–	–	–	–	–	–	–
^{133}Xe	5.243 d	–	–	–	–	–	–	–
^{133m}Xe	2.19 d	–	–	–	–	–	–	–
^{135}Xe	9.10 h	–	–	–	–	–	–	–
^{135m}Xe	15.3 min	–	–	–	–	–	–	–
^{90}Y	2.67 d	1.47×10^4	1.47×10^4	1.00	2.13×10^4	2.13×10^4	1.00	1.45
^{91}Y	58.5 d	–	–	–	–	–	–	–
^{91m}Y	49.7 min	–	–	–	–	–	–	–
^{92}Y	3.54 h	–	–	–	–	–	–	–
^{93}Y	10.2 h	–	–	–	–	–	–	–
^{95}Zr	64.02 d	–	–	–	–	–	–	–
^{97}Zr	16.8 h	–	–	–	–	–	–	–

Table 4. Nuclides of interest for criticality safety (g/assembly at discharge)

	Two cycles			Three cycles			
Nuclide	No WABAs	With WABAs	With/ non-WABAs	No WABAs	With WABAs	With/ non-WABAs	Average ratio of three to two cycles
^{16}O	6.20×10^{4}	6.20×10^{4}	1.00	6.20×10^{4}	6.20×10^{4}	1.00	1.00
^{95}Mo	2.57×10^{2}	2.54×10^{2}	0.99	3.93×10^{2}	3.91×10^{2}	0.99	1.53
^{101}Ru	3.97×10^{2}	3.93×10^{2}	0.99	5.81×10^{2}	5.77×10^{2}	0.99	1.47
^{99}Tc	3.73×10^{2}	3.68×10^{2}	0.99	5.26×10^{2}	5.22×10^{2}	0.99	1.41
^{103}Rh	3.30×10^{2}	3.26×10^{2}	0.99	4.24×10^{2}	4.25×10^{2}	1.00	1.29
^{109}Ag	9.27×10^{1}	8.83×10^{1}	0.95	1.19×10^{2}	1.16×10^{2}	0.97	1.30
^{143}Nd	3.78×10^{2}	3.81×10^{2}	1.01	5.16×10^{2}	5.26×10^{2}	1.02	1.37
^{145}Nd	2.83×10^{2}	2.80×10^{2}	0.99	4.02×10^{2}	4.01×10^{2}	1.00	1.43
^{147}Sm	3.21×10^{1}	3.01×10^{1}	0.94	5.04×10^{1}	4.85×10^{1}	0.96	1.59
^{149}Sm	3.22×10^{0}	3.91×10^{0}	1.21	2.69×10^{0}	3.20×10^{0}	1.19	0.83
^{150}Sm	1.58×10^{2}	1.61×10^{2}	1.02	2.35×10^{2}	2.38×10^{2}	1.01	1.48
^{151}Sm	1.76×10^{1}	2.08×10^{1}	1.18	1.84×10^{1}	2.10×10^{1}	1.14	1.03
^{152}Sm	7.74×10^{1}	7.38×10^{1}	0.95	1.02×10^{2}	1.01×10^{2}	0.99	1.34
^{151}Eu	2.98×10^{-2}	3.62×10^{-2}	1.21	2.64×10^{-2}	3.23×10^{-2}	1.22	0.89
^{153}Eu	7.33×10^{1}	7.23×10^{1}	0.99	1.15×10^{2}	1.13×10^{2}	0.98	1.57
^{155}Gd	6.92×10^{-2}	8.95×10^{-2}	1.29	1.18×10^{-1}	1.42×10^{-1}	1.20	1.64
^{234}U	5.56×10^{0}	5.34×10^{0}	0.96	4.93×10^{0}	4.86×10^{0}	0.99	0.90
^{235}U	4.53×10^{2}	4.65×10^{2}	1.03	2.92×10^{2}	3.10×10^{2}	1.06	0.66
^{236}U	9.38×10^{1}	9.62×10^{1}	1.03	1.16×10^{2}	1.18×10^{2}	1.02	1.23
^{238}U	4.29×10^{5}	4.27×10^{5}	1.00	4.22×10^{5}	4.21×10^{5}	1.00	0.98
^{237}Np	6.08×10^{1}	6.79×10^{1}	1.12	8.44×10^{1}	8.84×10^{1}	1.05	1.34
^{238}Pu	4.24×10^{1}	4.74×10^{1}	1.12	1.14×10^{2}	1.21×10^{2}	1.06	2.62
^{239}Pu	8.95×10^{3}	1.05×10^{4}	1.17	6.25×10^{3}	7.32×10^{3}	1.17	0.70
^{240}Pu	3.83×10^{3}	3.77×10^{3}	0.98	3.75×10^{3}	3.84×10^{3}	1.02	1.00
^{241}Pu	2.35×10^{3}	2.50×10^{3}	1.06	2.49×10^{3}	2.68×10^{3}	1.08	1.07
^{242}Pu	4.89×10^{2}	4.58×10^{2}	0.94	9.52×10^{2}	8.98×10^{2}	0.94	1.95
^{241}Am	1.04×10^{2}	1.12×10^{2}	1.08	1.45×10^{2}	1.60×10^{2}	1.10	1.41
^{242m}Am	2.83×10^{0}	3.34×10^{0}	1.18	4.09×10^{0}	4.81×10^{0}	1.18	1.44
^{243}Am	1.20×10^{2}	1.20×10^{2}	1.00	2.88×10^{2}	2.82×10^{2}	0.98	2.38
^{245}Cm	2.12×10^{0}	2.62×10^{0}	1.24	9.99×10^{0}	1.09×10^{1}	1.09	4.41

Table 5. Nuclides of interest for criticality safety (g/assembly at 10 y following discharge)

	Two cycles			Three cycles			
Nuclide	No WABAs	With WABAs	With/ non-WABAs	No WABAs	With WABAs	With/ non-WABAs	Average ratio of three to two cycles
^{16}O	6.20×10^{4}	6.20×10^{4}	1.00	6.20×10^{4}	6.20×10^{4}	1.00	1.00
^{95}Mo	3.07×10^{2}	3.04×10^{2}	0.99	4.41×10^{2}	4.40×10^{2}	1.00	1.44
^{101}Ru	3.97×10^{2}	3.93×10^{2}	0.99	5.81×10^{2}	5.77×10^{2}	0.99	1.47
^{99}Tc	3.75×10^{2}	3.70×10^{2}	0.99	5.28×10^{2}	5.24×10^{2}	0.99	1.41
^{103}Rh	3.60×10^{2}	3.56×10^{2}	0.99	4.53×10^{2}	4.55×10^{2}	1.00	1.27
^{109}Ag	9.28×10^{1}	8.84×10^{1}	0.95	1.19×10^{2}	1.17×10^{2}	0.98	1.30
^{143}Nd	3.88×10^{2}	3.91×10^{2}	1.01	5.26×10^{2}	5.37×10^{2}	1.02	1.36
^{145}Nd	2.83×10^{2}	2.81×10^{2}	0.99	4.03×10^{2}	4.01×10^{2}	1.00	1.43
^{147}Sm	1.14×10^{2}	1.09×10^{2}	0.96	1.36×10^{2}	1.34×10^{2}	0.99	1.21
^{149}Sm	4.00×10^{0}	4.70×10^{0}	1.18	3.52×10^{0}	4.03×10^{0}	1.14	0.87
^{150}Sm	1.58×10^{2}	1.61×10^{2}	1.02	2.35×10^{2}	2.38×10^{2}	1.01	1.48
^{151}Sm	1.65×10^{1}	1.94×10^{1}	1.18	1.72×10^{1}	1.96×10^{1}	1.14	1.03
^{152}Sm	7.75×10^{1}	7.38×10^{1}	0.95	1.02×10^{2}	1.01×10^{2}	0.99	1.34
^{151}Eu	1.33×10^{0}	1.59×10^{0}	1.20	1.40×10^{0}	1.60×10^{0}	1.14	1.03
^{153}Eu	7.39×10^{1}	7.29×10^{1}	0.99	1.15×10^{2}	1.14×10^{2}	0.99	1.56
^{155}Gd	2.85×10^{0}	2.92×10^{0}	1.02	4.99×10^{0}	5.03×10^{0}	1.01	1.74
^{234}U	1.03×10^{1}	1.04×10^{1}	1.01	1.65×10^{1}	1.70×10^{1}	1.03	1.62
^{235}U	4.55×10^{2}	4.68×10^{2}	1.03	2.94×10^{2}	3.12×10^{2}	1.06	0.66
^{236}U	9.78×10^{1}	1.00×10^{2}	1.02	1.20×10^{2}	1.22×10^{2}	1.02	1.22
^{238}U	4.29×10^{5}	4.27×10^{5}	1.00	4.22×10^{5}	4.21×10^{5}	1.00	0.98
^{237}Np	7.12×10^{1}	7.90×10^{1}	1.11	9.61×10^{1}	1.01×10^{2}	1.05	1.31
^{238}Pu	5.96×10^{1}	6.46×10^{1}	1.08	1.46×10^{2}	1.52×10^{2}	1.04	2.40
^{239}Pu	8.99×10^{3}	1.06×10^{4}	1.18	6.29×10^{3}	7.37×10^{3}	1.17	0.70
^{240}Pu	3.84×10^{3}	3.78×10^{3}	0.98	3.79×10^{3}	3.87×10^{3}	1.02	1.01
^{241}Pu	1.45×10^{3}	1.54×10^{3}	1.06	1.54×10^{3}	1.65×10^{3}	1.07	1.07
^{242}Pu	4.89×10^{2}	4.58×10^{2}	0.94	9.52×10^{2}	8.98×10^{2}	0.94	1.95
^{241}Am	9.95×10^{2}	1.06×10^{3}	1.07	1.09×10^{3}	1.17×10^{3}	1.07	1.10
^{242m}Am	2.69×10^{0}	3.18×10^{0}	1.18	3.89×10^{0}	4.58×10^{0}	1.18	1.44
^{243}Am	1.20×10^{2}	1.20×10^{2}	1.00	2.88×10^{2}	2.82×10^{2}	0.98	2.38
^{245}Cm	2.12×10^{0}	2.61×10^{0}	1.23	9.98×10^{0}	1.09×10^{1}	1.09	4.41

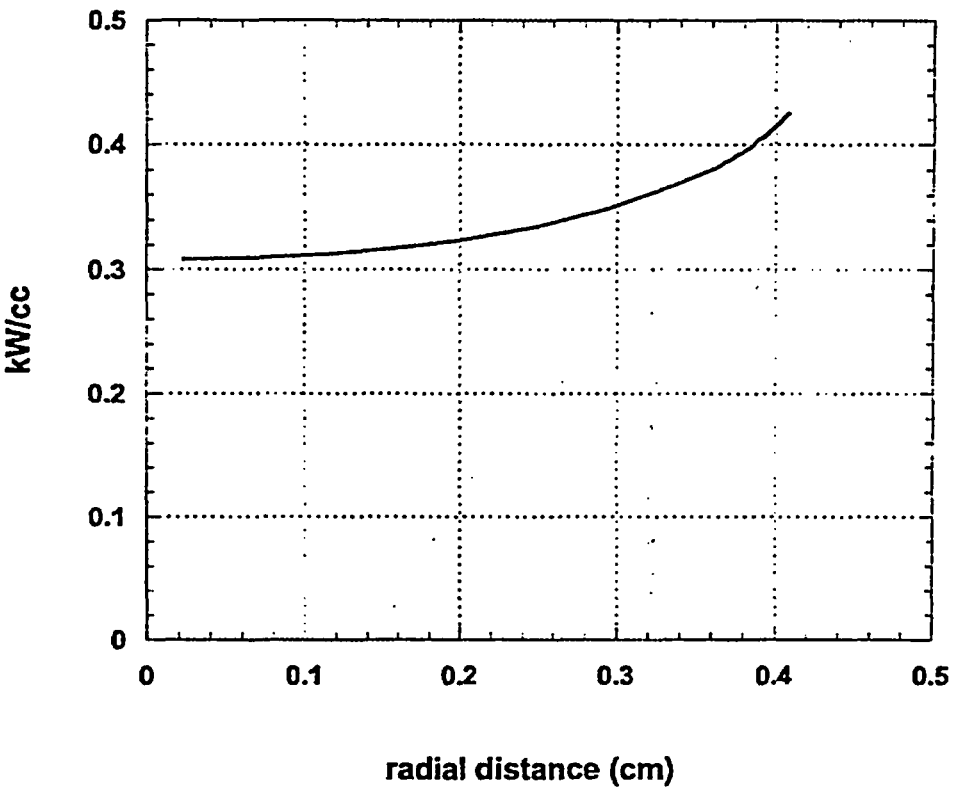

Fig. 14. Calculated radial power profile for a MOX fuel pin.

4. REFERENCES

1. B. D. Murphy, *Characteristics of Spent Mixed-Oxide Fuel From Light-Water Reactors, Vol. 1: The Combustion Engineering System 80+ Pressurized-Water-Reactor Design*, ORNL/TM-13170/V1, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., June 1996.

2. *Plutonium Disposition in Existing Pressurized Water Reactors*, DE-AC03-93SF19683, Westinghouse Electric Corporation, Pittsburgh, PA, June 1, 1994.

3. *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, Vols. I–III, NUREG/CR-0200, Rev. 4 (ORNL/NUREG/CSD-2/R4), Lockheed Martin Energy Systems, Inc., Oak Ridge Natl. Lab., April 1995. Available from Radiation Shielding Information Center as CCC-545.

4. O. W. Hermann and C. V. Parks, "SAS2H: A Coupled One-Dimensional Depletion and Shielding Analysis Code," Sect. S5 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, Vols. I–III, NUREG/CR-0200, Rev. 4 (ORNL/NUREG/CSD-2/R4), Lockheed Martin Energy Systems, Inc., Oak Ridge Natl. Lab., April 1995. Available from Radiation Shielding Information Center as CCC-545.

5. O. W. Hermann and R. M. Westfall, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," Sect. F7 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, Vols. I–III, NUREG/CR-0200, Rev. 4 (ORNL/NUREG/CSD-2/R4), Lockheed Martin Energy Systems, Inc., Oak Ridge Natl. Lab., April 1995. Available from Radiation Shielding Information Center as CCC-545.

6. J. R. Knight, C. V. Parks, S. M. Bowman, L. M. Petrie, and J. A. Bucholz, "SAS1: A One-Dimensional Shielding Analysis Module," Sect. S1 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, Vols. I–III, NUREG/CR-0200, Rev. 4 (ORNL/NUREG/CSD-2/R4), Lockheed Martin Energy Systems, Inc., Oak Ridge Natl. Lab., April 1995. Available from Radiation Shielding Information Center as CCC-545.

7. B. L. Broadhead, J. S. Tang, R. L. Childs, C. V. Parks, and H. Taniuchi, "Evaluation of Shielding Analysis Methods in Spent-Fuel Cask Environments," *Nucl. Tech.*, 117, 206-22 (1997).

APPENDIX A

DETAILED CALCULATED RESULTS

This appendix contains the calculated numerical values corresponding to the graphical results presented in the main body of this report. The tables that follow show calculated values for actinide concentrations, activities and decay heat, and gamma dose and neutron dose rates for unshielded assemblies as well as for the exterior of transportation casks containing spent fuel assemblies. The tables cover a variety of cooling times, they refer to assemblies burned for two and three reactor cycles, and they show the effects of burnable absorbers in all cases where these effects are nontrivial. The conditions that are pertinent to the particular results reported in Tables A.1–A.11 are explained in the table captions.

Table A.1. Actinide concentrations from MOX fuel following two cycles in Westinghouse PWR (fuel did not contain burnable absorbers)

	Plutonium per assembly (g)					
	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	Total plutonium
Charge	0.00×10^{0}	1.97×10^{4}	1.24×10^{3}	8.42×10^{1}	2.11×10^{1}	2.10×10^{4}
Discharge	4.25×10^{1}	8.95×10^{3}	3.83×10^{3}	2.35×10^{3}	4.89×10^{2}	1.57×10^{4}
1 d	4.27×10^{1}	8.96×10^{3}	3.83×10^{3}	2.35×10^{3}	4.89×10^{2}	1.57×10^{4}
1 year	5.94×10^{1}	8.99×10^{3}	3.83×10^{3}	2.24×10^{3}	4.89×10^{2}	1.56×10^{4}
5 years	6.20×10^{1}	8.99×10^{3}	3.84×10^{3}	1.85×10^{3}	4.89×10^{2}	1.52×10^{4}
10 years	5.97×10^{1}	8.99×10^{3}	3.84×10^{3}	1.45×10^{3}	4.89×10^{2}	1.48×10^{4}
15 years	5.74×10^{1}	8.99×10^{3}	3.84×10^{3}	1.14×10^{3}	4.89×10^{2}	1.45×10^{4}
30 years	5.11×10^{1}	8.99×10^{3}	3.85×10^{3}	5.51×10^{2}	4.89×10^{2}	1.39×10^{4}
100 years	2.99×10^{1}	8.97×10^{3}	3.83×10^{3}	1.87×10^{1}	4.90×10^{2}	1.33×10^{4}
300 years	6.53×10^{0}	8.92×10^{3}	3.75×10^{3}	4.23×10^{-3}	4.90×10^{2}	1.32×10^{4}
1000 years	5.02×10^{-2}	8.75×10^{3}	3.48×10^{3}	2.86×10^{-3}	4.89×10^{2}	1.27×10^{4}
3000 years	1.50×10^{-6}	8.28×10^{3}	2.82×10^{3}	2.43×10^{-3}	4.87×10^{2}	1.16×10^{4}
10,000 years	–	6.81×10^{3}	1.35×10^{3}	1.38×10^{-3}	4.81×10^{2}	8.64×10^{3}
30,000 years	–	3.86×10^{3}	1.63×10^{2}	2.69×10^{-4}	4.64×10^{2}	4.49×10^{3}
100,000 years	–	5.17×10^{2}	1.00×10^{-1}	8.92×10^{-7}	4.07×10^{2}	9.24×10^{2}
250,000 years	–	6.92×10^{0}	2.24×10^{-8}	–	3.08×10^{2}	3.15×10^{2}

	Other actinides per assembly (g)							
	^{235}U	^{236}U	^{238}U	^{237}Np	^{241}Am	^{242m}Am	^{243}Am	^{244}Cm
Charge	8.81×10^{2}	4.41×10^{0}	4.40×10^{5}	0.0	0.0	0.0	0.0	0.0
Discharge	4.53×10^{2}	9.38×10^{1}	4.29×10^{5}	6.11×10^{1}	1.04×10^{2}	2.83×10^{0}	1.20×10^{2}	3.63×10^{1}
1 d	4.53×10^{2}	9.38×10^{1}	4.29×10^{5}	6.12×10^{1}	1.04×10^{2}	2.83×10^{0}	1.20×10^{2}	3.64×10^{1}
1 year	4.53×10^{2}	9.42×10^{1}	4.29×10^{5}	6.25×10^{1}	2.15×10^{2}	2.81×10^{0}	1.20×10^{2}	3.50×10^{1}
5 years	4.54×10^{2}	9.58×10^{1}	4.29×10^{5}	6.52×10^{1}	6.05×10^{2}	2.76×10^{0}	1.20×10^{2}	3.01×10^{1}
10 years	4.55×10^{2}	9.78×10^{1}	4.29×10^{5}	7.15×10^{1}	9.95×10^{2}	2.69×10^{0}	1.20×10^{2}	2.48×10^{1}
15 years	4.57×10^{2}	9.98×10^{1}	4.29×10^{5}	8.06×10^{1}	1.30×10^{3}	2.63×10^{0}	1.20×10^{2}	2.05×10^{1}
30 years	4.60×10^{2}	1.06×10^{2}	4.29×10^{5}	1.19×10^{2}	1.84×10^{3}	2.44×10^{0}	1.20×10^{2}	1.15×10^{1}
100 years	4.78×10^{2}	1.34×10^{2}	4.29×10^{5}	3.53×10^{2}	2.14×10^{3}	1.73×10^{0}	1.19×10^{2}	7.91×10^{-1}
300 years	5.29×10^{2}	2.12×10^{2}	4.29×10^{5}	9.35×10^{2}	1.57×10^{3}	6.47×10^{-1}	1.17×10^{2}	3.73×10^{-4}
1000 years	7.04×10^{2}	4.75×10^{2}	4.29×10^{5}	1.97×10^{3}	5.10×10^{2}	2.07×10^{-2}	1.09×10^{2}	–
3000 years	1.18×10^{3}	1.13×10^{3}	4.29×10^{5}	2.45×10^{3}	2.08×10^{1}	1.11×10^{-6}	9.05×10^{1}	–
10,000 years	2.67×10^{3}	2.57×10^{3}	4.29×10^{5}	2.47×10^{3}	4.17×10^{-2}	–	4.68×10^{1}	–
30,000 years	5.61×10^{3}	3.74×10^{3}	4.29×10^{5}	2.45×10^{3}	8.11×10^{-3}	–	7.14×10^{0}	–
100,000 years	8.90×10^{3}	3.89×10^{3}	4.29×10^{5}	2.40×10^{3}	2.69×10^{-5}	–	9.88×10^{-3}	–
250,000 years	9.40×10^{3}	3.87×10^{3}	4.29×10^{5}	2.29×10^{3}	1.38×10^{-10}	–	1.38×10^{-6}	–

Table A.2. Actinide concentrations from MOX fuel following two cycles in Westinghouse PWR (fuel did contain burnable absorbers).

	Plutonium per assembly (g)					
	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	Total Plutonium
Charge	0.00×10^{0}	1.97×10^{4}	1.24×10^{3}	8.42×10^{1}	2.11×10^{1}	2.10×10^{4}
Discharge	4.76×10^{1}	1.05×10^{4}	3.77×10^{3}	2.50×10^{3}	4.58×10^{2}	1.73×10^{4}
1 d	4.77×10^{1}	1.05×10^{4}	3.77×10^{3}	2.50×10^{3}	4.58×10^{2}	1.73×10^{4}
1 year	6.47×10^{1}	1.06×10^{4}	3.77×10^{3}	2.38×10^{3}	4.58×10^{2}	1.73×10^{4}
5 years	6.73×10^{1}	1.06×10^{4}	3.77×10^{3}	1.96×10^{3}	4.58×10^{2}	1.69×10^{4}
10 years	6.47×10^{1}	1.06×10^{4}	3.77×10^{3}	1.54×10^{3}	4.58×10^{2}	1.64×10^{4}
15 years	6.23×10^{1}	1.06×10^{4}	3.78×10^{3}	1.21×10^{3}	4.58×10^{2}	1.61×10^{4}
30 years	5.55×10^{1}	1.05×10^{4}	3.78×10^{3}	5.86×10^{2}	4.58×10^{2}	1.54×10^{4}
100 years	3.24×10^{1}	1.05×10^{4}	3.77×10^{3}	1.99×10^{1}	4.58×10^{2}	1.48×10^{4}
300 years	7.14×10^{0}	1.05×10^{4}	3.69×10^{3}	4.97×10^{-3}	4.58×10^{2}	1.47×10^{4}
1000 years	5.70×10^{-2}	1.03×10^{4}	3.42×10^{3}	3.50×10^{-3}	4.58×10^{2}	1.42×10^{4}
3000 years	1.77×10^{-6}	9.71×10^{3}	2.77×10^{3}	2.97×10^{-3}	4.56×10^{2}	1.29×10^{4}
10,000 years	2.01×10^{-21}	7.98×10^{3}	1.32×10^{3}	1.68×10^{-3}	4.51×10^{2}	9.75×10^{3}
30,000 years	–	4.52×10^{3}	1.60×10^{2}	3.29×10^{-4}	4.34×10^{2}	5.11×10^{3}
100,000 years	–	6.05×10^{2}	9.84×10^{-2}	1.09×10^{-6}	3.81×10^{2}	9.86×10^{2}
250,000 years	–	8.10×10^{0}	2.67×10^{-8}	5.29×10^{-12}	2.89×10^{2}	2.97×10^{2}

	Other actinides per assembly (g)							
	^{235}U	^{236}U	^{238}U	^{237}Np	^{241}Am	^{242m}Am	^{243}Am	^{244}Cm
Charge	8.81×10^{2}	4.41×10^{0}	4.40×10^{5}	0.0	0.0	0.0	0.0	0.0
Discharge	4.65×10^{2}	9.62×10^{1}	4.27×10^{5}	6.83×10^{1}	1.12×10^{2}	3.34×10^{0}	1.20×10^{2}	3.92×10^{1}
1 d	4.65×10^{2}	9.62×10^{1}	4.27×10^{5}	6.84×10^{1}	1.13×10^{2}	3.34×10^{0}	1.20×10^{2}	3.93×10^{1}
1 year	4.66×10^{2}	9.66×10^{1}	4.27×10^{5}	6.97×10^{1}	2.30×10^{2}	3.32×10^{0}	1.20×10^{2}	3.79×10^{1}
5 years	4.67×10^{2}	9.82×10^{1}	4.27×10^{5}	7.25×10^{1}	6.45×10^{2}	3.26×10^{0}	1.20×10^{2}	3.25×10^{1}
10 years	4.68×10^{2}	1.00×10^{2}	4.27×10^{5}	7.93×10^{1}	1.06×10^{3}	3.18×10^{0}	1.20×10^{2}	2.68×10^{1}
15 years	4.57×10^{2}	9.98×10^{1}	4.29×10^{5}	8.06×10^{1}	1.30×10^{3}	2.63×10^{0}	1.20×10^{2}	2.05×10^{1}
30 years	4.60×10^{2}	1.06×10^{2}	4.29×10^{5}	1.19×10^{2}	1.84×10^{3}	2.44×10^{0}	1.20×10^{2}	1.15×10^{1}
100 years	4.78×10^{2}	1.34×10^{2}	4.29×10^{5}	3.53×10^{2}	2.14×10^{3}	1.73×10^{0}	1.19×10^{2}	7.91×10^{-1}
300 years	5.29×10^{2}	2.12×10^{2}	4.29×10^{5}	9.35×10^{2}	1.57×10^{3}	6.47×10^{-1}	1.17×10^{2}	3.73×10^{-4}
1000 years	7.04×10^{2}	4.75×10^{2}	4.29×10^{5}	1.97×10^{3}	5.10×10^{2}	2.07×10^{-2}	1.09×10^{2}	–
3000 years	1.18×10^{3}	1.13×10^{3}	4.29×10^{5}	2.45×10^{3}	2.08×10^{1}	1.11×10^{-6}	9.05×10^{1}	–
10,000 years	2.67×10^{3}	2.57×10^{3}	4.29×10^{5}	2.47×10^{3}	4.17×10^{-2}	–	4.68×10^{1}	–
30,000 years	5.61×10^{3}	3.74×10^{3}	4.29×10^{5}	2.45×10^{3}	8.11×10^{-3}	–	7.14×10^{0}	–
100,000 years	8.90×10^{3}	3.89×10^{3}	4.29×10^{5}	2.40×10^{3}	2.69×10^{-5}	–	9.88×10^{-3}	–
250,000 years	9.40×10^{3}	3.87×10^{3}	4.29×10^{5}	2.29×10^{3}	1.38×10^{-10}	–	1.38×10^{-6}	–

Table A.3. Actinide concentrations from MOX fuel following three cycles in Westinghouse PWR (fuel did not contain burnable absorbers)

	Plutonium per assembly (g)					
	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	Total plutonium
Charge	0.00×10^{0}	1.97×10^{4}	1.24×10^{3}	8.42×10^{1}	2.11×10^{1}	2.10×10^{4}
Discharge	1.15×10^{2}	6.25×10^{3}	3.75×10^{3}	2.49×10^{3}	9.52×10^{2}	1.36×10^{4}
1 d	1.15×10^{2}	6.26×10^{3}	3.75×10^{3}	2.49×10^{3}	9.52×10^{2}	1.36×10^{4}
1 year	1.48×10^{2}	6.29×10^{3}	3.76×10^{3}	2.37×10^{3}	9.52×10^{2}	1.35×10^{4}
5 years	1.52×10^{2}	6.29×10^{3}	3.77×10^{3}	1.96×10^{3}	9.52×10^{2}	1.31×10^{4}
10 years	1.46×10^{2}	6.29×10^{3}	3.79×10^{3}	1.54×10^{3}	9.52×10^{2}	1.27×10^{4}
15 years	1.40×10^{2}	6.29×10^{3}	3.80×10^{3}	1.21×10^{3}	9.52×10^{2}	1.24×10^{4}
30 years	1.25×10^{2}	6.29×10^{3}	3.83×10^{3}	5.84×10^{2}	9.52×10^{2}	1.18×10^{4}
100 years	7.25×10^{1}	6.28×10^{3}	3.84×10^{3}	1.99×10^{1}	9.52×10^{2}	1.12×10^{4}
300 years	1.55×10^{1}	6.25×10^{3}	3.76×10^{3}	1.49×10^{-2}	9.52×10^{2}	1.10×10^{4}
1000 years	9.65×10^{-2}	6.14×10^{3}	3.49×10^{3}	1.28×10^{-2}	9.51×10^{2}	1.06×10^{4}
3000 years	2.17×10^{-6}	5.84×10^{3}	2.83×10^{3}	1.09×10^{-2}	9.48×10^{2}	9.62×10^{3}
10,000 years	2.46×10^{-21}	4.87×10^{3}	1.35×10^{3}	6.16×10^{-3}	9.36×10^{2}	7.16×10^{3}
30,000 years	–	2.81×10^{3}	1.63×10^{2}	1.21×10^{-3}	9.02×10^{2}	3.88×10^{3}
100,000 years	–	3.78×10^{2}	1.00×10^{-1}	3.99×10^{-6}	7.93×10^{1}	1.17×10^{3}
250,000 years	–	5.06×10^{0}	1.68×10^{-7}	1.94×10^{-11}	6.00×10^{2}	6.05×10^{2}

	Other actinides per assembly (g)							
	^{235}U	^{236}U	^{238}U	^{237}Np	^{241}Am	^{242m}Am	^{243}Am	^{244}Cm
Charge	8.81×10^{2}	4.41×10^{0}	4.40×10^{5}	0.0	0.0	0.0	0.0	0.0
Discharge	2.92×10^{2}	1.16×10^{2}	4.22×10^{5}	8.49×10^{1}	1.45×10^{2}	4.09×10^{0}	2.88×10^{2}	1.33×10^{2}
1 d	2.92×10^{2}	1.16×10^{2}	4.22×10^{5}	8.50×10^{1}	1.45×10^{2}	4.09×10^{0}	2.89×10^{2}	1.33×10^{2}
1 year	2.92×10^{2}	1.17×10^{2}	4.22×10^{5}	8.65×10^{1}	2.62×10^{2}	4.07×10^{0}	2.89×10^{2}	1.28×10^{2}
5 years	2.93×10^{2}	1.18×10^{2}	4.22×10^{5}	8.95×10^{1}	6.75×10^{2}	3.99×10^{0}	2.89×10^{2}	1.10×10^{2}
10 years	2.94×10^{2}	1.20×10^{2}	4.22×10^{5}	9.66×10^{1}	1.09×10^{3}	3.89×10^{0}	2.88×10^{2}	9.08×10^{1}
15 years	2.95×10^{2}	1.22×10^{2}	4.22×10^{5}	1.06×10^{2}	1.41×10^{3}	3.80×10^{0}	2.88×10^{2}	7.50×10^{1}
30 years	2.98×10^{2}	1.28×10^{2}	4.22×10^{5}	1.47×10^{2}	1.99×10^{3}	3.53×10^{0}	2.88×10^{2}	4.22×10^{1}
100 years	3.10×10^{2}	1.56×10^{2}	4.22×10^{5}	3.99×10^{2}	2.30×10^{3}	2.50×10^{0}	2.86×10^{2}	2.89×10^{0}
300 years	3.45×10^{2}	2.35×10^{2}	4.22×10^{5}	1.02×10^{3}	1.68×10^{3}	9.35×10^{-1}	2.81×10^{2}	1.36×10^{-3}
1000 years	4.68×10^{2}	4.99×10^{2}	4.22×10^{5}	2.14×10^{3}	5.48×10^{2}	3.00×10^{-2}	2.63×10^{2}	–
3000 years	8.07×10^{2}	1.15×10^{3}	4.22×10^{5}	2.65×10^{3}	2.25×10^{1}	1.61×10^{-6}	2.18×10^{2}	–
10,000 years	1.86×10^{3}	2.61×10^{3}	4.22×10^{5}	2.67×10^{3}	1.86×10^{-1}	–	1.13×10^{2}	–
30,000 years	3.99×10^{3}	3.77×10^{3}	4.22×10^{5}	2.66×10^{3}	3.63×10^{-2}	–	1.72×10^{1}	–
100,000 years	6.39×10^{3}	3.92×10^{3}	4.22×10^{5}	2.60×10^{3}	1.20×10^{-4}	–	2.38×10^{-2}	–
250,000 years	6.76×10^{3}	3.91×10^{3}	4.23×10^{5}	2.48×10^{3}	6.16×10^{-10}	–	1.50×10^{-5}	–

Table A.4. Actinide concentrations from MOX fuel following three cycles in Westinghouse PWR (fuel did contain burnable absorbers)

	Plutonium per assembly (g)					
	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	Total Plutonium
Charge	0.00×10^{0}	1.97×10^{4}	1.24×10^{3}	8.42×10^{1}	2.11×10^{1}	2.10×10^{4}
Discharge	1.21×10^{2}	7.32×10^{3}	3.84×10^{3}	2.68×10^{3}	8.98×10^{2}	1.49×10^{4}
1 d	1.21×10^{2}	7.33×10^{3}	3.84×10^{3}	2.67×10^{3}	8.98×10^{2}	1.49×10^{4}
1 year	1.55×10^{2}	7.37×10^{3}	3.84×10^{3}	2.55×10^{3}	8.98×10^{2}	1.48×10^{4}
5 years	1.59×10^{2}	7.37×10^{3}	3.86×10^{3}	2.10×10^{3}	8.98×10^{2}	1.44×10^{4}
10 years	1.53×10^{2}	7.37×10^{3}	3.87×10^{3}	1.65×10^{3}	8.98×10^{2}	1.39×10^{4}
15 years	1.47×10^{2}	7.37×10^{3}	3.89×10^{3}	1.30×10^{3}	8.98×10^{2}	1.36×10^{4}
30 years	1.31×10^{2}	7.36×10^{3}	3.91×10^{3}	6.28×10^{2}	8.98×10^{2}	1.29×10^{4}
100 years	7.59×10^{1}	7.35×10^{3}	3.92×10^{3}	2.14×10^{1}	8.98×10^{2}	1.23×10^{4}
300 years	1.63×10^{1}	7.31×10^{3}	3.84×10^{3}	1.60×10^{-2}	8.98×10^{2}	1.21×10^{4}
1000 years	1.06×10^{-1}	7.18×10^{3}	3.57×10^{3}	1.39×10^{-2}	8.97×10^{2}	1.16×10^{4}
3000 years	2.55×10^{-6}	6.83×10^{3}	2.89×10^{3}	1.18×10^{-2}	8.94×10^{2}	1.06×10^{4}
10,000 years	2.89×10^{-21}	5.67×10^{3}	1.38×10^{3}	6.66×10^{-3}	8.83×10^{2}	7.93×10^{3}
30,000 years	–	3.26×10^{3}	1.67×10^{2}	1.30×10^{-3}	8.51×10^{2}	4.28×10^{3}
100,000 years	–	4.38×10^{2}	1.03×10^{-1}	4.32×10^{-6}	7.48×10^{2}	1.19×10^{3}
250,000 years	–	5.87×10^{0}	1.92×10^{-7}	2.10×10^{-11}	5.66×10^{2}	5.72×10^{2}

	Other actinides per assembly (g)							
	^{235}U	^{236}U	^{238}U	^{237}Np	^{241}Am	^{242m}Am	^{243}Am	^{244}Cm
Charge	8.81×10^{2}	4.41×10^{0}	4.40×10^{5}	0.0	0.0	0.0	0.0	0.0
Discharge	3.10×10^{2}	1.18×10^{2}	4.21×10^{5}	8.89×10^{1}	1.60×10^{2}	4.81×10^{0}	2.82×10^{2}	1.33×10^{2}
1 d	3.10×10^{2}	1.18×10^{2}	4.21×10^{5}	8.90×10^{1}	1.61×10^{2}	4.81×10^{0}	2.82×10^{2}	1.33×10^{2}
1 year	3.10×10^{2}	1.18×10^{2}	4.21×10^{5}	9.05×10^{1}	2.86×10^{2}	4.79×10^{0}	2.82×10^{2}	1.28×10^{2}
5 years	3.11×10^{2}	1.20×10^{2}	4.21×10^{5}	9.38×10^{1}	7.31×10^{2}	4.70×10^{0}	2.82×10^{2}	1.10×10^{2}
10 years	3.12×10^{2}	1.22×10^{2}	4.21×10^{5}	1.01×10^{2}	1.17×10^{3}	4.58×10^{0}	2.82×10^{2}	9.10×10^{1}
15 years	3.13×10^{2}	1.24×10^{2}	4.21×10^{5}	1.12×10^{2}	1.52×10^{3}	4.47×10^{0}	2.81×10^{2}	7.51×10^{1}
30 years	3.16×10^{2}	1.30×10^{2}	4.21×10^{5}	1.56×10^{2}	2.14×10^{3}	4.15×10^{0}	2.81×10^{2}	4.23×10^{1}
100 years	3.31×10^{2}	1.58×10^{2}	4.21×10^{5}	4.27×10^{2}	2.47×10^{3}	2.94×10^{0}	2.79×10^{2}	2.90×10^{0}
300 years	3.72×10^{2}	2.39×10^{2}	4.21×10^{5}	1.10×10^{3}	1.81×10^{3}	1.10×10^{0}	2.74×10^{2}	1.37×10^{-3}
1000 years	5.16×10^{2}	5.08×10^{2}	4.21×10^{5}	2.30×10^{3}	5.90×10^{2}	3.53×10^{-2}	2.57×10^{2}	–
3000 years	9.12×10^{2}	1.18×10^{3}	4.21×10^{5}	2.86×10^{3}	2.43×10^{1}	1.89×10^{-6}	2.13×10^{2}	–
10,000 years	2.15×10^{3}	2.66×10^{3}	4.21×10^{5}	2.88×10^{3}	2.01×10^{-1}	–	1.10×10^{2}	–
30,000 years	4.61×10^{3}	3.85×10^{3}	4.21×10^{5}	2.86×10^{3}	3.93×10^{-2}	–	1.68×10^{1}	–
100,000 years	7.40×10^{3}	4.01×10^{3}	4.21×10^{5}	2.80×10^{3}	1.30×10^{-4}	–	2.32×10^{-2}	–
250,000 years	7.82×10^{3}	3.99×10^{3}	4.21×10^{5}	2.66×10^{3}	6.66×10^{-10}	–	1.61×10^{-5}	–

Table A.5. Activity from Westinghouse PWR (two cycles) burning plutonium

	Light elements	Actinides	Fission products	Total
		Without WABAs (Ci/assembly)		
Discharge	2.35×10^{5}	2.06×10^{7}	8.30×10^{7}	1.04×10^{8}
1 d	1.31×10^{5}	7.82×10^{6}	1.75×10^{7}	2.55×10^{7}
1 year	8.54×10^{3}	2.53×10^{5}	1.21×10^{6}	1.47×10^{6}
5 years	2.45×10^{3}	1.98×10^{5}	2.14×10^{5}	4.14×10^{5}
10 years	1.18×10^{3}	1.58×10^{5}	1.27×10^{5}	2.86×10^{5}
15 years	6.47×10^{2}	1.26×10^{5}	1.05×10^{5}	2.32×10^{5}
30 years	1.97×10^{2}	6.67×10^{4}	7.14×10^{4}	1.38×10^{5}
100 years	7.81×10^{2}	1.14×10^{4}	1.38×10^{4}	2.53×10^{4}
300 years	2.11×10^{1}	6.96×10^{3}	1.80×10^{2}	7.16×10^{3}
1000 years	2.29×10^{0}	3.14×10^{3}	9.09×10^{0}	3.15×10^{3}
3000 years	2.00×10^{0}	1.27×10^{3}	8.82×10^{0}	1.28×10^{3}
10,000 years	1.67×10^{0}	7.54×10^{2}	8.62×10^{0}	7.64×10^{2}
30,000 years	1.19×10^{0}	2.88×10^{2}	8.09×10^{0}	2.97×10^{2}
100,000 years	5.77×10^{-1}	4.57×10^{1}	6.51×10^{0}	5.28×10^{1}
250,000 years	2.14×10^{-1}	1.91×10^{1}	4.29×10^{0}	2.36×10^{1}
		With WABAs (Ci/assembly)		
Discharge	2.50×10^{5}	2.16×10^{7}	8.33×10^{7}	1.05×10^{8}
1 d	1.40×10^{5}	8.21×10^{6}	1.76×10^{7}	2.60×10^{7}
1 year	9.04×10^{3}	2.68×10^{5}	1.21×10^{6}	1.49×10^{6}
5 years	2.50×10^{3}	2.11×10^{5}	2.14×10^{5}	4.28×10^{5}
10 years	1.20×10^{3}	1.68×10^{5}	1.27×10^{5}	2.96×10^{5}
15 years	6.52×10^{2}	1.34×10^{5}	1.05×10^{5}	2.40×10^{5}
30 years	1.98×10^{2}	7.10×10^{4}	7.12×10^{4}	1.42×10^{5}
100 years	7.78×10^{1}	1.21×10^{4}	1.37×10^{4}	2.59×10^{4}
300 years	2.10×10^{1}	7.40×10^{3}	1.88×10^{2}	7.61×10^{3}
1000 years	2.37×10^{0}	3.33×10^{3}	9.06×10^{0}	3.34×10^{3}
3000 years	2.08×10^{0}	1.35×10^{3}	8.75×10^{0}	1.36×10^{3}
10,000 years	1.73×10^{0}	8.22×10^{2}	8.56×10^{0}	8.32×10^{2}
30,000 years	1.22×10^{0}	3.29×10^{2}	8.03×10^{0}	3.38×10^{2}
100,000 years	5.87×10^{-1}	5.18×10^{1}	6.48×10^{0}	5.89×10^{1}
250,000 years	2.26×10^{-1}	2.02×10^{1}	4.28×10^{0}	2.47×10^{1}

Table A.6. Activity from Westinghouse PWR (three cycles) burning plutonium

	Light elements	Actinides	Fission products	Total
		Without WABAs (Ci/assembly)		
Discharge	2.57×10^{5}	2.30×10^{7}	8.42×10^{7}	1.07×10^{8}
1 d	1.45×10^{5}	8.64×10^{6}	1.79×10^{7}	2.67×10^{7}
1 year	1.10×10^{4}	2.91×10^{5}	1.42×10^{6}	1.72×10^{6}
5 years	3.54×10^{3}	2.18×10^{5}	2.95×10^{5}	5.17×10^{5}
10 years	1.73×10^{3}	1.74×10^{5}	1.85×10^{5}	3.61×10^{5}
15 years	9.61×10^{2}	1.40×10^{5}	1.54×10^{5}	2.95×10^{5}
30 years	3.08×10^{2}	7.43×10^{4}	1.04×10^{5}	1.79×10^{5}
100 years	1.27×10^{2}	1.29×10^{4}	2.01×10^{4}	3.31×10^{4}
300 years	3.41×10^{1}	7.43×10^{3}	2.44×10^{2}	7.71×10^{3}
1000 years	3.56×10^{0}	3.17×10^{3}	1.28×10^{1}	3.19×10^{3}
3000 years	3.11×10^{0}	1.18×10^{3}	1.25×10^{1}	1.20×10^{3}
10,000 years	2.59×10^{0}	6.66×10^{2}	1.22×10^{1}	6.81×10^{2}
30,000 years	1.84×10^{0}	2.31×10^{2}	1.14×10^{1}	2.44×10^{2}
100,000 years	8.94×10^{-1}	4.27×10^{1}	9.22×10^{0}	5.28×10^{1}
250,000 years	3.33×10^{-1}	2.48×10^{1}	6.08×10^{0}	3.12×10^{1}
		With WABAs (Ci/assembly)		
Discharge	2.60×10^{5}	2.30×10^{7}	8.42×10^{7}	1.07×10^{8}
1 d	1.47×10^{5}	8.65×10^{6}	1.79×10^{7}	2.67×10^{7}
1 year	1.10×10^{4}	3.10×10^{5}	1.41×10^{6}	1.73×10^{6}
5 years	3.48×10^{3}	2.33×10^{5}	2.94×10^{5}	5.30×10^{5}
10 years	1.70×10^{3}	1.86×10^{5}	1.84×10^{5}	3.72×10^{5}
15 years	9.41×10^{2}	1.50×10^{5}	1.54×10^{5}	3.05×10^{5}
30 years	3.00×10^{2}	7.96×10^{4}	1.04×10^{5}	1.84×10^{5}
100 years	1.23×10^{2}	1.38×10^{4}	2.00×10^{4}	3.39×10^{4}
300 years	3.31×10^{1}	7.97×10^{3}	2.50×10^{2}	8.25×10^{3}
1000 years	3.56×10^{0}	3.40×10^{3}	1.28×10^{1}	3.42×10^{3}
3000 years	3.11×10^{0}	1.26×10^{3}	1.24×10^{1}	1.28×10^{3}
10,000 years	2.58×10^{0}	7.22×10^{2}	1.22×10^{1}	7.37×10^{2}
30,000 years	1.83×10^{0}	2.60×10^{2}	1.14×10^{1}	2.73×10^{2}
100,000 years	8.81×10^{-1}	4.72×10^{1}	9.21×10^{0}	5.73×10^{1}
250,000 years	3.37×10^{-1}	2.61×10^{1}	6.09×10^{0}	3.25×10^{1}

Table A.7. Gamma dose rates from Westinghouse bare assembly following two reactor cycles

Time	Surface (rem/h)	At 1 m (rem/h)
	MOX burning without WABAs	
1 d	5.86×10^{6}	3.77×10^{5}
1 year	1.85×10^{5}	1.19×10^{4}
5 years	4.87×10^{4}	3.12×10^{3}
10 years	2.65×10^{4}	1.69×10^{3}
15 years	1.98×10^{4}	1.27×10^{3}
30 years	1.20×10^{4}	7.63×10^{2}
100 years	2.20×10^{3}	1.40×10^{2}
300 years	2.56×10^{1}	1.63×10^{0}
1000 years	2.57×10^{0}	1.63×10^{-1}
	MOX burning with WABAs	
1 d	5.91×10^{6}	3.81×10^{5}
1 year	1.86×10^{5}	1.19×10^{4}
5 years	4.82×10^{4}	3.09×10^{3}
10 years	2.60×10^{4}	1.66×10^{3}
15 years	1.94×10^{4}	1.24×10^{3}
30 years	1.18×10^{4}	7.52×10^{2}
100 years	2.19×10^{3}	1.40×10^{2}
300 years	2.56×10^{1}	1.63×10^{0}
1000 years	2.73×10^{0}	1.73×10^{-1}

Table A.8. Gamma dose rates from Westinghouse bare assembly following three reactor cycles

Time	Surface (rem/h)	At 1 m (rem/h)
	MOX burning without WABAs	
1 d	6.09×10^{6}	3.93×10^{5}
1 year	2.65×10^{5}	1.69×10^{4}
5 years	7.83×10^{4}	5.02×10^{3}
10 years	4.14×10^{4}	2.65×10^{3}
15 years	3.02×10^{4}	1.94×10^{3}
30 years	1.79×10^{4}	1.14×10^{3}
100 years	3.24×10^{3}	2.06×10^{2}
300 years	3.72×10^{1}	2.36×10^{0}
1000 years	3.85×10^{0}	2.43×10^{-1}
	MOX burning with WABAs	
1 d	6.13×10^{6}	3.95×10^{5}
1 year	2.60×10^{5}	1.66×10^{4}
5 years	7.55×10^{4}	4.84×10^{3}
10 years	3.97×10^{4}	2.54×10^{3}
15 years	2.91×10^{4}	1.86×10^{3}
30 years	1.75×10^{4}	1.12×10^{3}
100 years	3.23×10^{3}	2.06×10^{2}
300 years	3.73×10^{1}	2.37×10^{0}
1000 years	4.02×10^{0}	2.54×10^{-1}

Table A.9. Neutron dose rates from Westinghouse bare assembly following two reactor cycles

Time	Surface (rem/h)	At 1 m (rem/h)
	MOX burning without WABAs	
1 d	6.96×10^{0}	4.84×10^{-1}
1 year	3.43×10^{0}	2.38×10^{-1}
5 years	2.16×10^{0}	1.50×10^{-1}
10 years	1.81×10^{0}	1.26×10^{-1}
15 years	1.51×10^{0}	1.05×10^{-1}
30 years	9.05×10^{-1}	6.30×10^{-2}
100 years	1.65×10^{-1}	1.15×10^{-2}
300 years	9.07×10^{-2}	6.31×10^{-3}
1000 years	6.02×10^{-2}	4.19×10^{-3}
	MOX burning with WABAs	
1 d	7.38×10^{0}	5.14×10^{-1}
1 year	4.08×10^{0}	2.84×10^{-1}
5 years	2.78×10^{0}	1.93×10^{-1}
10 years	2.32×10^{0}	1.61×10^{-1}
15 years	1.93×10^{0}	1.35×10^{-1}
30 years	1.14×10^{0}	7.95×10^{-2}
100 years	1.81×10^{-1}	1.26×10^{-2}
300 years	9.02×10^{-2}	6.27×10^{-3}
1000 years	5.94×10^{-2}	4.13×10^{-3}

Table A.10. Neutron dose rates from Westinghouse bare assembly following three reactor cycles

Time	Surface (rem/h)	At 1 m (rem/h)
	MOX burning without WABAs	
1 d	1.80×10^{1}	1.25×10^{0}
1 year	1.10×10^{1}	7.67×10^{-1}
5 years	7.91×10^{0}	5.51×10^{-1}
10 years	6.54×10^{0}	4.55×10^{-1}
15 years	5.43×10^{0}	3.78×10^{-1}
30 years	3.15×10^{0}	2.19×10^{-1}
100 years	3.96×10^{-1}	2.75×10^{-2}
300 years	1.66×10^{-1}	1.16×10^{-2}
1000 years	1.26×10^{-1}	8.74×10^{-3}
	MOX burning with WABAs	
1 d	1.92×10^{1}	1.34×10^{0}
1 year	1.25×10^{1}	8.72×10^{-1}
5 years	9.31×10^{0}	6.48×10^{-1}
10 years	7.72×10^{0}	5.37×10^{-1}
15 years	6.41×10^{0}	4.46×10^{-1}
30 years	3.69×10^{0}	2.57×10^{-1}
100 years	4.17×10^{-1}	2.90×10^{-2}
300 years	1.49×10^{-1}	1.04×10^{-2}
1000 years	1.10×10^{-1}	7.62×10^{-3}

Table A.11. Neutron and gamma dose rates from 24 Westinghouse assemblies in cask following three reactor cycles

Time	Surface (rem/h)		At 1 m (rem/h)	
	Neutron	Gamma	Neutron	Gamma
MOX burning without WABAs				
1 d	5.30×10^{-2}	1.91×10^{1}	2.03×10^{-2}	8.74×10^{0}
1 year	3.23×10^{-2}	3.35×10^{-1}	1.24×10^{-2}	1.52×10^{-1}
5 years	2.31×10^{-2}	6.10×10^{-2}	8.85×10^{-3}	2.68×10^{-2}
10 years	1.91×10^{-2}	3.00×10^{-2}	7.32×10^{-3}	1.29×10^{-2}
15 years	1.59×10^{-2}	2.01×10^{-2}	6.08×10^{-3}	8.56×10^{-3}
30 years	9.22×10^{-3}	8.50×10^{-3}	3.53×10^{-3}	3.56×10^{-3}
100 years	1.17×10^{-3}	8.98×10^{-4}	4.50×10^{-4}	3.71×10^{-4}
300 years	4.97×10^{-4}	2.98×10^{-4}	1.90×10^{-4}	1.21×10^{-4}
1000 years	3.72×10^{-4}	2.24×10^{-4}	1.42×10^{-4}	9.10×10^{-5}
MOX burning with WABAs				
1 d	5.67×10^{-2}	1.95×10^{1}	2.17×10^{-2}	8.93×10^{0}
1 year	3.67×10^{-2}	3.32×10^{-1}	1.41×10^{-2}	1.50×10^{-1}
5 years	2.72×10^{-2}	6.04×10^{-2}	1.04×10^{-2}	2.64×10^{-2}
10 years	2.26×10^{-2}	3.02×10^{-2}	8.64×10^{-3}	1.29×10^{-2}
15 years	1.87×10^{-2}	2.06×10^{-2}	7.17×10^{-3}	8.73×10^{-3}
30 years	1.08×10^{-2}	9.14×10^{-3}	4.13×10^{-3}	3.81×10^{-3}
100 years	1.23×10^{-3}	9.32×10^{-4}	4.72×10^{-4}	3.84×10^{-4}
300 years	4.46×10^{-4}	2.67×10^{-4}	1.71×10^{-4}	1.08×10^{-4}
1000 years	3.25×10^{-4}	1.95×10^{-4}	1.24×10^{-4}	7.94×10^{-5}

APPENDIX B

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 microdisks.

The SAS2H code sequence was used to simulate the burnup of the fuel assemblies studied. Via ORIGEN-S the output from SAS2H 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 SAS1 code sequence to determine neutron and gamma dose rates for spent-fuel assemblies.

The files contained on the disk attached to this report are as follows:

W2SAS.IN:	The SAS2H input for two cycles with WABAs.
W3SAS.IN:	The SAS2H input for three cycles with WABAs.
NW2SAS.IN:	The SAS2H input for two cycles without WABAs.
NW3SAS.IN:	The SAS2H input for three cycles without WABAs.
C1ORG1.OUT:	ORIGEN-S output for Case 1 (1day, 1 year, 5 years, and 10 years).
C2ORG1.OUT:	ORIGEN-S output for Case 2 (1day, 1 year, 5 years, and 10 years).
C3ORG1.OUT:	ORIGEN-S output for Case 3 (1day, 1 year, 5 years, and 10 years).
C4ORG1.OUT:	ORIGEN-S output for Case 4 (1day, 1 year, 5 years, and 10 years).
C1ORG2.OUT:	ORIGEN-S output for Case 1 (15 years, 30 years, 100 years, 300 years, 1000 years, 3000 years, 10,000 years, 30,000 years, 100,000 years, and 250,000 years)
C2ORG2.OUT:	Same output for Case 2.
C3ORG2.OUT:	Same output for Case 3.
C4ORG2.OUT:	Same output for Case 4.
WSAS1.IN:	The SAS1 input for a case containing WABAs. This input can be used for a two or three-cycle situation by reading the appropriate source file, as the case may be, from unit 71 as written by ORIGEN-S.
NWSAS1.IN:	The SAS1 input for a case containing no WABAs.
CWSAS1.IN:	The SAS input for a shielding cask containing 24 assemblies (with WABAs).

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