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Physical and Decay Characteristics of Commercial LWR Spent Fuel

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NOTICE This document contains information of a preliminary nature. It is subject to revision or correction and therefore does not represent a final report.

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PHYSICAL AND DECAY CHARACTERISTICS OF COMMERCIAL LWR SPENT FUEL

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ABSTRACT

Information was collected from the literature and from major manufacturers that will be useful in the design and construction of a mined geologic repository for the disposal of light-water-reactor spent fuel. Pertinent data are included on mechanical design characteristics and materials of construction for fuel assemblies and fuel rods and computed values for heat generation rates, radioactivity, and photon and neutron emission rates as a function of time for four reference cases.

Calculations were made with the ORIGEN2 computer code for burnups of 27,500 and 40,000 MWd for a typical boiling-water reactor and 33,000 and 60,000 MWd for a typical pressurized-water reactor. The results are presented in figures depicting the individual contributions per metric ton of initial heavy metal for the activation products, fission products, and actinides and their daughters to the radioactivity and thermal power as a function of time. Tables are also presented that list the contribution of each major nuclide to the radioactivity, thermal power, and photons and neutrons emitted for disposal periods from 1 to 100,000 years.

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1. INTRODUCTION

With the signing of the Nuclear Waste Policy Act of 1982 on January 7, 1983, President Reagan authorized the U.S. Department of Energy (DOE) to site, design, and construct mined geologic repositories for the disposal of spent nuclear fuel and high-level radioactive wastes (HLW) and to obtain licenses from the Nuclear Regulatory Commission (NRC) to achieve this mandate.[1] The Act also requires the DOE to operate the repositories in such a manner as to adequately protect the public and the environment for an extended period of time (>10,000 years). In order to meet these tasks, DOE developed a program to fulfill the requirements of the Act which is described in its Mission Plan.[2]

The purpose of this study, the first in a series, is to provide reference quantities and characteristics for selected waste and waste forms that will be used to support the various testing, design, and analytic activities of the national repository program. This document is to augment Appendix B of the report entitled Generic Requirements for a Mined Geologic Disposal System.[3] Specifically, information on the following areas will be delineated for commercial light-water-reactor (LWR) spent fuel: (1) mechanical design characteristics and materials of construction for assemblies and rods and (2) computed values for heat generated, radioactivity present, and photons and neutrons emitted for four reference cases, two different burnups each for a boiling-water (BWR) and a pressurized-water reactor (PWR).

Information was gathered from a literature survey and from experts in the field. Requests were made to the major vendors to supply pertinent data on their fuel assemblies/rods. The computerized literature search facilities were used to access Energy Research Abstracts, Nuclear Science Abstracts, the Nuclear Safety Information Center Data Base, and the Power Reactor Docket Information Data Base. To augment these sources and to reduce the possible omission of important records, a routine search was also made of the more important journals in the field of nuclear energy.

For organizational purposes, references cited in each section are included at the end of that section along with the figures and tables cited in the preceding text.

1.1 REFERENCES FOR SECT. 1

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2. MECHANICAL DESIGN CHARACTERISTICS AND MATERIALS OF CONSTRUCTION

A light-water reactor (LWR) fuel rod consists essentially of a stack of uranium oxide (UO_2) pellets encapsulated in a helium atmosphere within a Zircaloy tube (in four of the early LWRs, stainless steel cladding was used). The fuel assemblies are constructed from a number of individual fuel rods arranged in square arrays. In the older PWR designs, the fuel assemblies (also called fuel bundles) consisted of 14 × 14 and 15 × 15 fuel rod arrays and, in the more recent designs, 16 × 16 and 17 × 17 arrays.[1] For the BWRs, the older design utilized a 7 × 7 array and the more recent design, an 8 × 8 array. A new 9 × 9 array is now being marketed.

In this section, information is given on the mechanical description of all major types of LWR fuel assemblies in existence in the United States based on the dimensions prior to reactor use or on as-designed dimensions. Small dimensional changes occur during reactor operations, particularly in the fuel rods. The types of materials found in any U.S. fuel assembly are listed in Table 2.1.[2] The mechanical design parameters, which include data on fuel assemblies, fuel rods, fuel pellets, control guide and instrument tubes, tie plates, grid spacers, and springs, for the fuel assemblies that are produced by each of the manufacturers are described individually in the following sections.

2.1 FUEL ASSEMBLIES OF PRESSURIZED-WATER REACTORS (PWRs)

The PWR fuel assemblies are currently manufactured by Westinghouse Electric Corporation, Combustion Engineering, Babcock and Wilcox, and Exxon Nuclear. All PWR fuel assemblies, or fuel bundles, have much in common — stacked UO_2 pellets under helium pressure encased in Zircaloy tubing and arranged in square arrays. In three of the early PWRs (Haddam Neck, Yankee Rowe, and San Onofre-1) the cladding was stainless steel, which is still used in the Haddam Neck reactor.

2.1.1 Westinghouse Fuel Assembly Designs

The Westinghouse PWR fuel assemblies are products of evolutionary procedures to improve performance and economics. The first commercial PWR plant was Yankee Rowe, which was originally fueled with stainless steel clad fuel rods in a 15 × 15 array of the fuel rods. In 1968, the fuel clad material was changed from stainless steel to Zircaloy with its first commercial use being in the Jose Cabrera reactor plant in Spain. In 1970, the guide thimbles were changed from stainless steel to Zircaloy, and prepressurization of the fuel rods with helium was introduced.[3]

The most recent version of the Westinghouse fuel assembly incorporates 264 fuel rods mechanically joined in a 17 × 17 square array, with the rods supported at intervals by grid assemblies (see Fig. 2.1 for a typical fuel rod and Fig. 2.2 for a typical fuel assembly from ref. 4). The grid assembly is an egg-crate arrangement of interlocked Inconel straps which contain spring fingers for rod support and mixing vanes. The fuel rods contain cylindrical pellets of slightly enriched UO_2 enclosed in cold-worked Zircaloy-4 tubing that is plugged and seal-welded at the ends. The plenum springs inside the rods are Inconel-718. The center position on the assembly holds the in-core instrumentation, and 24 positions in the array are equipped with Zircaloy-4 guide thimbles joined to the grids and top and bottom nozzles.[2] The assemblies with the smaller numbers of fuel rods are qualitatively similar in construction.

Fuel assemblies with 14 × 14, 15 × 15, 16 × 16, and 17 × 17 square arrays of fuel rods manufactured by Westinghouse are currently in use or stored in pools as spent fuel. A small number of the 15 × 15 array elements used in the Yankee Rowe reactor have been reprocessed at West Valley.[5,6] Over the years, these types of fuel elements have been subjected to improvements with regard to pellet-clad interactions (PCI), neutron economy, and achieving higher burnup. The older versions are designated as "standard" (Std) and the more recent improved versions as either "optimized fuel assemblies" (OFA) or as VANTAGE 5. The VANTAGE 5 assembly (Fig. 2.3) is a modified version of a 17 × 17 OFA element currently being marketed and is scheduled for commercial use in 1987.

The OFA types have been through demonstration phases, and full-core reloads with 14 × 14, 15 × 15, and 17 × 17 OFAs have begun at a number of reactor plants.[7] All the OFA types have Zircaloy intermediate grids substituted for the Inconel-718 ones in the standard version. The 14 × 14 and 17 × 17 OFAs have fuel rods of a slightly smaller diameter, and the 14 × 14 and 15 × 15 OFAs have instrument tubes of a slightly smaller diameter than the standard versions. The dimensions of the 17 × 17 VANTAGE 5 fuel assembly are essentially identical with the 17 × 17 OFA (except for a 0.3-in. longer length). The difference is that the VANTAGE 5 possesses natural uranium axial blankets, an intermediate flow mixer, integral fuel burnable poisons in the form of a thin, boride coating on the fuel pellet surfaces, and a top nozzle that has a snap-lock design to permit rapid removal and relock for fuel rod replacement or assembly reconstitution.

Detailed design dimensions for Westinghouse fuel assemblies and other characteristics of these assemblies, as shown in Table 2.2, were supplied by Westinghouse.[8]

2.1.2 Combustion Engineering Fuel Assembly Designs

The most recent version of the Combustion Engineering fuel assembly consists of 236 fuel rod locations in a 16 × 16 array with 4 control element guide tubes, a centrally located instrumentation guide tube, 12 spacer grids, upper and lower end fittings, and a hold-down device. Guide tubes, spacer grids, and end fittings form the structural framework of the assembly. The Zircaloy-4 spacer grid and guide tubes are welded together, but stainless steel end fittings are mechanically attached to the five guide tubes. The fuel rods consist of slightly enriched UO_2 cylindrical pellets, a round wire stainless steel compression spring, and an aluminum spacer disc located at each end of the fuel column, all of which are encapsulated within a cold-worked Zircaloy-4 tube that is internally pressurized with helium during assembly. Zircaloy-4 caps are seal-welded on the ends. All grids, except the bottom Inconel spacer grid, are fabricated from preformed Zircaloy strips that are interlocked in egg-crate fashion and welded together.[2]

Schematic drawings of a fuel rod and fuel assemblies for the St. Lucie-1 plant, which employs 14 × 14 fuel assemblies, are shown in Figs. 2.4 and 2.5. Figures 2.6 and 2.7 show a fuel rod and a 16 × 16 array fuel assembly from the Arkansas Nuclear One, Unit 2 plant. Some details of the dimensions and other characteristics of both the 16 × 16 and the 14 × 14R designs, as shown in Table 2.3, were furnished by Combustion Engineering.[9] Table 2.4 shows the number of Combustion Engineering fuel assemblies active (in reactor) and discharged from various reactors for 14 × 14, 16 × 16, and 15 × 15 fuel rod arrays as of November 1, 1984.

2.1.3 Babcock and Wilcox Fuel Assembly Designs

The newer Babcock and Wilcox fuel assembly (Mark C) incorporates fuel rods in a 17 × 17 square array (see Fig. 2.8 for a typical fuel rod and Fig. 2.9 for a fuel assembly), whereas the earlier Mark B design used a 15 × 15 array. The earliest version of a Babcock and Wilcox fuel element consisted of a 14 × 14 array, but these fuel elements (from the Indian Point-1 Plant) have all been reprocessed at West Valley.[10] The fuel is slightly enriched UO_2 pellets clad in Zircaloy-4 tubing under helium pressure and sealed by Zircaloy-4 end caps welded at each end. Spring spacers between the fuel column and end caps are located at both the top and bottom of the fuel rods, with insulating spacers between the fuel column and spring spacers. The newer Mark C fuel assembly is made up of 264 fuel rods, 24 Zircaloy-4 control rod guide tubes, 1 Zircaloy-4 instrumentation tube assembly, 8 Zircaloy-4 spacer grids, and 2 stainless steel end fittings. These guide tubes, spacer grids, and end fittings form a structural cage, with the center position in the assembly reserved for instrumentation.[2]

Some details of the dimensions and other characteristics of both the 17 × 17 and the older 15 × 15 designs were furnished by Babcock and Wilcox[11] and are shown in Table 2.5. The 15 × 15 stainless steel assembly listed is manufactured only for the Haddam Neck reactor, which has not yet been licensed to operate with Zircaloy-clad fuel rods. Table 2.6 shows some details of control rod and burnable poison rod assemblies. The number of fuel assemblies shipped to the various reactors by Babcock and Wilcox is shown in Table 2.7.

2.1.4 Exxon Nuclear Fuel Assembly Designs

Exxon Nuclear has been fabricating fuel assemblies for both PWR and BWR reactors. These are designed with mechanically locked upper tie plates to allow removal and replacement of individual rods. In PWR fuel assemblies, cap screws lock the lower tie plate to the guide tubes or guide bars.[12] Reload batches only have been produced; no known plans exist for providing initial fuel loads, but that could change in the future. Consequently, the Exxon assemblies are almost identical to those manufactured by the other LWR vendors already described. Exxon Nuclear claims to have the capability to supply reload assemblies for PWRs that would be compatible with Westinghouse's 15 × 15 and 17 × 17 arrays and Combustion Engineering's 14 × 14 and 15 × 15 arrays. It appears that they do not fabricate fuel for reloads for Babcock and Wilcox reactors or for Combustion Engineering's 16 × 16 design. As of January 1983, Exxon Nuclear reloads have replaced Combustion Engineering's 15 × 15 (Palisades), Westinghouse's 15 × 15 (DC Cook-1 and Yankee Rowe), and Westinghouse's 14 × 14 (Ginna and Kewaunee).[13] See Table 2.8 for some mechanical design parameters of Exxon Nuclear fuel assemblies.

2.2 FUEL ASSEMBLIES OF BOILING-WATER REACTORS (BWRs)

The generation of commercial nuclear power began with the startup of the Dresden-1 BWR in 1959. The oldest spent fuel elements in storage were discharged from the Dresden-1 reactor in 1969. All spent fuel discharged in prior years has been reprocessed at West Valley.[1] Fuel assemblies for BWRs are currently being manufactured by General Electric and Exxon Nuclear, with a new design being offered commercially by Westinghouse. The three improved BWR designs being offered commercially in the United States at this time include: (1) the 8 × 8 with a zirconium liner (or barrier as it is called) by General Electric, (2) the 9 × 9 by Exxon Nuclear, and (3) the 8 × 8 water-cross design (QUAD+) by Westinghouse.[14]

2.2.1 General Electric Fuel Assembly Designs

The BWR designs by General Electric have evolved in steps from the BWR/1 class or system starting with the Dresden-1 plant through

BWR/2, 3, 4, and 5 to the BWR/6.[15] Table 2.9 shows the significant characteristics of each reactor system and the year of introduction. Except for the BWR/6 class, changes in the elements were minor for each new class and were primarily in the system external to the fuel elements, particularly in the control rod arrangements. With the BWR/6 class, however, the fuel element array was increased to an 8 × 8 fuel rod array. Except for some early BWR/1 fuel elements, all fuel elements for earlier classes were composed of 7 × 7 arrays. Some early loadings were 6 × 6 arrays in Dresden-1 and 9 × 9 in the Big Rock Point reactor. One of these arrays was apparently used as an early loading at Humboldt Bay. All of these earlier fuel assemblies have been reprocessed at West Valley.

The newest version of the 8 × 8 fuel elements for BWR/6 class has a 0.003-in. zirconium (barrier) liner bonded metallurgically to the Zircaloy-2 cladding on the inner diameter, which is the only difference from the "standard" 8 × 8 version. The total clad thickness for both fuel elements is the same, 0.032 in. The BWR/6 system, however, employs no new technology that has not been applied previously or has not been demonstrated in test facilities.[15]

General Electric designs its fuel assembly to consist of a fuel bundle and its surrounding channel made of Zircaloy-4. A standard BWR/6 fuel bundle contains 63 fuel rods and 1 water rod (later models have used two water rods), which are spaced in a square 8 × 8 array by lower and upper tie plates fabricated from type 304 stainless steel castings. Each fuel rod consists of high-density UO_2 fuel pellets stacked in a Zircaloy-2 cladding tube that is evacuated, filled with helium, and sealed by welding on Zircaloy end plugs. A plenum spring prevents fuel column movement inside the fuel rod during handling. Three types of rods used in a fuel bundle are tie rods, one nonfuel water rod, and standard rods. The eight tie rods, containing fuel pellets in each bundle, have lower end plugs that thread into the lower tie plate casting and upper end plugs that extend through the upper tie plate casting. A stainless steel, hexagonal nut and locking tab are installed in the upper end plug to hold the assembly together. The one rod containing water in each fuel bundle is a hollow Zircaloy-2 tube with a square bottom end plug used to position seven Zircaloy-4 rod spacers vertically in the bundle. The spacers equipped with Inconel-X springs maintain rod-to-rod spacing.

The remaining 55 rods in a bundle are standard rods, with a tube of Zircaloy-2 encapsulating the same length of the fuel pellets as the tie rods. The end plugs of the standard fuel rods fit into the lower tie plate. An Inconel compression spring located over the upper end plug of each fuel rod keeps the rod in place.

The fuel channel enclosing the fuel bundle has a square cross section with rounded corners and is fabricated from Zircaloy-4. The channel provides a barrier to separate coolant flow paths, to guide the control rod, and to provide rigidity and protection for the fuel bundle during handling. When attached to the upper tie plate by the channel fastener assembly, it spaces the four fuel assemblies in a core cell properly.[2,16] The nominal inside width of the fuel channel for BWR/6 reactors is 5.215 in. and 5.278 in. for all BWR/2, 3, 4, and 5 reactors. The nominal length is 162.156, 166.906, and 167.36 in. for BWR/2 and 3, BWR/4 and 5, and BWR/6 reactors, respectively. Fuel channels having wall thicknesses of 0.080, 0.100, and 0.120 in. are currently in production, with the newer designs having the thicker walls.[17] The 100-mil-thick channel weighs 45.5 kg.[18] In the earlier loadings, a fuel channel had a lifetime of one fuel assembly. With improvements, some began to be reused. The newer, thicker designs are expected to last through a number of reloadings. Estimates of the fraction of spent BWR assemblies that are accompanied by fuel channels are not available. A typical General Electric fuel rod and a fuel assembly are shown in Fig. 2.10, and more detailed cross sections of an 8 × 8 fuel assembly are shown in Fig. 2.11.

The 7 × 7 fuel assemblies are very similar in construction with the same overall transverse dimensions, length, and width. The fuel rod diameters are a little larger, and the pitch is greater to make up for the smaller number of fuel rods.

About 30 production fuel types have been manufactured and operated in BWRs. Besides the small dimensional changes made over the years in the fuel rods, there are variations in the enrichments and the distribution of gadolinium burnable poisons that are more or less reactor specific. Reload (or retrofit) fuel assemblies designated by 7 × 7R and 8 × 8R have also incorporated small changes, mostly in the fuel rods. Table 2.10 indicates some changes that have taken place over the years. Some of the 8 × 8R fuel assemblies prepared to replace 7 × 7 fuel assemblies in the BWR/2, 3, and 4 reactor classes are quite similar to the standard design for the BWR/6 class, with the exception that an additional water rod is used in place of a fuel rod.[19] In addition, the BWR/4 reload bundles have an active fuel length and/or gas plenum length that is slightly longer than in the BWR/2 and 3 reload bundles, which is consistent with previous fuel designs for these plants.[19] Table 2.11 shows some mechanical design parameters and ranges (includes standards and reloads) for General Electric fuel assemblies.

2.2.2 Westinghouse Electric Corporation Fuel Assembly Design

The "water-cross" design being offered by Westinghouse was developed by ASEA-ATOM (AA) (Sweden), and the first reloads were offered commercially for AA and Kraftwerk Union AG (KWU) plants starting in 1982. The trade name for AA's design is SVEA. The Westinghouse Electric Corporation signed a technology exchange agreement with AA in 1980 and entered the BWR reload market with the adaptation of the SVEA design, which has the trade name QUAD+. The general configuration of the QUAD+ design is the same as SVEA's; nevertheless, there are significant differences between the two designs which reflect different constraints, especially those related to transient analyses of the reactors. The first commercial QUAD+ reload offers were made in 1982. The QUAD+ design has an 8 × 8 fuel rod array subdivided into 4 × 4 subarrays, or minibundles, which are separated by a hollow Zircaloy sheet-metal cross filled with nonboiling water.[14] The overall configuration is shown in Figs. 2.12 and 2.13. The "channel assembly," which is the combination of the water cross, the outer channel, and the bottom nozzle, is the most novel mechanical design feature of the QUAD+.

The channel assembly, the structural part of the fuel assembly, consists of the Zircaloy channel welded to the "water cross," as shown in Fig. 2.14. The channel is attached mechanically to the lower nozzle by three Inconel screws per side and to the upper nozzle through four rectangular cross-section Zircaloy bars welded to the inside surface of the channel and bolted to the top nozzle. The top nozzle has a standard bail for lifting the assembly. The water cross is made of a 0.028-in.-Zircaloy-4 sheet, with the walls spaced by welded dimples to produce a 0.240-in. water gap. The bottom of the cross is sealed by appropriately shaped end plugs that are welded to the walls of the cross. The channel, water cross, and upper and lower nozzle assembly form a basket that accommodates the four 4 × 4 array minibundles. Each 4 × 4 minibundle is an independently removable subassembly that consists of an upper and lower tie plate, two tie rods, and six spacers as shown in Figs. 2.15 and 2.16. Disassembly of the fuel can be accomplished by detaching the upper nozzle from the four posts and lifting the upper nozzle off by a standard tool, but a special handling tool is needed to grapple and move the minibundles out of the channel assembly. The fuel assembly does not have any water rods since the water is provided in the water cross. The tie rods are attached mechanically to the top and bottom tie plates, which serve the normal functions of spacing the fuel rods and orificing bundle flow. The spacers are made of Zircaloy-4 with integral Zircaloy-4 springs and are captured by cylindrical Zircaloy tabs welded to the Zircaloy cladding of a fuel rod. Table 2.11 gives some dimensional details and materials of construction of the QUAD+ assembly (see ref. 14 for more details of the fuel assembly).

2.2.3 Exxon Nuclear Fuel Assembly Designs

As previously mentioned, Exxon Nuclear has been fabricating fuel assemblies for both PWRs and BWRs for reload batches only, and these assemblies are almost identical to those manufactured by the other vendors. Exxon Nuclear, however, does not make fuel channels. They have produced reloads for the BWR/2, 3, 4, 5, and 6 General Electric reactor classes.

The BWR fuel assemblies are also designed with mechanically locked upper tie plates to allow removal and replacement of individual fuel rods. The assemblies are fabricated by screwing the threaded lower end cap of the tie rods into the lower tie plate. The standard fuel rods are held in position in the lower tie plate by compression springs bearing against the upper tie plate.

Exxon Nuclear is now offering a new type fuel assembly with a 9 × 9 array of fuel rods.[20] The primary difference between the 9 × 9 and 8 × 8 array is the number of fuel rods per assembly. Either type of fuel array could be designed with various numbers of water rods; however, the commercial design being offered at this time by Exxon Nuclear and the new 8 × 8 array by General Electric contain two water rods. The general mechanical design features are not changed by increasing the number of fuel rods per assembly, even though the detailed dimensions of the components are changed. The spacers have to accommodate more rods that require additional strips and springs, and the tie plates are similarly affected.[14] Exxon fuel assemblies are designed with mechanically locked upper tie plates to allow removal and replacement of individual fuel rods. Table 2.12 shows some details of the mechanical design parameters of the 7 × 7 and 8 × 8 fuel assemblies that have been fabricated by Exxon Nuclear as well as the new 9 × 9 fuel assembly.

2.2.4 Allis-Chalmers Fuel Assembly Design

Allis-Chalmers supplied the early loadings for the La Crosse reactor but ceased production after a few loadings. Fuel is now supplied by Exxon Nuclear.

The fuel rods are arranged in a 10 × 10 array in the early Allis Chalmers fuel assemblies. Each fuel rod is a stainless steel tube loaded with UO_2 fuel pellets, with a stainless steel spring provided at each end space to prevent gaps between the pellets. The tube is filled with helium at atmospheric pressure, and end plugs are welded to the tube. Each fuel assembly is composed of 88 regular fuel rods, 8 hold-down fuel rods, and 4 segmented fuel rods. The regular fuel rod consists of fuel pellets, a 3.93-in. space with spring, and end plugs with bullet-shaped tips that rest in the upper and lower fuel assembly grids. The hold-down fuel rod consists of fuel pellets, a 3.67-in. space with spring, and threaded end plugs that are bolted onto the fuel assembly grid. The segmented fuel rod consists of four separate segments (three with pellets and a 1-in. gap and one with pellets and a 3.93-in. gap with spring). The segments are joined together by stainless steel fittings that lock the intermediate grids in place at three positions along the length.[21] Some details of the fuel assembly are shown in Table 2.13.

2.3 FUTURE FUEL ASSEMBLY DESIGNS

The commercial fuel assembly designs have evolved through numerous design changes since the first reactors, with the process continuing. The focus of new designs has been, and undoubtedly will continue to be, on increasing burnup and developing increased resistance to pellet-clad interactions (PCI). Other developments that have occurred involved the fuel rod connections with the assembly body. Newer designs permit easy fuel rod removals and replacements (either partially or entirely), which permit improved fuel management and easy disassembly for reprocessing or disposal in canisters packed with fuel rods.

It is difficult to foresee any further design changes in the fuel assemblies that will have a large effect from a waste disposal viewpoint. For existing reactors and those under construction, the overall dimensions of the fuel assembly cannot be changed significantly without redesign of the core, and commercial development of any new core concept is in the distant future for the United States. Such factors as small changes in the rod diameter and use of additional water rods in the newer BWR designs can have little effect on canisters packed with fuel rods.

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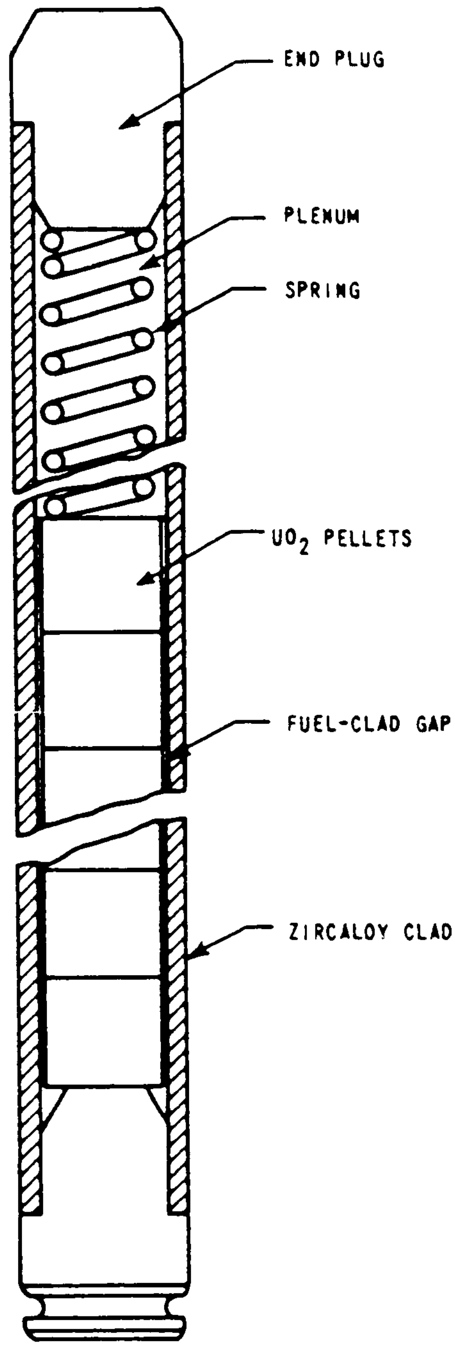

SPECIFIC DIMENSIONS DEPEND ON DESIGN VARIABLES
SUCH AS PREPRESSURIZATION, POWER HISTORY, AND
DISCHARGE BURNUP.

Fig. 2.1. Schematic of a typical Westinghouse fuel rod. (Source: F. J. Frank, Westinghouse Electric Corporation, letter to H. C. Claiborne, Oak Ridge National Laboratory, July 18, 1985.)

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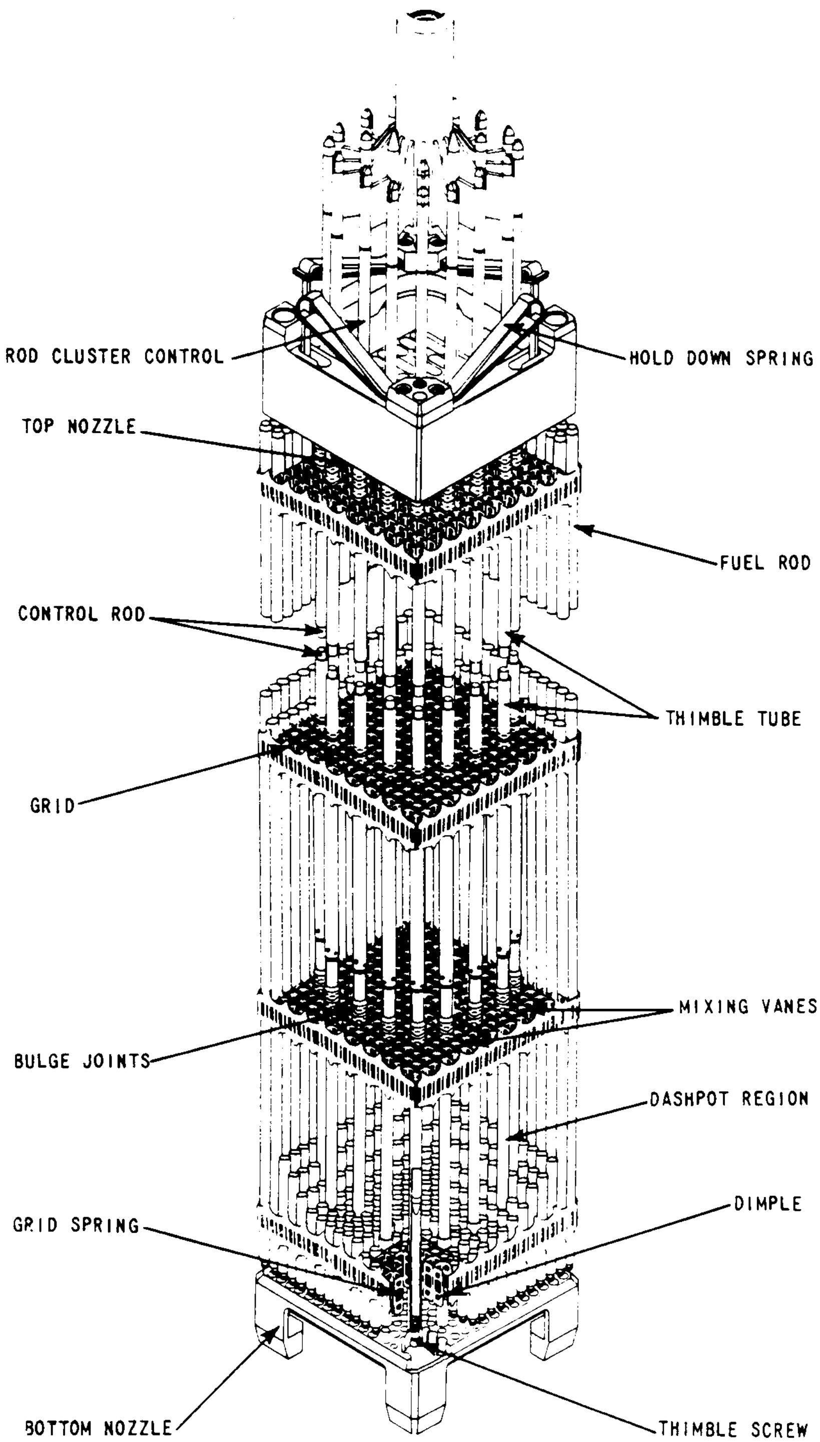

Fig. 2.2. Schematic of a typical Westinghouse fuel assembly. (Source: F. J. Frank, Westinghouse Electric Corporation, letter to H. C. Claiborne, Oak Ridge National Laboratory, July 18, 1985.)

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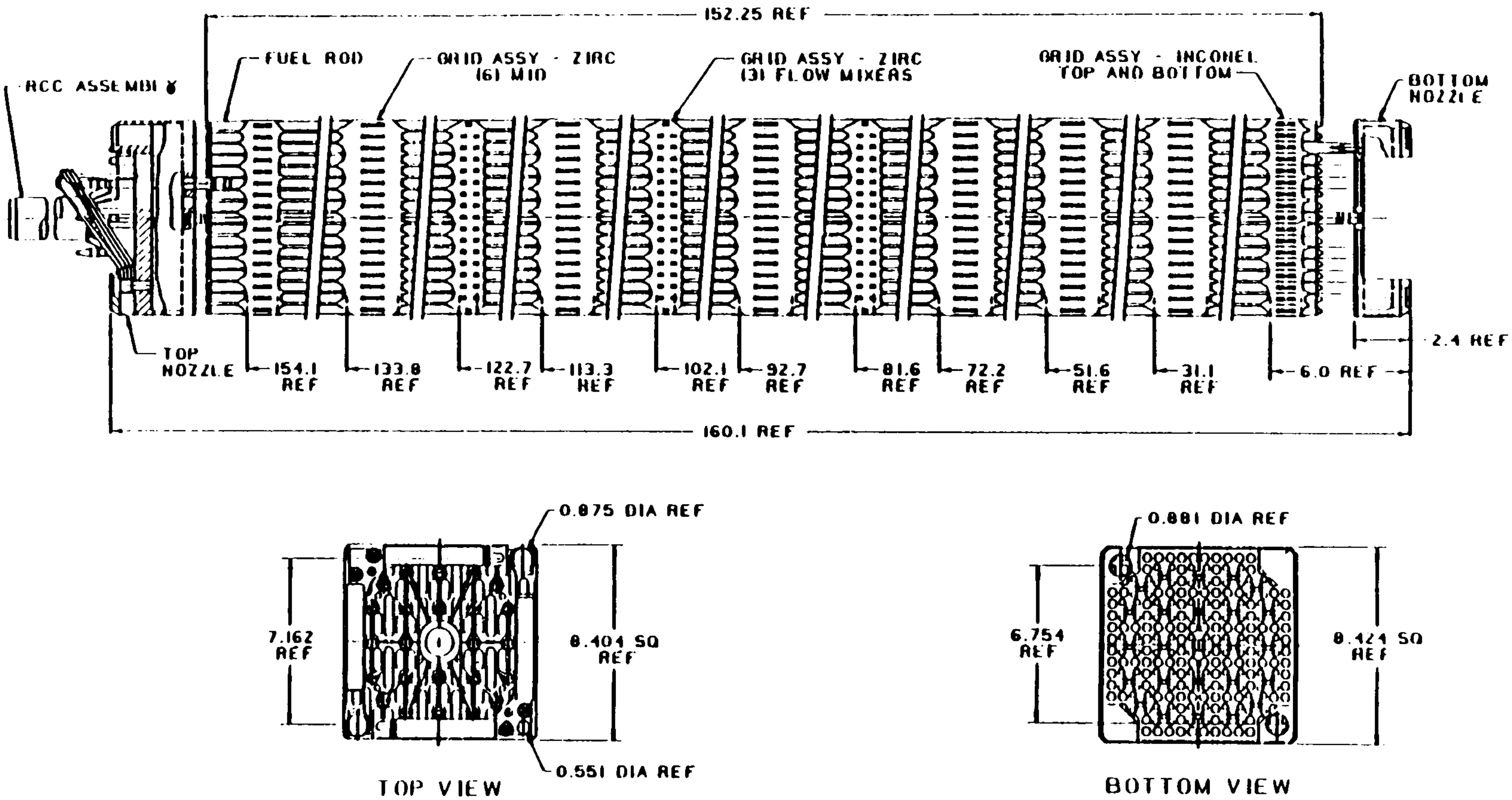

Fig. 2.3. Schematic of VANTAGE 5 fuel assembly.
(Source: F. J. Frank, Westinghouse Electric Corporation, letter to H. C. Claiborne, Oak Ridge National Laboratory, July 18, 1985.)

ORNL DWG 85-989

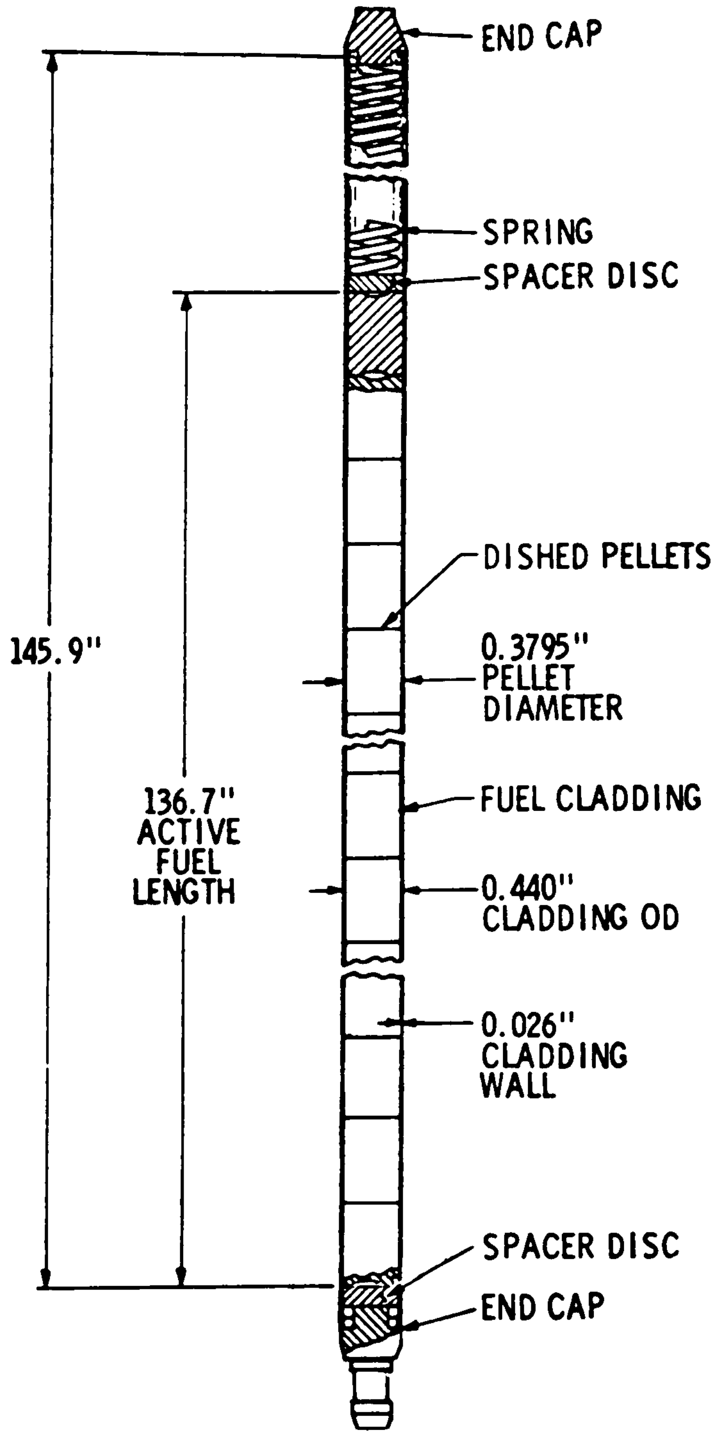

Fig. 2.4. Schematic of a fuel rod from St. Lucie Plant-1 — 14 × 14 array. (Source: M. G. Andrews, C-E Power Systems, Combustion Engineering, Inc., letter to J. W. Roddy, Oak Ridge National Laboratory, February 11, 1985.)

ORNL DWG 85-988

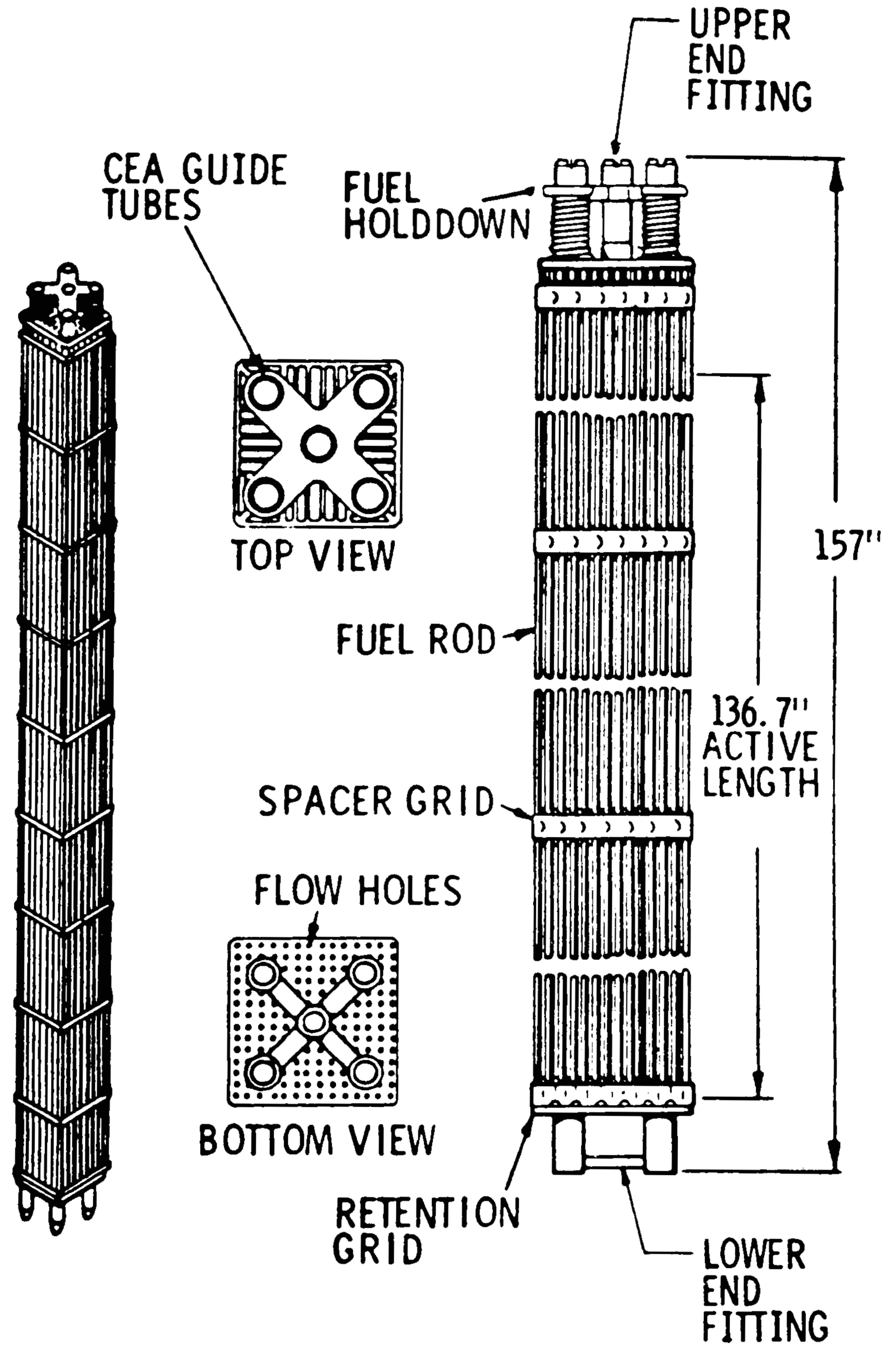

Fig. 2.5. Schematic of a fuel assembly from St. Lucie Plant-1 — 14 × 14 array. (Source: M. G. Andrews, C-E Power Systems, Combustion Engineering, Inc., letter to J. W. Roddy, Oak Ridge National Laboratory, February 11, 1985.)

ORNL DWG 85-986

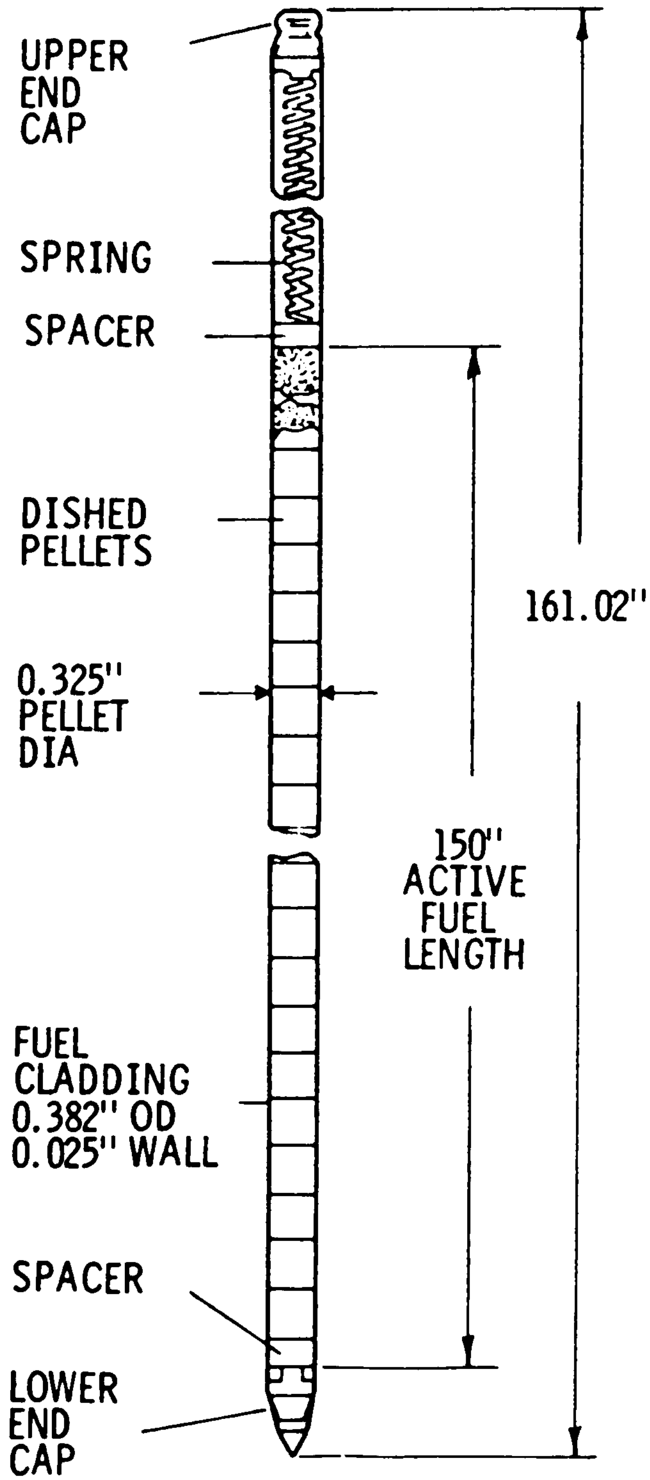

Fig. 2.6. Schematic of a fuel rod from Arkansas Nuclear One, Unit 2 — 16 × 16 array. (Source: M. G. Andrews, C-E Power Systems, Combustion Engineering, Inc., letter to J. W. Roddy, Oak Ridge National Laboratory, February 11, 1985.)

ORNL DWG 85-987

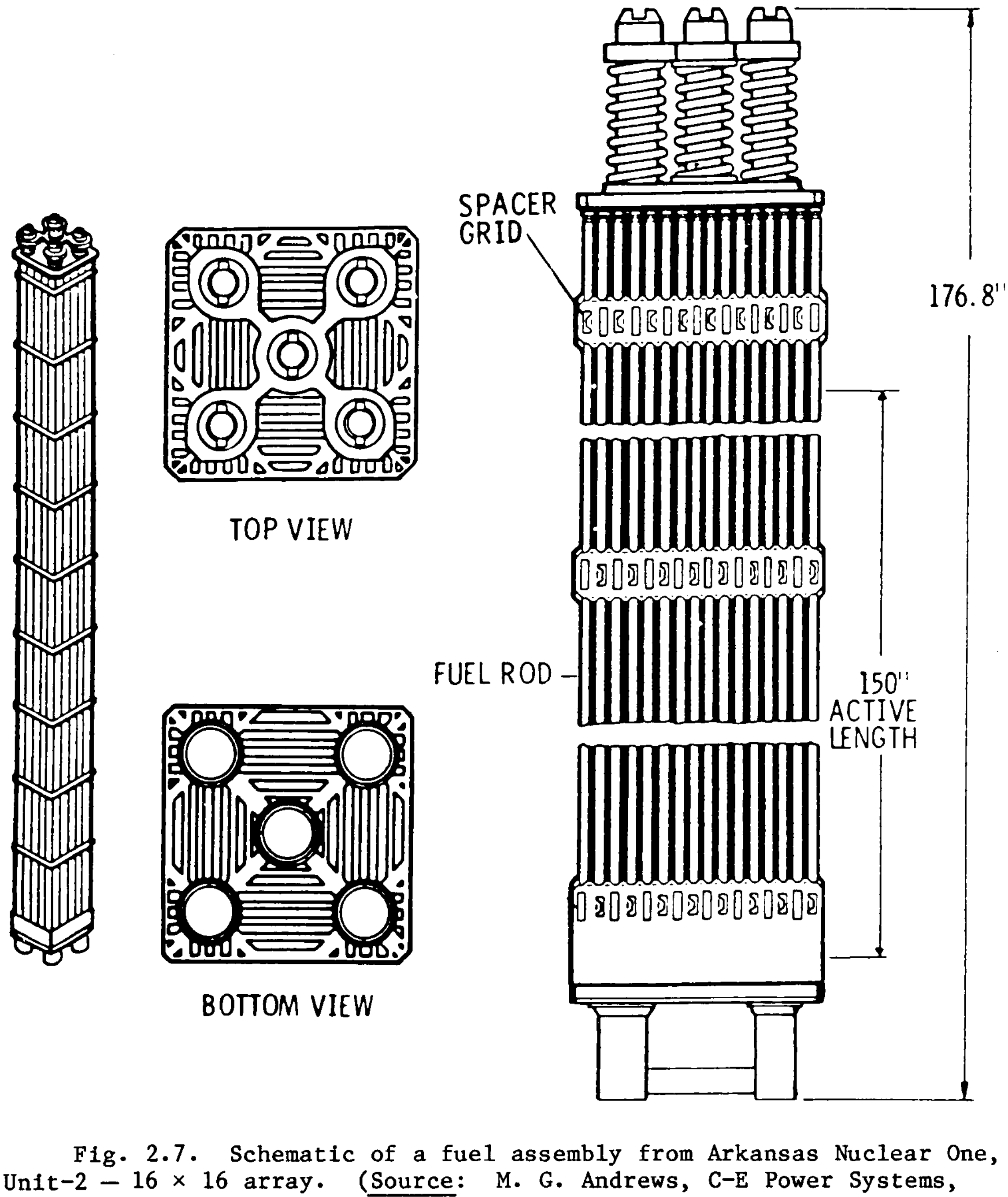

Fig. 2.7. Schematic of a fuel assembly from Arkansas Nuclear One, Unit-2 — 16 × 16 array. (Source: M. G. Andrews, C-E Power Systems, Combustion Engineering, Inc., letter to J. W. Roddy, Oak Ridge National Laboratory, February 11, 1985.)

ORNL DWG 85-984

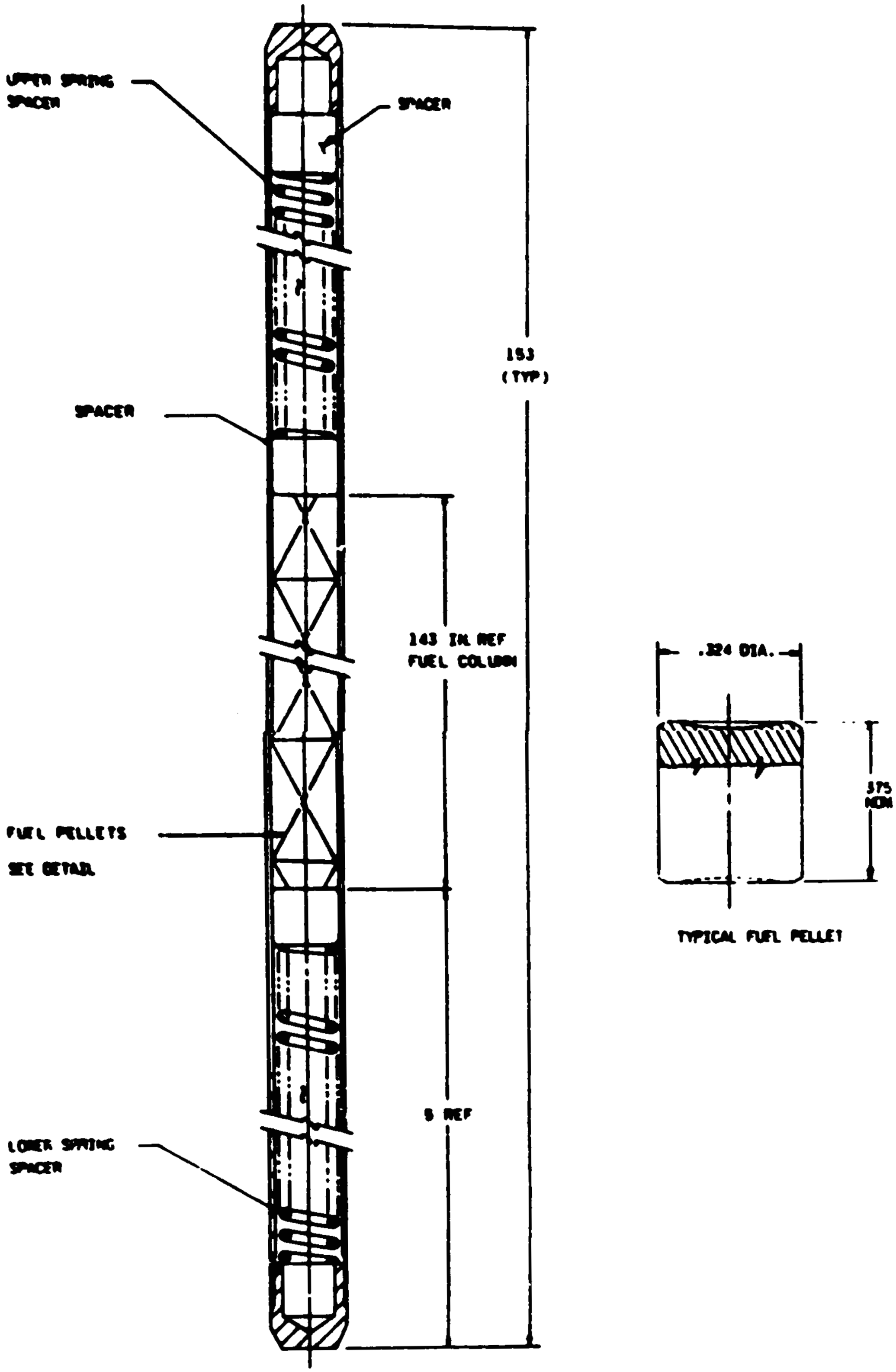

Fig. 2.8. Babcock and Wilcox fuel rod. (Source: E. M. Greene, Spent Fuel Data Base for Waste Storage Programs, HEDL-TME 79-20, September 1980.)

ORNL DWG 85-990

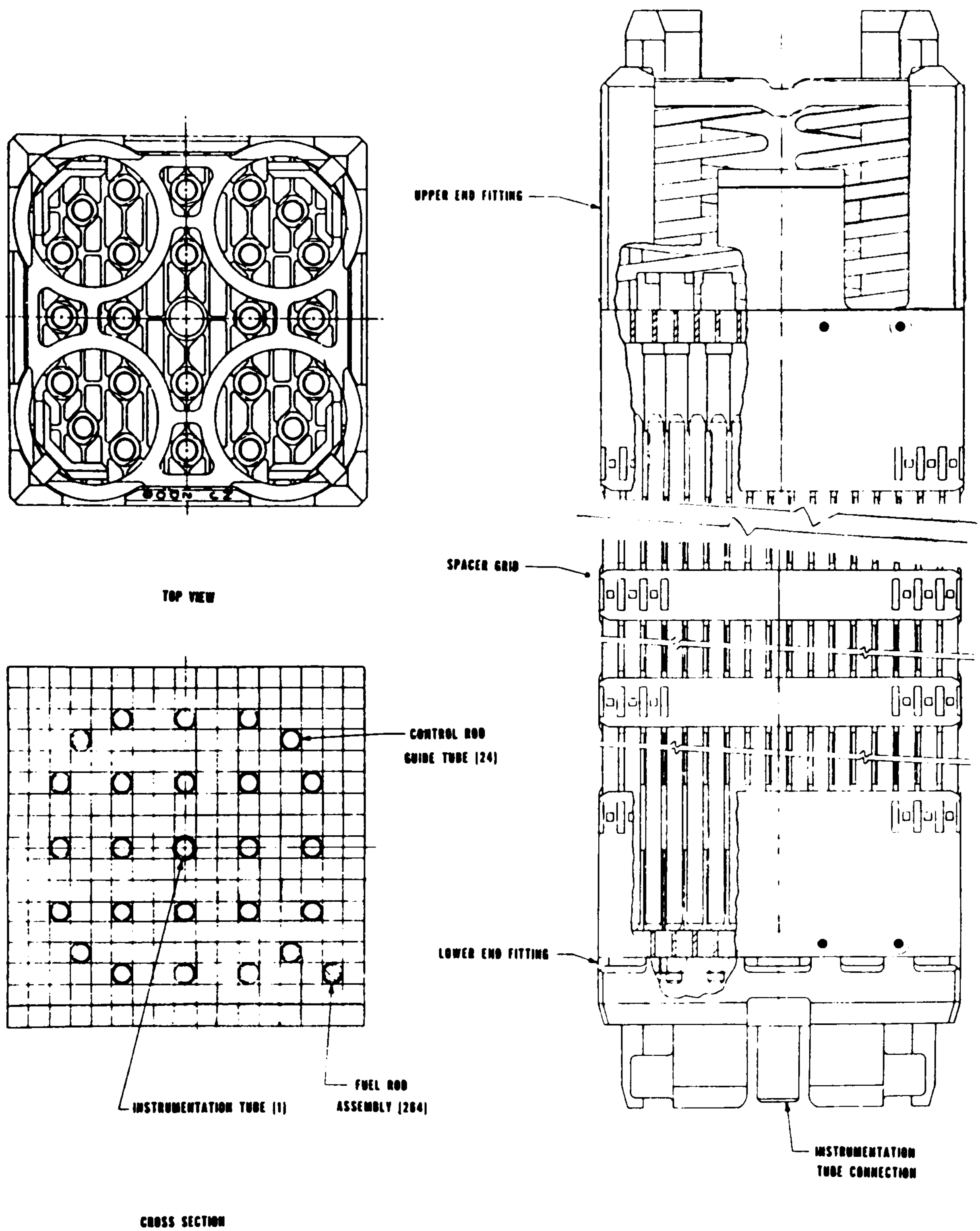

Fig. 2.9. Babcock and Wilcox fuel assembly. [Source: Babcock and Wilcox, Babcock and Wilcox Safety Analysis Report, Docket STN-50531 (B-SAR-205), February 1976.]

ORNL DWG 85-985

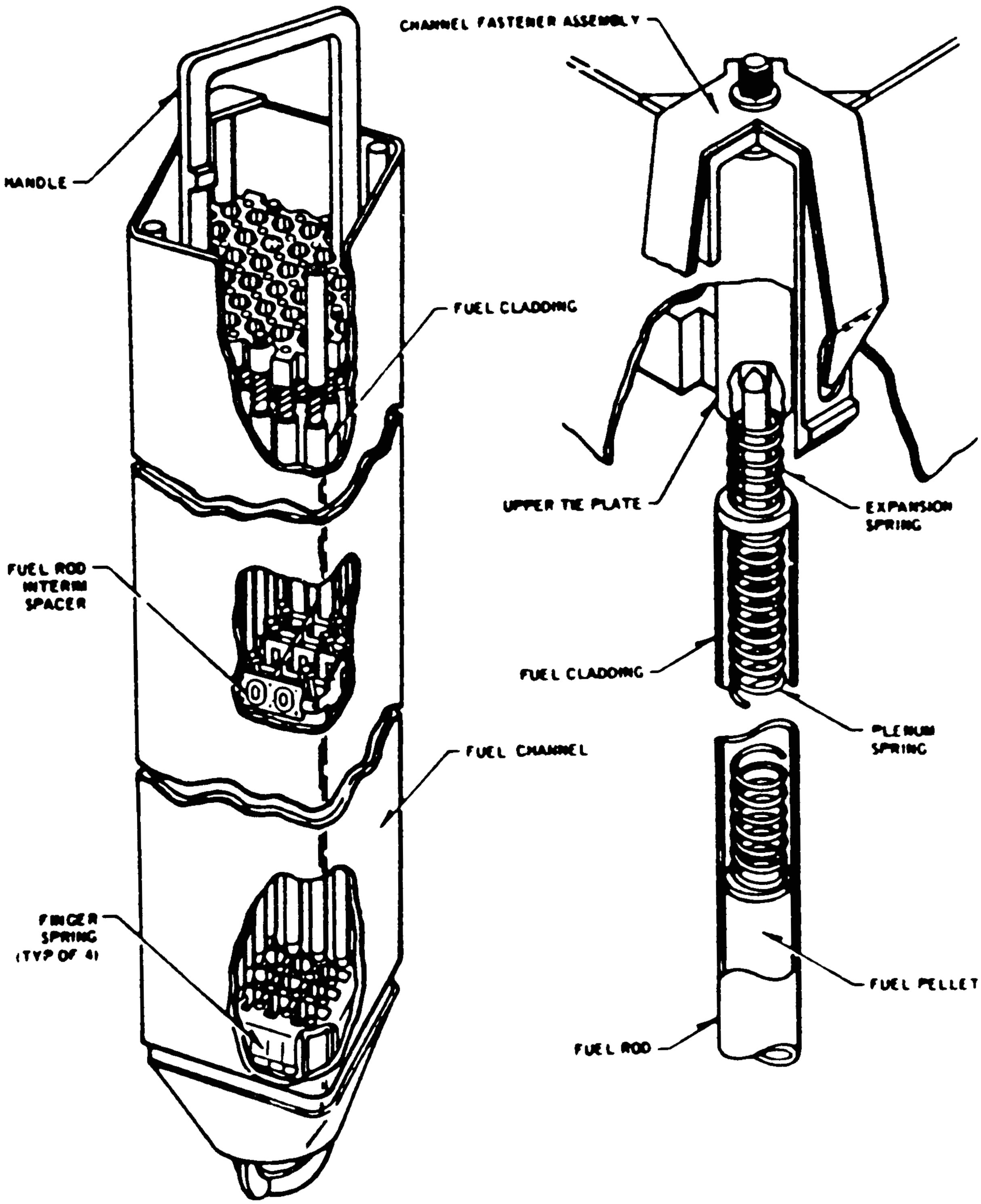

Fig. 2.10. Typical General Electric fuel rod and assembly. [Source: General Electric, General Electric Standard Safety Analysis Report, Docket STN-50531 (GESSAR-251), February 1975.]

ORNL DWG 85-983

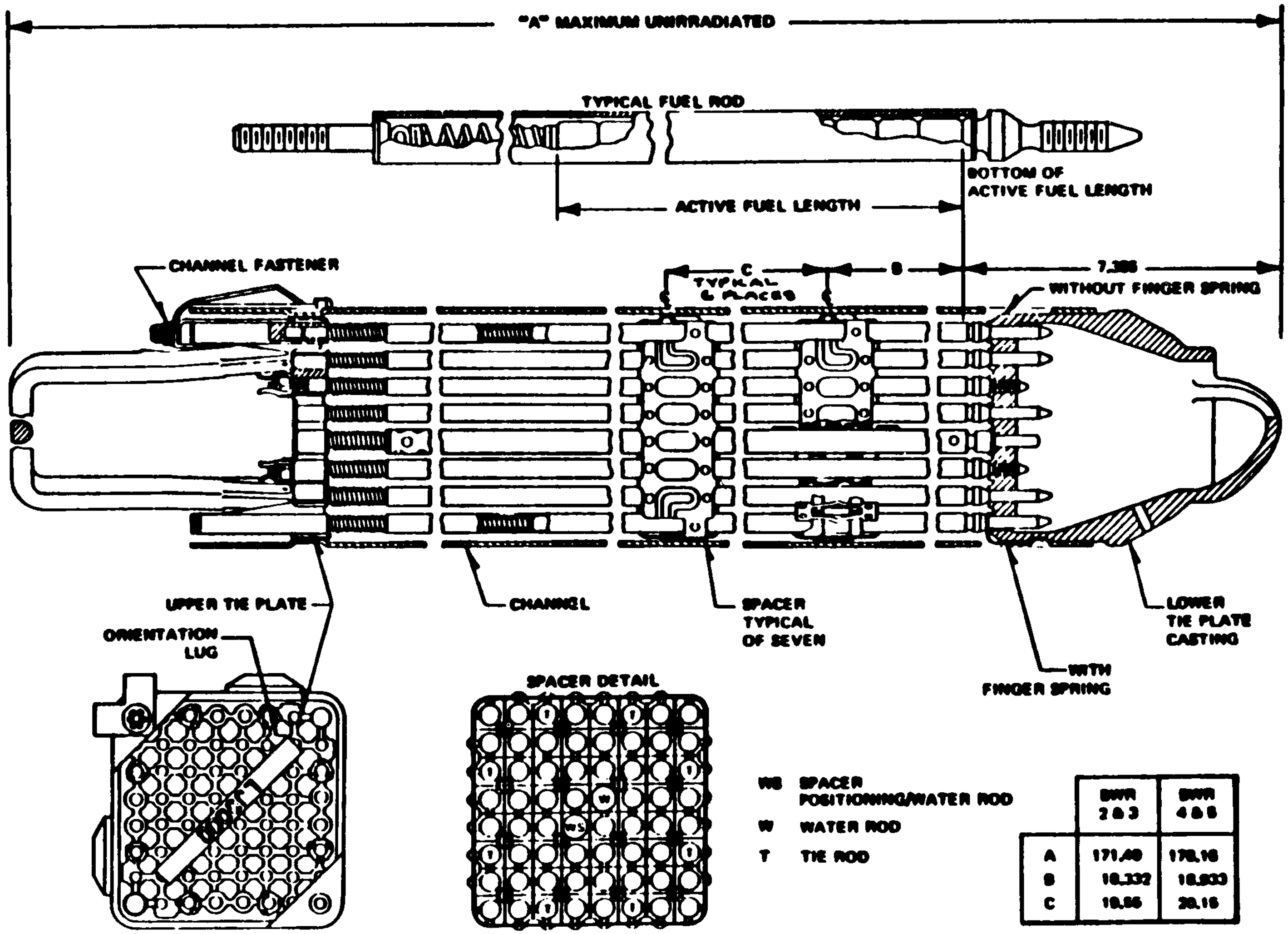

Fig. 2.11. Cutaway diagram of an 8 × 8 General Electric fuel assembly. (Source: R. A. Schmedt, Jr., General Electric Co., letter to J. W. Roddy, Oak Ridge National Laboratory, April 4, 1985.)

ORNL DWG 85-834

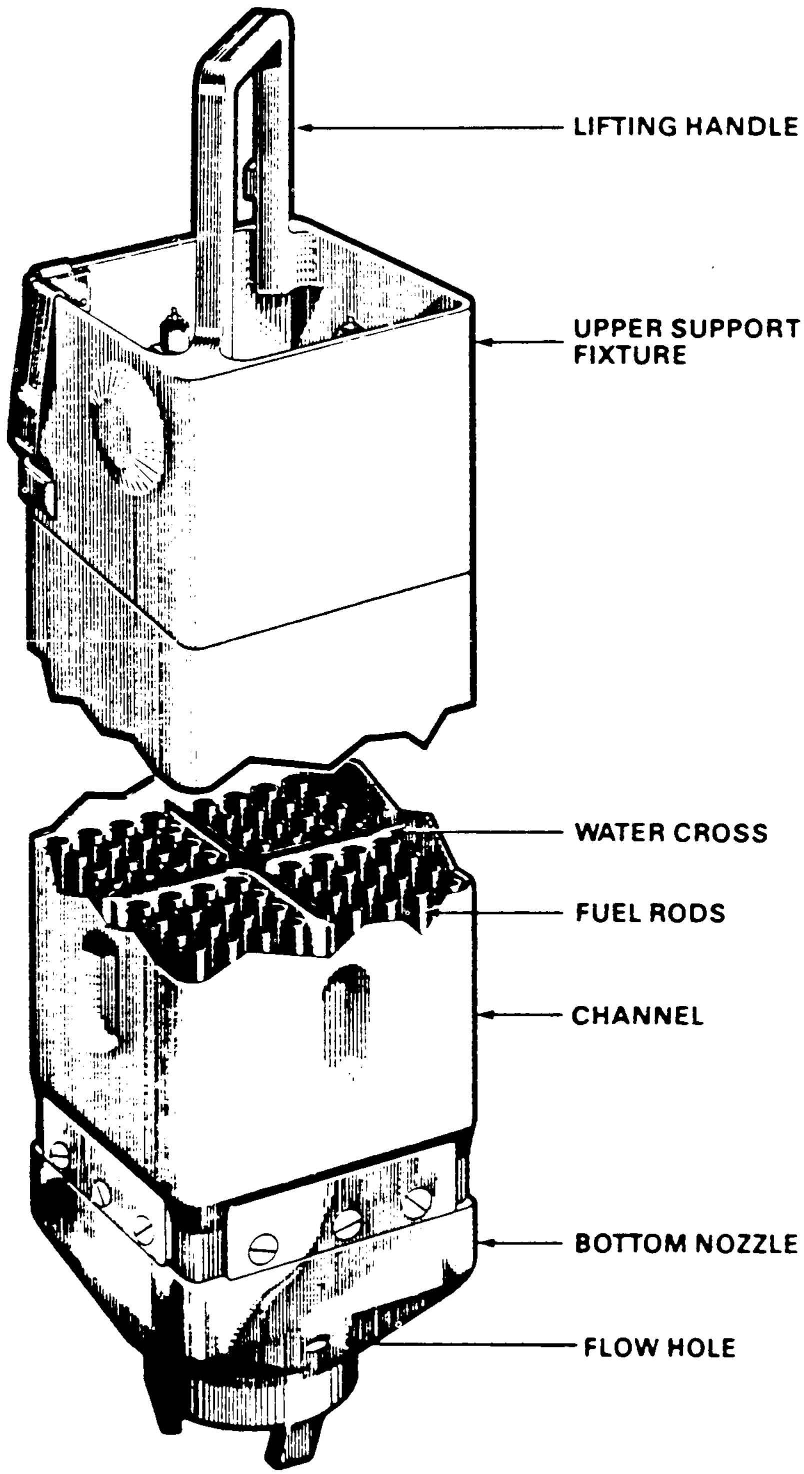

Fig. 2.12. Cutaway diagram of a QUAD+ fuel assembly. (Source: F. J. Frank, Westinghouse Electric Corporation, letter to H. C. Claiborne, Oak Ridge National Laboratory, July 18, 1985.)

ORNL DWG 85-833

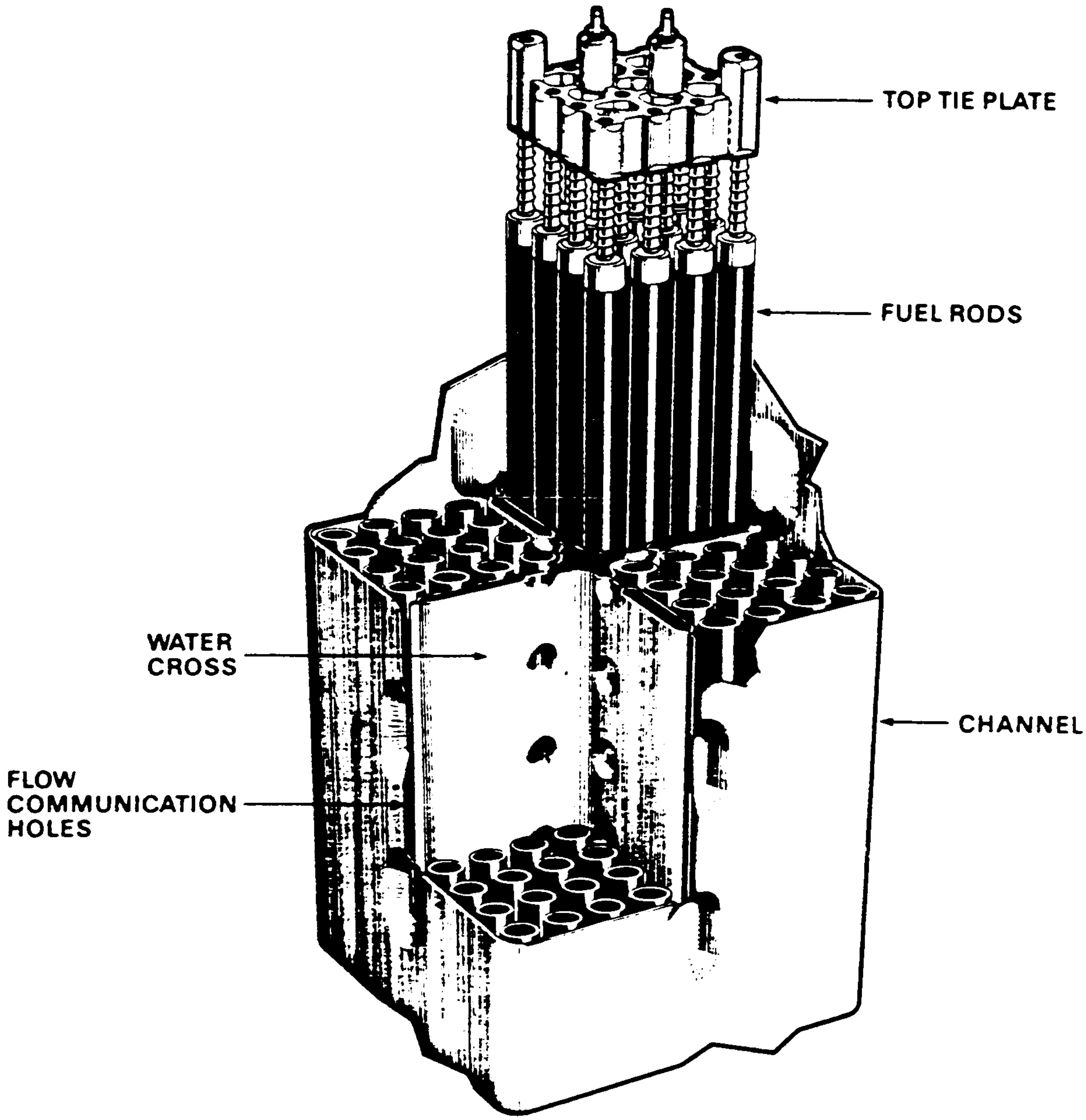

Fig. 2.13. Cutaway diagram of a partial QUAD+ fuel assembly showing internals. (Source: F. J. Frank, Westinghouse Electric Corporation, letter to H. C. Claiborne, Oak Ridge National Laboratory, July 18, 1985.)

ORNL DWG 85-835

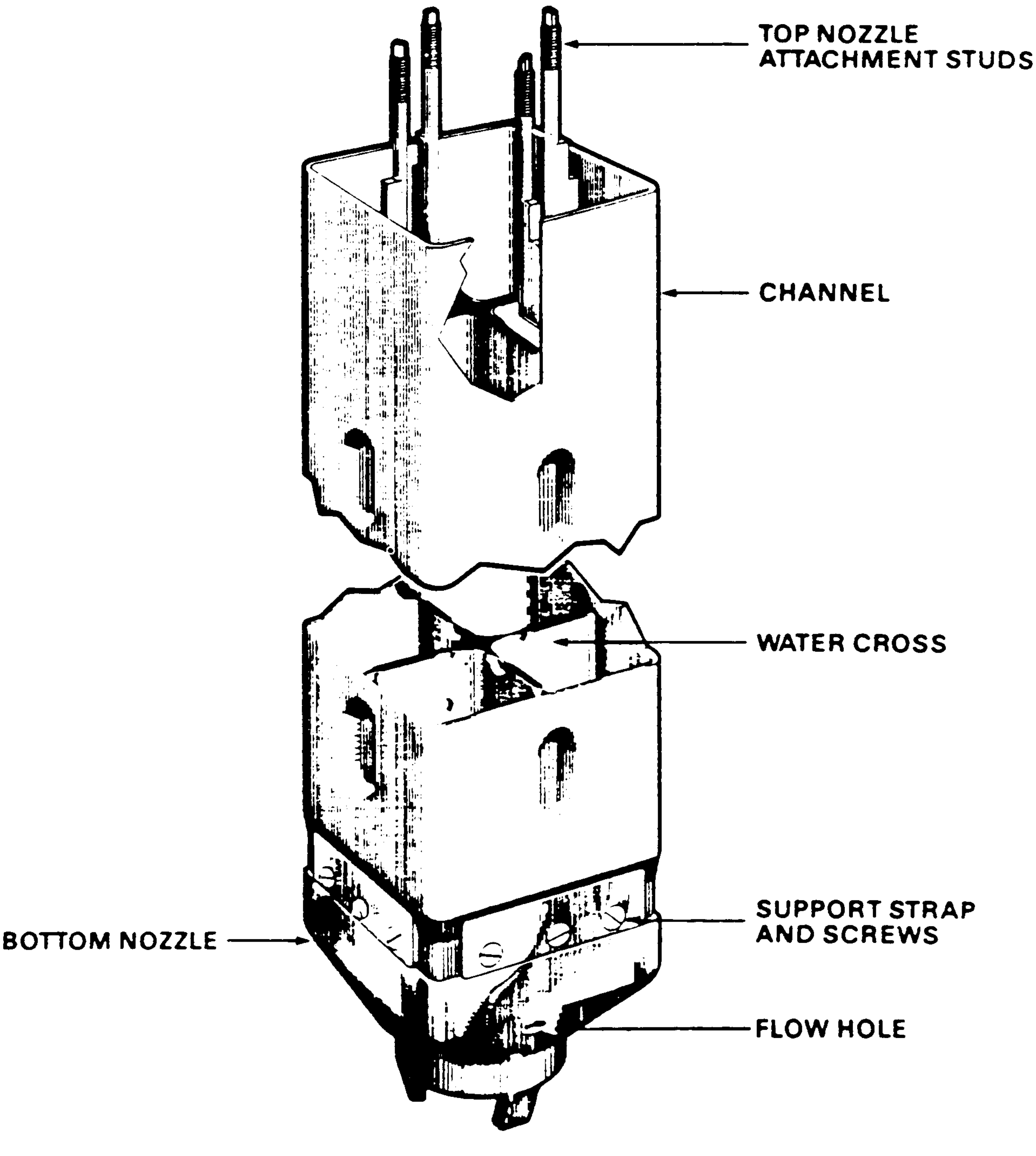

Fig. 2.14. Cutaway diagram of a QUAD+ fuel channel. (Source: F. J. Frank, Westinghouse Electric Corporation, letter to H. C. Claiborne, Oak Ridge National Laboratory, July 18, 1985.)

ORNL DWG 85-836

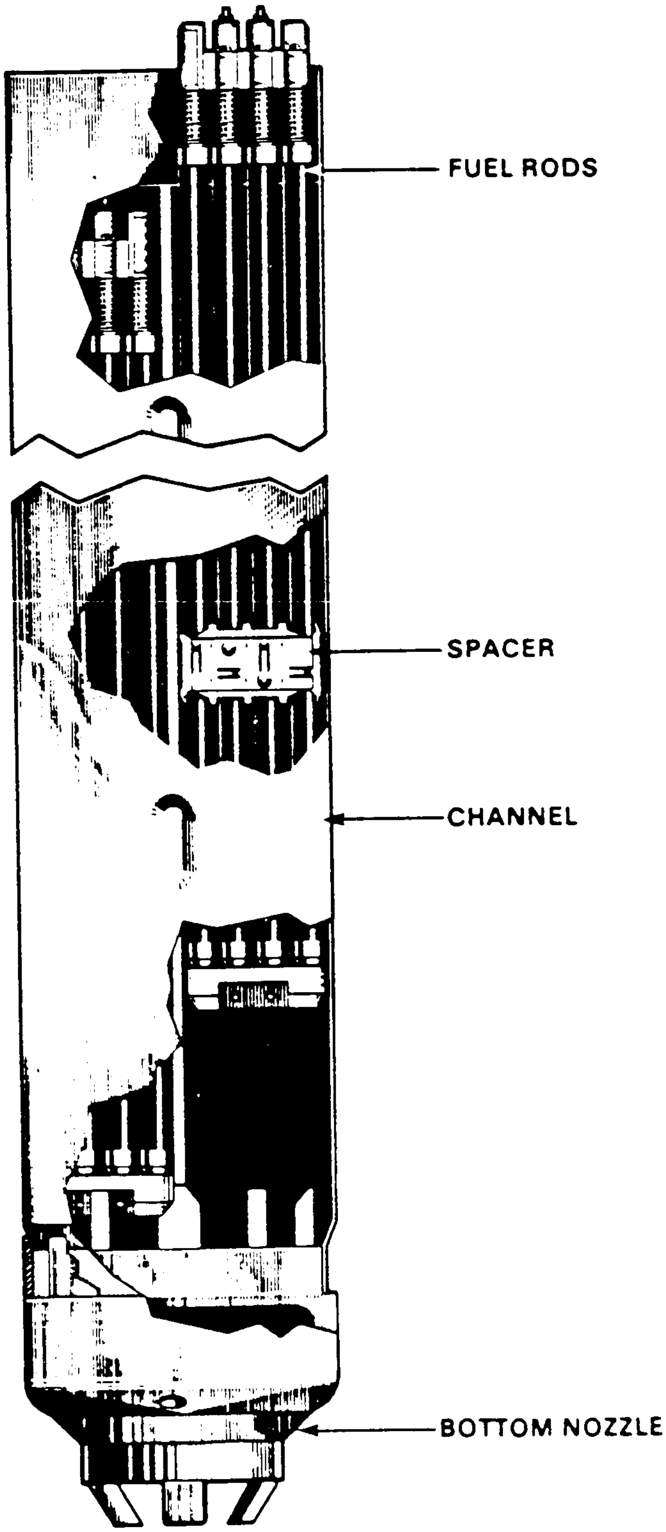

Fig. 2.15. Cutaway diagram of a QUAD+ fuel assembly with partially removed minibundle. (Source: F. J. Frank, Westinghouse Electric Corporation, letter to H. C. Claiborne, Oak Ridge National Laboratory, July 18, 1985.)

ORNL DWG 85-837

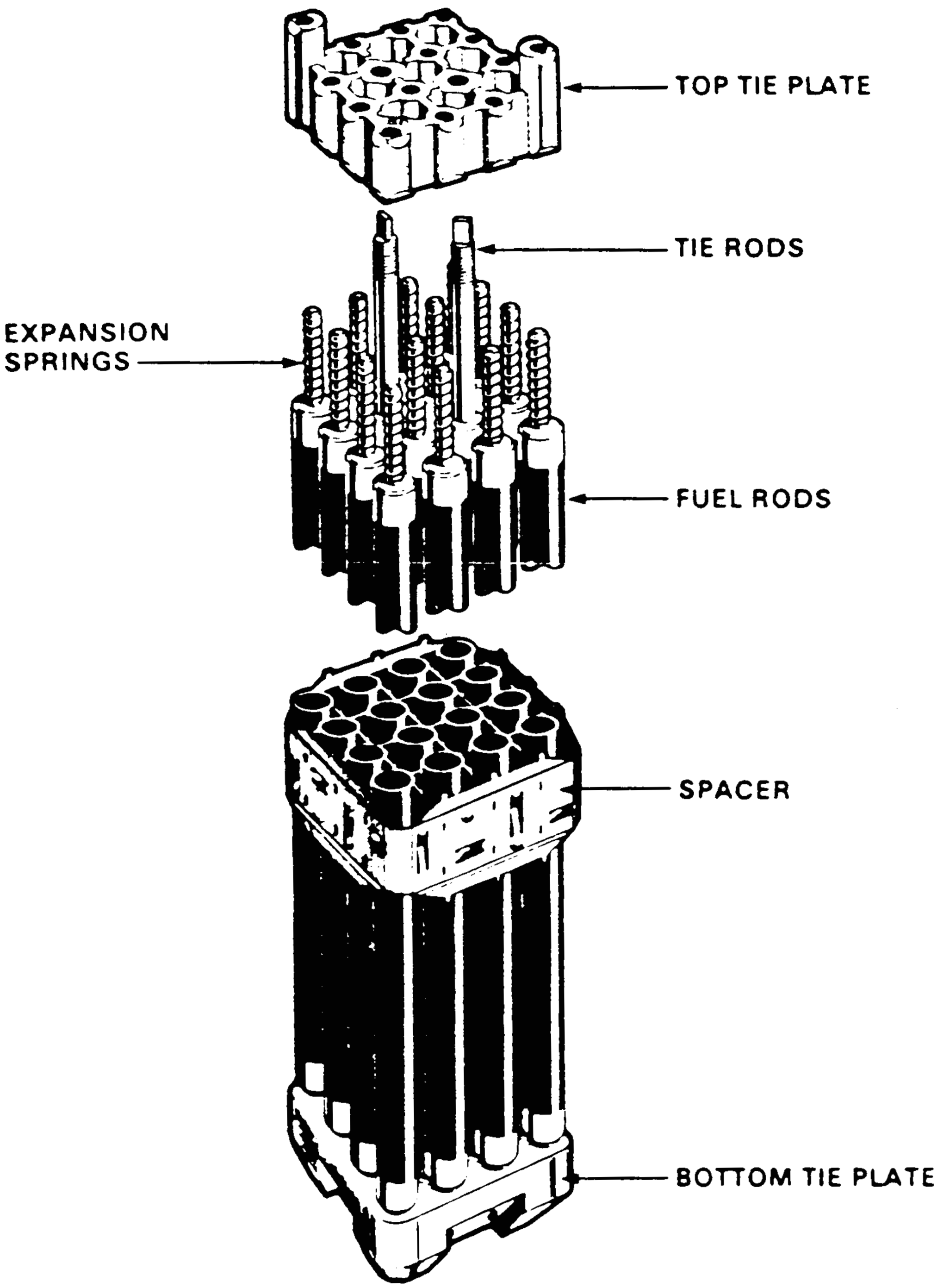

Fig. 2.16. Cutaway diagram of a QUAD+ fuel minibundle. (Source: F. J. Frank, Westinghouse Electric Corporation, letter to H. C. Claiborne, Oak Ridge National Laboratory, July 18, 1985.)

Table 2.1. Fuel assembly materials[a]

Design component	Subcomponent	Alloy or material
Fuel pellets		Uranium dioxide
Fuel rods		Zircaloy-2 (BWR) Zircaloy-4 (PWR)
		304 SS, 348H
Fuel spacers	Grid	304 SS
		Inconel 718
		Zircaloy-4
	Springs	Inconel 718, 625
		Zircaloy-4
Upper tie plates	Bail/tie plate	304 SS
	Bolts/nuts	304 SS
		Inconel 600
	Springs	Inconel 718, X750
Lower tie plates	Tie plate/nozzle	304 SS, CF-8
Tie rods		Zircaloy-4 304 SS

[a]Source: E. M. Greene, Spent Fuel Data for Waste Storage Programs, HEDL-TME 79-20, September 1980.

Table 2.2. Mechanical design parameters for Westinghouse PWR fuel assemblies[a]

Design component	Rod array: 17 × 17 Standard	17 × 17 OFA	17 × 17 VANTAGE	15 × 15 Standard	15 × 15 OFA	14 × 14 Standard	14 × 14 OFA	16 × 16[b] Standard
Assembly								
Transverse dimension, in.	8.426	8.426	8.426	8.426	8.426	7.763	7.763	7.763
Assembly weight, lb	1467	1365	1365	1440	1443	1274	1135	1310
Uranium/assembly, lb	1017.23	932.61	932.61	1011.86	1011.86	887.77	786.61	905.32
UO_2/assembly, lb	1154.00	1058.00	1058.00	1147.90	1147.90	1007.14	892.37	1027.04
Overall length, in.	159.8	159.8	160.1	159.765	159.765	159.71	159.71	159.8
Rod replacement capabilities	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Disassembly capabilities	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Fuel rods								
Date of commercial operation	1975	1984	1987	1967	1983	1969	1984	1981
Number per assembly	264	264	264	204	204	179	179	235
Rod pitch, in.	0.496	0.496	0.496	0.563	0.563	0.556	0.556	0.485
Length, in.	151.635	151.635	152.3	151.83	151.83	151.83	151.83	151.64
Fuel length, in.	144	144	144	144	144	144	144	144
OD, in.	0.374	0.36	0.36	0.422	0.422	0.422	0.400	0.374
Diametral gap, in.	0.0065	0.0062	0.0065	0.0075	0.0075	0.0075	0.0070	0.0065
Clad thickness, in.	0.0225	0.0225	0.0225	0.0243	0.0243	0.0243	0.0243	0.0225
Clad material	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4
Fuel pellets								
Type	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Density, % TD	95	95	95	95	95	95	95	95
Diameter, in.	0.3225	0.3088	0.3088	0.3659	0.3695	0.3659	0.3444	0.3225
Length, in.	0.53	0.51	0.51	0.60	0.60	0.60	0.565	0.53
Total weight/rod, lb	4.37	4.01	4.01	5.52	5.52	5.63	4.99	4.37
Spacer pellets	None	None	None	None	None	None	None	None
Plenum spring								
Working length, in.	6.90	6.90	7.405	7.136	7.136	7.136	7.158	6.90
Material	SS	SS	SS	SS	SS	SS	SS	SS
Miscellaneous								
Prepressurization, atm	Variable	Variable	Variable	Variable	Variable	Variable	Variable	Variable
Gas used	Helium	Helium	Helium	Helium	Helium	Helium	Helium	Helium
Spacer grids								
Top and bottom grids								
Number/assembly	2	2	2	2	2	2	2	2
Material	Inconel 718	Inconel 718	Inconel 718	Inconel 718	Inconel 718	Inconel 718	Inconel 718	Inconel 718
Intermediate grids								
Number/assembly	6	6	6	5	5	5	5	6
Material	Inconel 718	Zr-4	Zr-4	Inconel 718	Zr-4	Inconel 718	Zr-4	Inconel 718
Intermediate flow mixer	None	None		None	None	None	None	None
Number/assembly			3					
Material			Zr-4					
Guide tubes								
Number/assembly	24	24	24	20	20	16	16	20
OD, in.	0.474	0.474	0.474	0.546	0.532	0.539	0.527	0.471
Wall thickness, in.	0.016	0.016	0.016	0.017	0.017	0.017	0.017	0.016
Material	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4
Instrument tube								
Number/assembly	1	1	1	1	1	1	1	1
OD, in.	0.48	0.476	0.476	0.546	0.533	0.422	0.4019	0.473
Material	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4
Top and bottom nozzles material	SS	SS	SS	SS	SS	SS	SS	SS
Approximate no. of assemblies shipped by Westinghouse	6000	800	0	5200	400	4000	30	400

[a]*Source:* *L. Iyengar, Westinghouse Electric Corporation, letter to J. W. Roddy, Oak Ridge National Laboratory, December 17, 1984.*
[b]All of these assemblies have been exported.

Table 2.3. Mechanical design parameters for Combustion Engineering PWR fuel assemblies[a]

Design component	Rod array 14 × 14R	Rod array 16 × 16
Fuel assemblies		
Width dimension, in.	8.12	8.23
Assembly weight, lb (typical)	1204	1435
Overall length, in. (typical)	157	177
Rod replacement capabilities	Yes	Yes
Disassembly capabilities	Yes	Yes
Fuel rods		
Date of introduction (first criticality)	11/3/72	12/6/78
Number per assembly (unshimmed)	176	236
Rod pitch, in.	0.580	0.5063
Rod length, in. (typical)	146	161
Active fuel length, in.	136.7	150
OD, in.	0.440	0.382
Diametral gap, in.	0.0075	0.0070
Clad thickness, in.	0.028	0.025
Clad material (composition)	Zircaloy-4	Zircaloy-4
Total weight/rod, lb	6.7	5.7
Fuel pellets		
Density, % theoretical	95	95
Diameter, in.	0.3765	0.325
Length, in.	0.450	0.390
Total weight/rod, lb	5.4	4.5
Guide tubes[b]		
Number	5	5
OD, in	1.115	0.980
Wall thickness, in.	0.040	0.040
Tie plate		
Material	304 SS	304 SS
Total weight/assembly, lb	NA[c]	NA
Spacers		
Number (top and bottom)	2	2
Material (composition)	Al_2O_3	Al_2O_3
Total weight/rod, lb	0.004	0.005
Plenum springs		
Working length, in.	8.6	10.0
Material (composition)	SS	SS
Total weight/rod, lb	0.05	0.07
Miscellaneous		
Prepressurized to atm (typical)	Variable	Variable
Gas used	100% He	100% He

[a]Source: M. G. Andrews, C-E Power Systems, Combustion Engineering, Inc., letter to J. W. Roddy, Oak Ridge National Laboratory, February 11, 1985.

[b]Guide tubes may be used to guide the control rod assembly or to contain instrumentation which is located in the center guide tube.

[c]Not available.

Table 2.4. Number of Combustion Engineering PWR fuel assemblies active and discharged[a]

Reactor	Core assemblies per cycle	Total active and discharged		
		14 × 14	16 × 16	15 × 15
Arkansas Nuclear One-2	177	-	345	-
Calvert Cliffs 1	217	693	-	-
Calvert Cliffs 2	217	609	-	-
Fort Calhoun	133	289	-	-
Maine Yankee	217	650	-	-
Millstone 2	217	361	-	-
Palisades	204	-	-	272
St. Lucie-1	217	497	-	-
St. Lucie-2	217	-	297	-
SONGS-2	217	-	217	-
SONGS-3	217	-	217	-

[a]Source: M. G. Andrews, C-E Power Systems, Combustion Engineering, Inc., letter to J. W. Roddy, Oak Ridge National Laboratory, February 11, 1985.

Table 2.5. Mechanical design parameters for Babcock and Wilcox PWR fuel assemblies[a]

Design component	Rod array		
	15 × 15	17 × 17	15 × 15 SS
Assembly			
Transverse dimension, in.	8.536	8.536	8.466
Assembly weight, lb	1515	1506	NA[b]
Overall length, in.	165-5/8	165-23/32	137.066 + .565 spring protrusion
Rod replacement capabilities	None	None	Grippable top end
Disassembly capabilities	None	None	Locking cups on upper nuts
Fuel rods			
Date of introduction	1971	1976	1976
Number per assembly	208	264	204
Rod pitch, in.	0.568	0.502	0.563
Length, in.	153.68	152.688	126.68
Fueled length, in.	141.8	143.0	120.5
OD, in.	0.430	0.379	0.422
Diametral gap, in.	0.0084	0.0078	0.0065
Clad thickness, in.	0.0265	0.0240	0.0165
Clad material	Zircaloy-4	Zircaloy-4	304 SS
Total weight/rod, lb	7.0	4.9	5.9
Fuel pellets			
Density, % TD	95	95	95
Diameter, in.	0.3686	0.3232	0.3825
Length, in.	0.600	0.375	0.458
Total weight/rod, lb	5.58	Unavailable	Unavailable
Guide tubes			
Number	16	24	20
OD, in.	0.530	0.564	0.543, upper 106.8 in. 0.479, lower 20.95 in.
Wall thickness, in.	0.016	0.0175	0.012
Weight/assembly with end plugs, lb	16.5	24	17
Material	Zircaloy-4	Zircaloy-4	304 SS
Instrument tubes			
Number	1	1	1
OD, in.	0.493	0.420	0.422
Material (composition)	Zircaloy-4	Zircaloy-4	304 SS
Total weight/assembly, lb	0.7	0.7	0.78
Tie plate			
Material	NA	NA	NA
Spacers			
Number	3	3	--
Material (composition)	Zircaloy-4	Zircaloy-4	--
Total weight/rod, lb	.028	Unavailable	--
Plenum springs			
Working length, in.	7.435	5.9735	5.01
Material (composition)	302 SS	302 SS	302 SS
Total weight/rod, lb	Unavailable	Unavailable	Unavailable
Miscellaneous			
Prepressurized to, psig	465	435	40
Gas used	Helium	Helium	Helium

[a]Source: K. O. Stein, Nuclear Power Division, Babcock and Wilcox, letter to J. W. Roddy, Oak Ridge National Laboratory, January 25, 1985.

[b]Not available.

Table 2.6. Control and burnable poison rods in PWRs used by Babcock and Wilcox[a]

Design component	Rod array: Standard 15 × 15	Rod array: Long life 15 × 15	Rod array: 17 × 17
Control rod assembly			
Clad material	304 SS	UNS N06625[b]	304 SS
Clad length, in.	145.5	147.5	148-7/8
Clad OD, in.	0.440	0.441	0.377
Clad ID, in.	0.398	0.396	0.310
Pellet material	Ag-In-Cd	Ag-In-Cd	B_4C
Pellet OD, in.	0.392	0.386	0.285
Prepressure	1 atm He	465 psig He	1 atm He
Plenum volume, $in.^3$	0.4214	--	0.7075
Assembly weight, lb	130	130	65
Pellet stack length, in.	134	139	139
Burnable poison rod assembly			
Clad material	Zircaloy-4		Zircaloy-4
Clad length, in.	147-1/4		148
Clad OD, in.	0.430		0.371
Clad ID, in.	0.360		0.309
Pellet material	Al_2O_3—B_4C		Al_2O_3—B_4C
Pellet OD, in.	0.340		0.293
Prepressure	1 atm He		1 atm He
Plenum volume, $in.^3$	0.840		0.8774
Assembly weight, lb	57		60
Pellet stack length, in.	126		126

[a]Source: K. O. Stein, Utility Power Generation Division, Babcock and Wilcox, letter to J. W. Roddy, Oak Ridge National Laboratory, January 25, 1985.

[b]NiCrMoCb alloy.

Table 2.7. Number of PWR fuel assemblies shipped by Babcock and Wilcox[a]

Reactor	Rod array: 15 × 15	17 × 17	15 × 15 SS
Oconee 1	646	—	—
Oconee 2	533	2 MkC 2 MkCR	—
Oconee 3	521	—	—
ANO-1 Unit 1	493	—	—
Rancho Seco	432	—	—
Davis Besse	317	—	—
Crystal River	437	—	—
TMI-1	385	—	—
Conn Yankee	—	—	368
TVA Bellefonte I	—	205	—
TVA Bellefonte II	—	205	—

[a]Source: K. O. Stein, Nuclear Power Division, Babcock and Wilcox, letter to J. W. Roddy, Oak Ridge National Laboratory, January 25, 1985.

Table 2.8. Mechanical design parameters for Exxon Nuclear PWR fuel assemblies[a]

Design component	Rod array 14 × 14	15 × 15	17 × 17	14 × 14[b]
Assembly				
Transverse dimension, in.	7.763	8.426	8.426	8.105
Assembly weight, lb	NA[c]	1425	NA	1280
Overall length, in.	162	162	162	157
Rod replacement capability	Yes	Yes	Yes	Yes
Disassembly capability	Yes	Yes	Yes	Yes
Fuel rods				
Number per assembly	179	204	264	176
Rod pitch, in.	0.556	0.563	0.496	0.580
Length, in.	152	152	152	147
Fueled length, in.	144	144	144	137
OD, in.	0.417/0.424	0.424	0.360/0.376	0.440
Diametral gap, in.				
Clad thickness, in.	0.0295/0.030	0.030	0.025/0.024	0.031
Clad material	Zr-4	Zr-4	Zr-4	Zr-4
Total weight/rod, lb	NA	NA	NA	NA
Fuel pellets				
Type	UO_2	UO_2	UO_2	UO_2
Density, % TD	94	94	94	94
Diameter, in.	0.3505/0.3565	0.3565	0.303/0.321	0.370
Length, in.	NA	NA	NA	NA
Total weight/rod, lb	NA	NA	NA	NA
Spacers				
Number	7	7	7	7
Material	Zr-4/Inconel-718	Zr-4/Inconel-718	Zr-4/Inconel-718	Zr-4/Inconel-718
Total weight/rod, lb	2–3	2–3	2–3	2–3
Plenum springs				
Working length, in.	NA	NA	NA	NA
Material	Inconel-718	Inconel-718	Inconel-718	Inconel-718
Miscellaneous				
Prepressurization, atm	>20	>20	>20	>20
Gas used	Helium	Helium	Helium	Helium
Guide tubes				
Number	16	20	24	5
OD, in.	0.541	0.544	0.480	1.115
Wall thickness, in.	0.017	0.0165	0.016	0.040
Material	Zr-4	Zr-4	Zr-4	Zr-4
Instrument tubes				
Number	1	1	1	NA
OD, in.	NA	NA	NA	NA
Material	Zr-4	Zr-4	Zr-4	Zr-4
Tie plate				
Material	SS 304L, Inconel springs	SS 304L, Inconel springs	SS 304L, Inconel springs	SS 304L, Inconel springs
Total weight/assembly, lb	25	25	25	25

[a]Source: G. J. Busselman, Exxon Nuclear Company, Inc., letter to J. W. Roddy, Oak Ridge National Laboratory, March 28, 1985.

[b]Produced only for Combustion Engineering.

[c]Not available.

Table 2.9. General Electric BWR product lines and characteristics[a]

Product line class	Year of introduction	Plants and characteristics
BWR/1	1955	Dresden-1, Big Rock Point, Humboldt Bay, KRB — Initial commercial BWRs — First internal steam separation
BWR/2	1963	Oyster Creek — The first turnkey plant — Elimination of dual cycle
BWR/3	1965	Dresden-2 — The first jet pump application — Improved emergency core cooling system (ECCS)
BWR/4	1966	Browns Ferry — Increased power density 20%
BWR/5	1969	Zimmer — Improved safeguards — Valve flow control
BWR/6	1972	BWR/6 — 8 × 8 fuel bundle — Added fuel bundles, increased output — Improved recirculation system performance — Improved ECCS performance — Reduced fuel duty

[a]Source: E. D. Fuller, J. R. Finney, and H. E. Streeter, BWR/6 Nuclear System from General Electric — A Performance Description, NEDO-10569A, April 1972.

Table 2.10. Summary of General Electric BWR reactor fuel designs[a]

Design component	Rod array				
	7 × 7		7 × 7R	8 × 8	8 × 8R
Introduction date	1966	1968	1972	1973	1977
Fuel rod OD, in.	0.563	0.570	0.563	0.493	0.483
Fuel rod ID, in.	0.499		0.0489	0.425	0.419
Nominal cladding thickness, mil	32	35.5	37	34	32
Nominal diametral gap, mil	11	12	12	9	9
Pellet type	Long, sharp corners		Short, chamfered		
Hydrogen getter	No		Yes	Yes	Yes
Peak liner power, W/cm	607		607	440	440
Prepressurized to 3 atm	No		No	No	Yes

[a]Source: R. E. Woodley, The Characteristics of Spent LWR Fuel Relevant to Its Storage in Geologic Formations, HEDL-TME 83-28, October 1983.

Table 2.11. Mechanical design parameters for BWR fuel assemblies[a]

Design component	Rod array: BWR/1–5 (General Electric) 7 × 7	BWR/1–5 (General Electric) 8 × 8	QUAD+ (Westinghouse) 8 × 8
Fuel assemblies			
Transverse dimension, in.	5.518	5.518	5.50
Assembly weight, lb	600	600	600
Overall assembly length, in.	171.2	171.2–178.5	175.5
Fuel rods			
Number per assembly	49	62–63	64
Rod pitch, in.	0.738	0.640	0.609
Length, in.	161.1	161.1	160.6
Fueled length, in.	144–146	144–146	150
OD, in.	0.563–0.570	0.483–0.493	0.458
Diametral gap, in.	0.011–0.012	0.009	0.083
Cladding thickness, in.	0.032–0.037	0.032–0.034	0.029
Cladding material	Zircaloy-2	Zircaloy-2	Zircaloy-2
Fuel pellets			
Density, % TD	95	95	95
Diameter, in.	0.487	0.416	0.3913
Length, in.	0.500	0.420	0.470
Tie plate			
Material	304 SS	304 SS	304 SS
Spacers			
Number	7	7	
Material	Zircaloy-4	Zircaloy-4	Zircaloy-4
Springs	Inconel	Inconel	Zircaloy-4
Plenum springs			
Working length, in.	10.6	10.6-16.0	9.56
Material	Inconel	Inconel	302 SS
Compression springs			
Working length, in.	0.94	0.84	0.84
Material	Inconel	Inconel	Inconel

[a]Source: R. E. Woodley, The Characteristics of Spent LWR Fuel Relevant to Its Storage in Geologic Formations, HEDL-TME 83-28, October 1983 and E. M. Greene, Spent Fuel Data for Waste Storage Programs, HEDL-TME 79-20, September 1980.

Table 2.12. Mechanical design parameters for Exxon Nuclear BWR fuel assemblies[a]

Design component	Replacement array BWR/1–5 7 × 7	BWR/2–6 8 × 8	New 9 × 9
Fuel assemblies			
Transverse dimension, in.	5.25	5.25	5.25
Assembly weight, lb	590	580	570
Overall length, in.	174	174	174
Rod replacement capability	Yes	Yes	Yes
Disassembly capability	Yes	Yes	Yes
Fuel rods			
Date of introduction	1971	1974	1981
Number per assembly	49	63	79
Rod pitch, in.	0.73	0.64	0.57
Length, in.	145	145	145
OD, in.	0.59	0.48	0.42
Diametral gap, in.		None at end of life	
Clad thickness, in.	0.03	0.03	0.03
Clad material	Zr-2	Zr-2	Zr-2
Total weight/rod, lb	12.0	9.0	7.0
Fuel pellets			
Type	UO_2	UO_2	UO_2
Density, % TD	94	94	94
Diameter, in.	0.49	0.40	0.36
Length, in.	NA[b]	NA	NA
Total weight/rod, lb	NA	7.0	5.7
Plenum springs			
Working length, in.	10	10	13
Material	Inconel	Inconel	Inconel
Total weight/rod, lb	0.09	0.09	0.09
Compression springs			
Working length, in.	0.8	0.9	1.3
Material	Inconel	Inconel	Inconel
Total weight/rod, lb	0.007	0.007	0.007
Tie plate			
Material	CF-3 (304L)	CF-3 (304L)	CF-3 (304L)
Weight, lb	12	12	12

[a]Source: G. J. Busselman, Exxon Nuclear Company, Inc., letter to J. W. Roddy, Oak Ridge National Laboratory, March 28, 1985.
[b]Not available.

Table 2.13. Mechanical design parameters for Allis-Chalmers BWR fuel assemblies[a]

Design component	Rod array (10 × 10)
Fuel assemblies	
Transverse dimension, in.	NA[b]
Assembly weight, lb	NA
Overall assembly length, in.	NA
Fuel rods	
Number per assembly	100
Rod pitch, in.	0.565
Length, in.	NA
Fueled length, in.	83
OD, in.	0.396
Diametral gap, in.	0.006
Cladding thickness, in.	0.020
Cladding material	348 H SS
Fuel pellets	
Density, % TD	95
Diameter, in.	0.350
Length, in.	0.350–1.050
Tie plate	
Material	304 SS
Spacers	
Number	NA
Material	NA
Springs	NA
Plenum springs	
Working length, in.	NA
Material	NA
Compression springs	
Working length, in.	NA
Material	NA

[a]Source: Allis-Chalmers, Initial Testing of the La Crosse Boiling Water Reactor, ACNP-67533, December 1967.

[b]Not available.

3. DEVELOPMENT OF ORIGEN2 REFERENCE CASES

Since the health and safety of the public must be protected for present and future generations, the long-term disposal of LWR spent fuel in a mined geologic repository requires specific knowledge concerning the effects of residence time on the radioactive components in discharged fuel assemblies. The relative youth (~40 years) of the nuclear industry hinders the use of current experimental data to quantify the consequences of long storage periods. However, a wide variety of computer models have been developed to make such projections.[1-3] ORIGEN2, a computer program that is available through the Radiation Shielding Information Center,[4] was used to generate the data presented in this section. The first version[5] was written in the late 1960s and early 1970s by staff members of the Chemical Technology Division of Oak Ridge National Laboratory (ORNL) and has been extensively revised and updated in the intervening years.[6-11] It is a versatile point depletion and decay calculational procedure for use in simulating nuclear fuel cycles and estimating the buildup and depletion of isotopes contained therein and has been used extensively in waste management studies. A comparison of the precision of ORIGEN2 in predicting the heat output and radioactivity with values measured experimentally for selected discharged fuel is given in Appendix C.

The present code[10] was used to model two reference LWRs, a PWR and a BWR, and results were obtained for several burnups (5000 MWd increments to 60,000 MWd for the PWR and 40,000 MWd for the BWR). Although the physical characteristics, structural material distribution, elemental composition, and initial analysis of oxide fuel vary slightly from vendor to vendor (Sect. 2), the values listed in Tables 3.1-3.4 are reasonable compromises and were used in this study. A tabulation of the composition (g), total radioactivity (Ci), and thermal power (W) for the significant nuclides covering 38 decay periods (from 1 to 1 million years) for MTIHM has been produced and placed on a series of diskettes for interrogation (see Appendix A for a sample search and output). For inclusion into the data base, an isotope had to contribute $>1 \times 10^{-25}$ mol.

For quick reference and to summarize supplementary information not included in the computer-searchable file, the authors have provided a series of tables (Tables 3.5-3.28) and figures (Figs. 3.1-3.13) for four specific burnups. A series of selected decay chains are given in Appendix B. The variation in radioactivity and heat output with burnup for four decay times is given in Figs. 3.14-3.17.

Three separate and distinct categories for radioactivity produced (Figs. 3.1-3.4) and heat generated (Figs. 3.6-3.9) have been included in the figures. The activation products include the low-Z impurities and structural material. The actinides include the heavy isotopes (Z>90), their decay daughters, and final stable nuclides. The fission products comprise all nuclides that have a significant fission product yield (binary or ternary) plus some nuclides resulting from neutron capture of the fission products. The tables list all isotopes that either contribute >0.1% to the total for each specified time since discharge or exhibit some unique attribute (long half-life, serious health hazard, etc.).[14-18] The variation in alpha radioactivity with decay is presented in Fig. 3.5 for the four cases. All values are predicated on the burnup occurring over an essentially continuous 3- to 4-year irradiation period.

3.1 MAJOR CONTRIBUTORS TO RADIOACTIVITY

Although the three categories display minor differences in the radioactivity for the various reactor types and burnups, there are major variations with decay time (Figs. 3.1-3.4 and Tables 3.5-3.12). The fission products dominate the total radioactivity for the first 100 years after storage; an interim period (100 to 300 years) occurs during which both fission products and actinides contribute to the total and ultimately the long-lived actinides control after 300 years.

The major contributors to the total radioactivity after one year of storage include: four separate decay chains, $^{90}Sr \rightarrow {}^{90}Y$, $^{106}Ru \rightarrow {}^{106}Rh$, $^{137}Cs \rightarrow {}^{137m}Ba$, and $^{144}Ce \rightarrow {}^{144}Pr$; one additional fission product, ^{134}Cs; and one actinide, ^{241}Pu. After 100 years, the total activity will have decreased by a factor of 40, with the fission products (^{90}Sr, ^{90}Y, ^{137}Cs, and ^{137m}Ba) supplying ~80% of the total. The long-lived actinides control the activity after 1 (>98%) and 10 (>94%) millennia, respectively. The predominate nuclides include: ^{239}Pu, ^{240}Pu, and ^{241}Am for the first period; and ^{239}Np, ^{239}Pu, ^{240}Pu, and ^{243}Am for the second period. Following extremely long storage (100,000 years), one major fission product, ^{99}Tc, one reactor-produced actinide, ^{239}Pu, and the naturally occurring radioactive isotopes present in the uranium decay chain generate the major quantities of radioactivity.

3.2 MAJOR CONTRIBUTORS TO THERMAL POWER

The heat generated by a fuel assembly is an important factor in the design of casks for storage and shipping and of repositories. As is the case for the total radioactivity, the thermal power generated by a discharged fuel assembly has initially, as its roots, the fission products (Figs. 3.6-3.9 and Tables 3.13-3.16). Following decay of the fission products for 60 to 70 years, the actinides reach an equivalent output. The contribution from the activation products is small and barely exceeds 2% of the total, and that occurs in the first decade.

The initial loadings (1 year) placed on a storage facility stem from three fission products, ^{106}Rh, ^{134}Cs, and ^{144}Pr, all of which exhibit short half-lives. The thermal power of spent fuel decreases by a factor of 6 after the initial 10 years of aging. The major sources of this power decrease are ^{90}Y, ^{137m}Ba, and ^{137}Cs for all cases plus ^{238}Pu and ^{244}Cm for extended-burnup PWRs. There is an additional power decrease by a factor of 5 after 100 years of cooling. The effects from fission products decrease significantly after discharge of fuel from a reactor and contribute 1% or less after ~300 years. During intermediate storage periods (100 to 1000 years), the actinide isotopes of importance are ^{238}Pu, ^{239}Pu, and ^{241}Am, while ^{239}Pu and ^{240}Pu control in the 10,000-year time frame, and ^{239}Pu is the major heat generator at 100,000 years.

3.3 NEUTRON SOURCES

There are essentially two mechanisms, spontaneous fission (Tables 3.17-3.20) and alpha interaction with an isotope (Tables 3.21-3.24), that generate neutrons in a discharged fuel assembly. Spontaneous fission (Figs. 3.10-3.13) produces >80% of the neutrons for all but the intermediate decay periods (100 to 1000 years) when its contribution is reduced to ~60%. The curium nuclides, ^{242}Cm and ^{244}Cm, dominate this production during the first 10 years, and the plutonium isotopes, the specific ones depending on the reactor type and burnup, are the major contributors in the 10,000- to 100,000-year time frame. With its nearly 400,000-year half-life, ^{242}Pu is the only isotope of consequence after 100,000 years of storage. A mixture of plutonium and curium nuclides and ^{241}Am produces the neutrons at 1000 years.

3.4 PHOTON PRODUCTION

The ORIGEN2 photon data base[8] supplies the number of photons per decay in an 18-energy-group structure. These values are used to output a table giving the number of photons and the photon energy emission rate in these energy groups as a function of irradiation and/or decay time (Tables 3.25-3.28). They are used to generate summary tables listing the principal nuclide contributions as a function of each energy group. The types of photons that have been included in the data bases are primary gamma rays, X-rays, conversion photons, (α,n) gamma rays, prompt and fission product gamma rays from spontaneous fission, and bremsstrahlung. Individual tables for each of the energy groups and the three separate categories (activation products, fission products, and actinides) have not been included; however, a brief synopsis of the major contributors and their mean energies will be discussed.

The number of photons produced by the activation products never exceeds >3% of the total occurring in the first decade and again at 10,000 years after discharge of fuel. As might be expected, ^{60}Co is the major photon producer, with minor contributions from ^{95}Zr, ^{95}Nb, and ^{54}Mn for the 5- and 10-year periods. Nickel-63 and ^{94}Nb are the chief nuclides after a century. Niobium-94 and ^{93}Zr are the only isotopes of consequence after 10 centuries, and ^{93}Zr is the only isotope at 100,000 years.

Several fission products produce photons in the first few years after fuel storage. Their percentage of total photon production drops from ~99% at 1 year to 90% at 100 years and ultimately to <1% at decay times >1000 years. The major γ-emitting isotopes at 1 year include ^{106}Rh, ^{144}Pr, and ^{134}Cs. After one decade, ^{90}Sr, ^{90}Y, and ^{137}Ba replace the previous nuclides, and ^{90}Sr, with its relatively short half-life, becomes inconsequential at 100 years.

The actinides and their daughters are relatively poor photon generators and exceed the output of the fission products only after ~200 years of storage. After 1000 years, two americium isotopes, ^{241}Am and ^{243}Am, and two plutonium isotopes, ^{239}Pu and ^{240}Pu, predominate in varying amounts up to the maximum storage period of 100,000 years.

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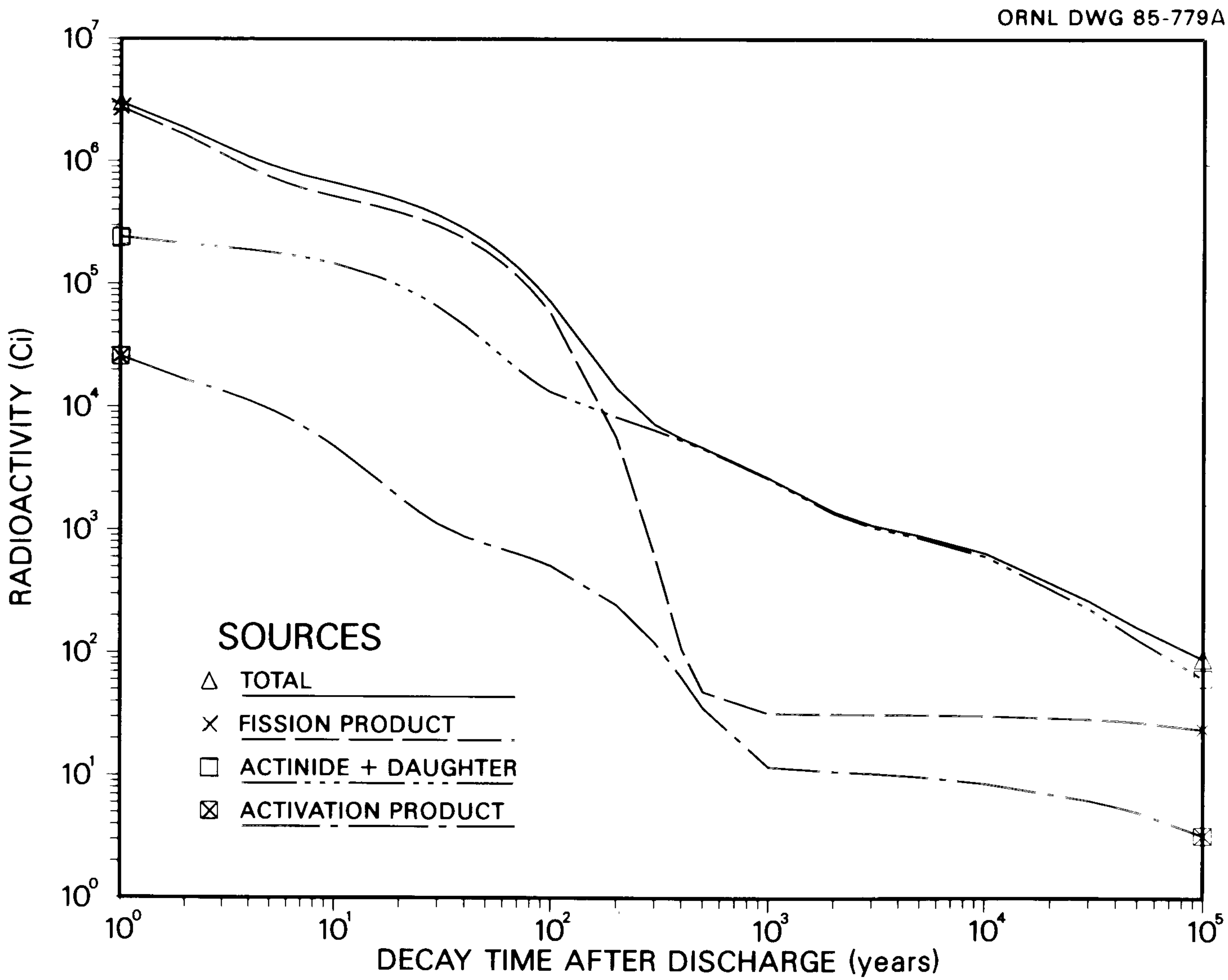


Fig. 3.1. Radioactivity produced by 1 metric ton of initial heavy metal: PWR; 60,000 MWd.

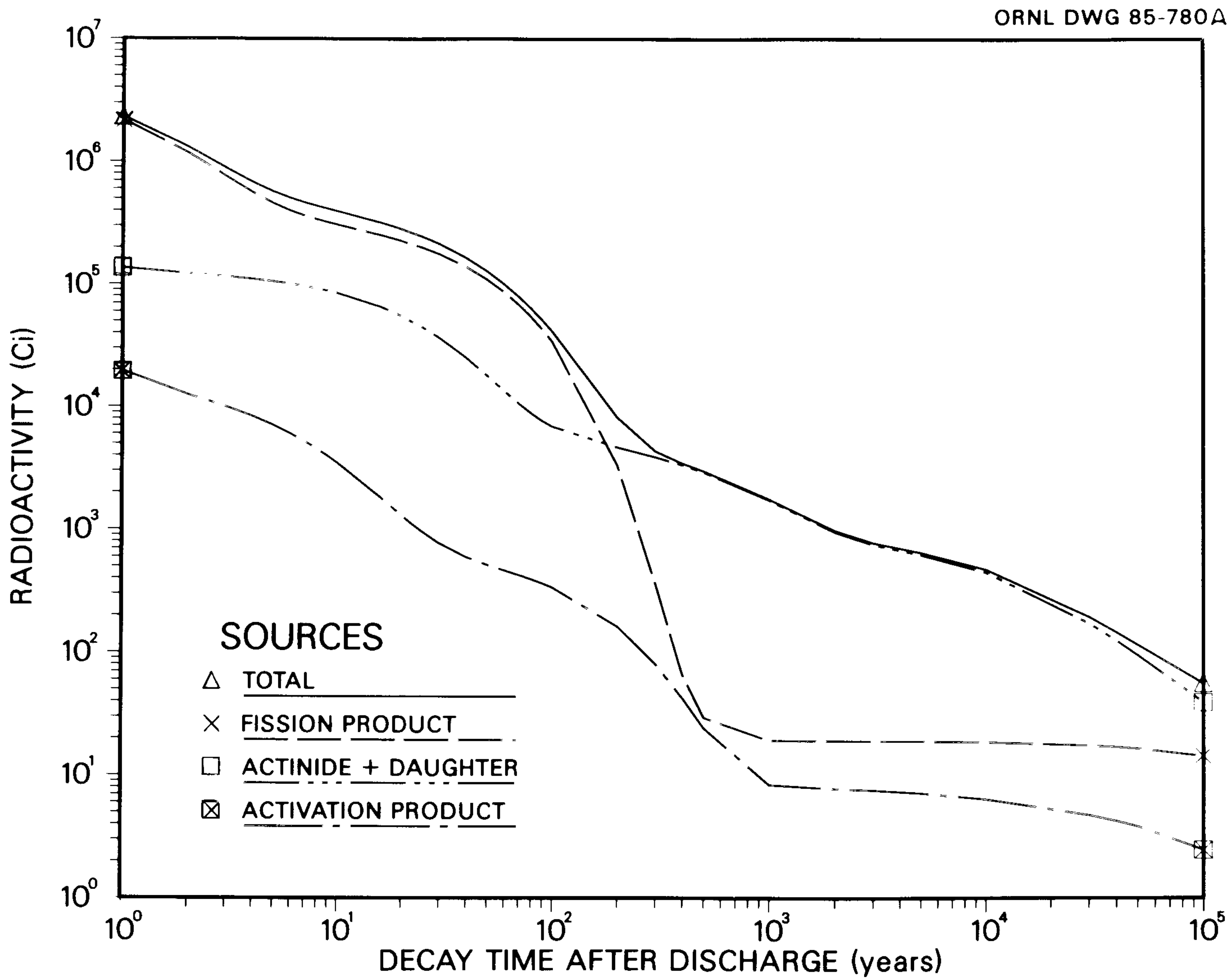


Fig. 3.2. Radioactivity produced by 1 metric ton of initial heavy metal: PWR; 33,000 MWd.

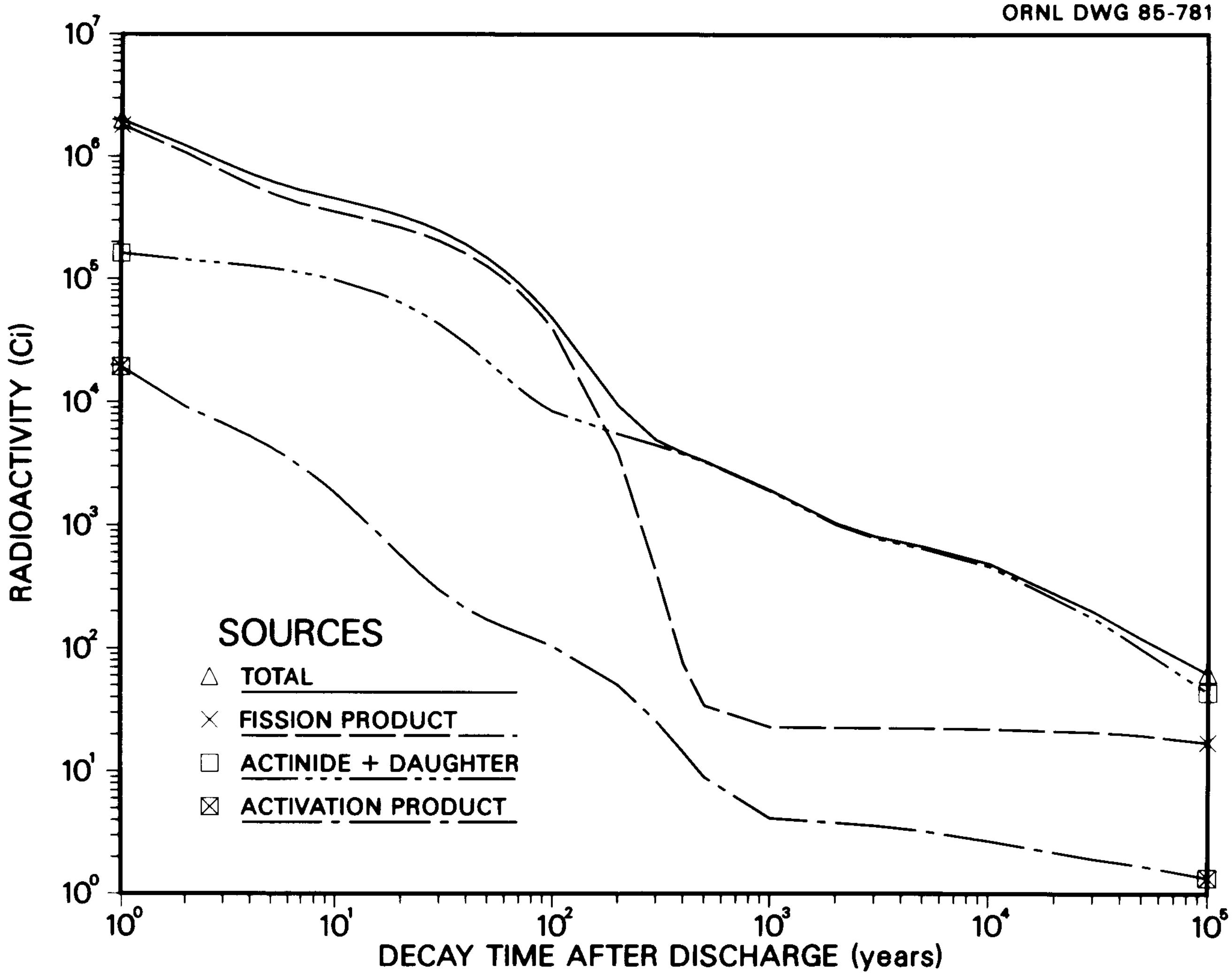


Fig. 3.3. Radioactivity produced by 1 metric ton of initial heavy metal: BWR; 40,000 MWd.

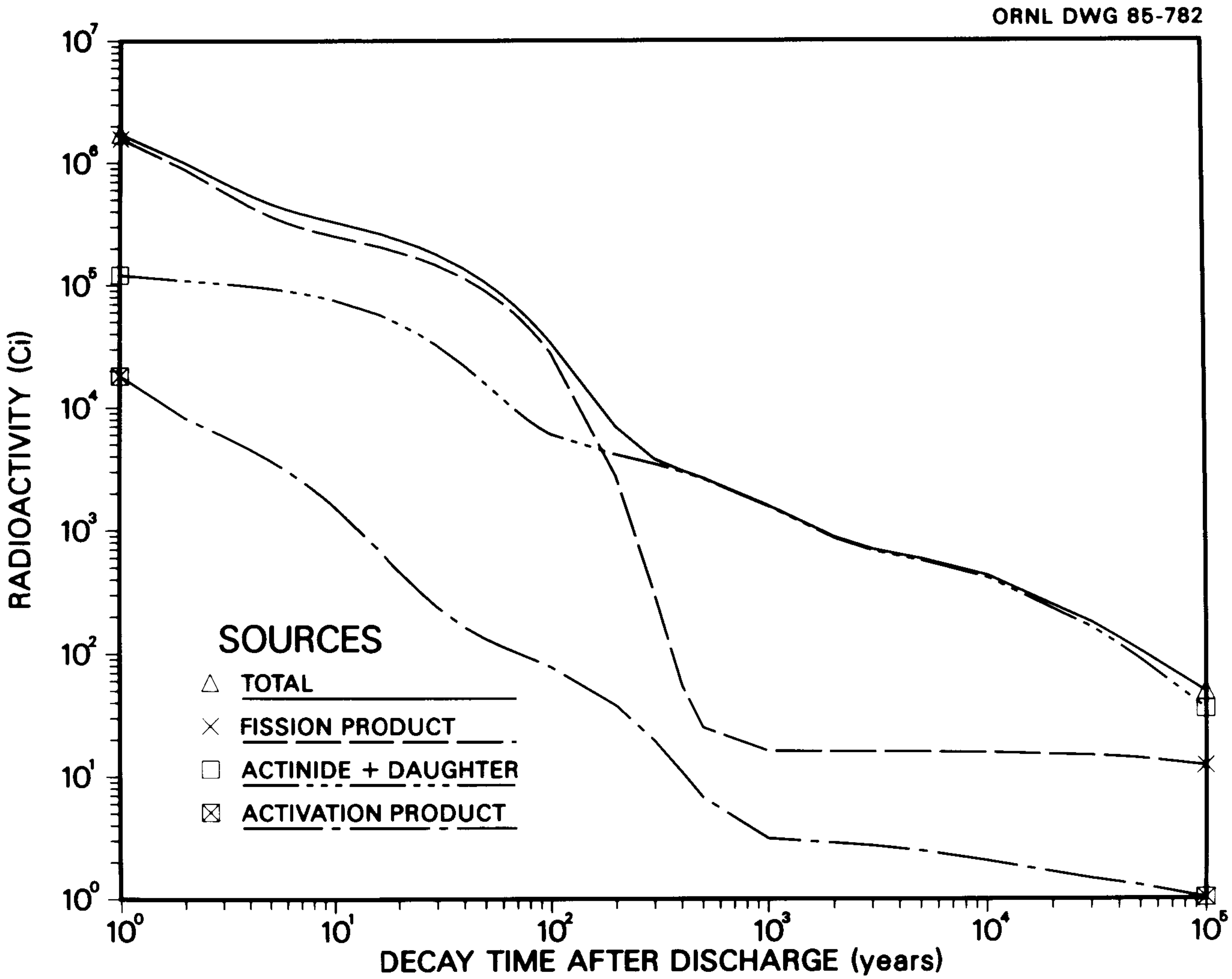


Fig. 3.4. Radioactivity produced by 1 metric ton of initial heavy metal: BWR; 27,500 MWd.

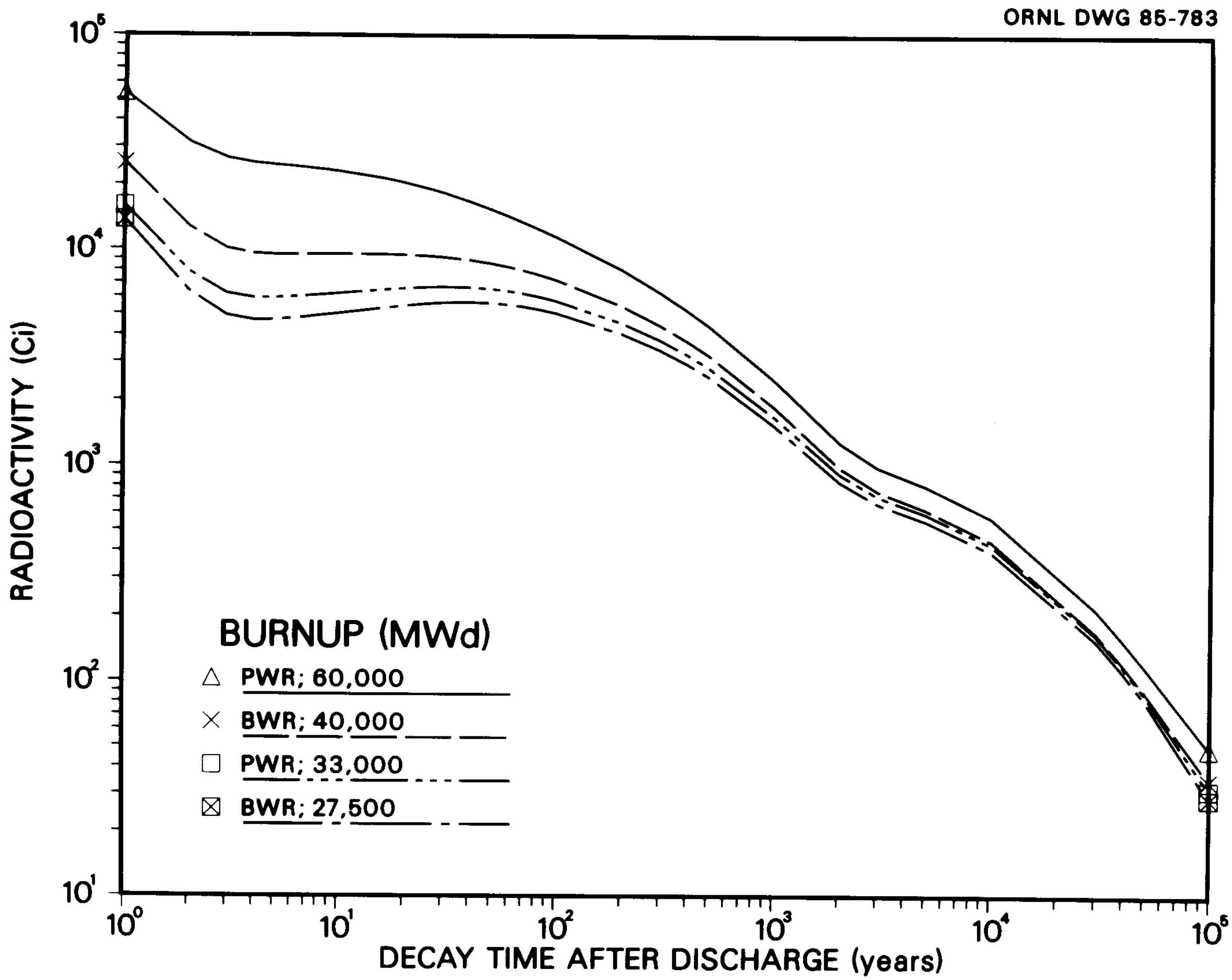


Fig. 3.5. Alpha radioactivity produced by 1 metric ton of initial heavy metal.

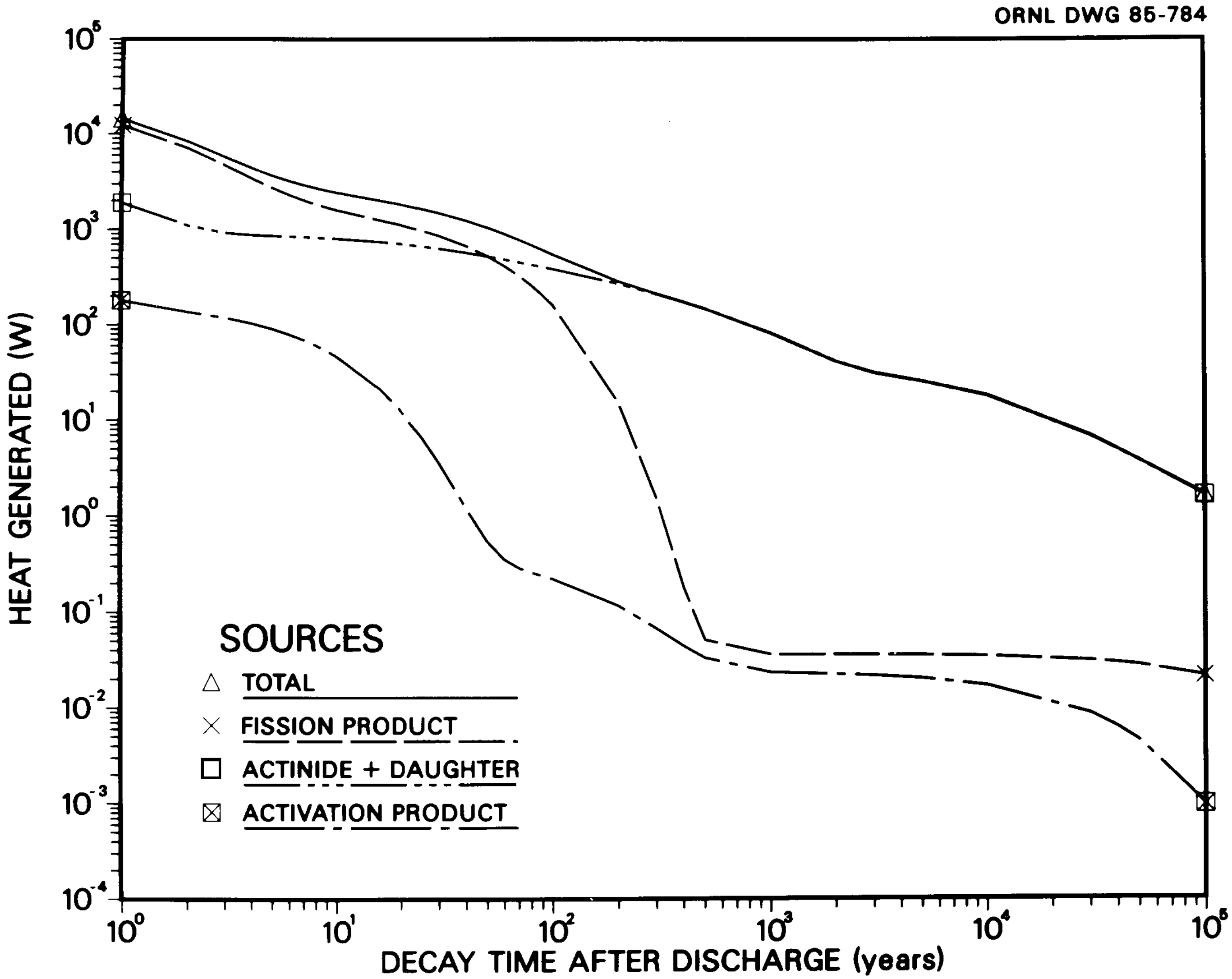


Fig. 3.6. Heat generated by 1 metric ton of initial heavy metal: PWR; 60,000 MWd.

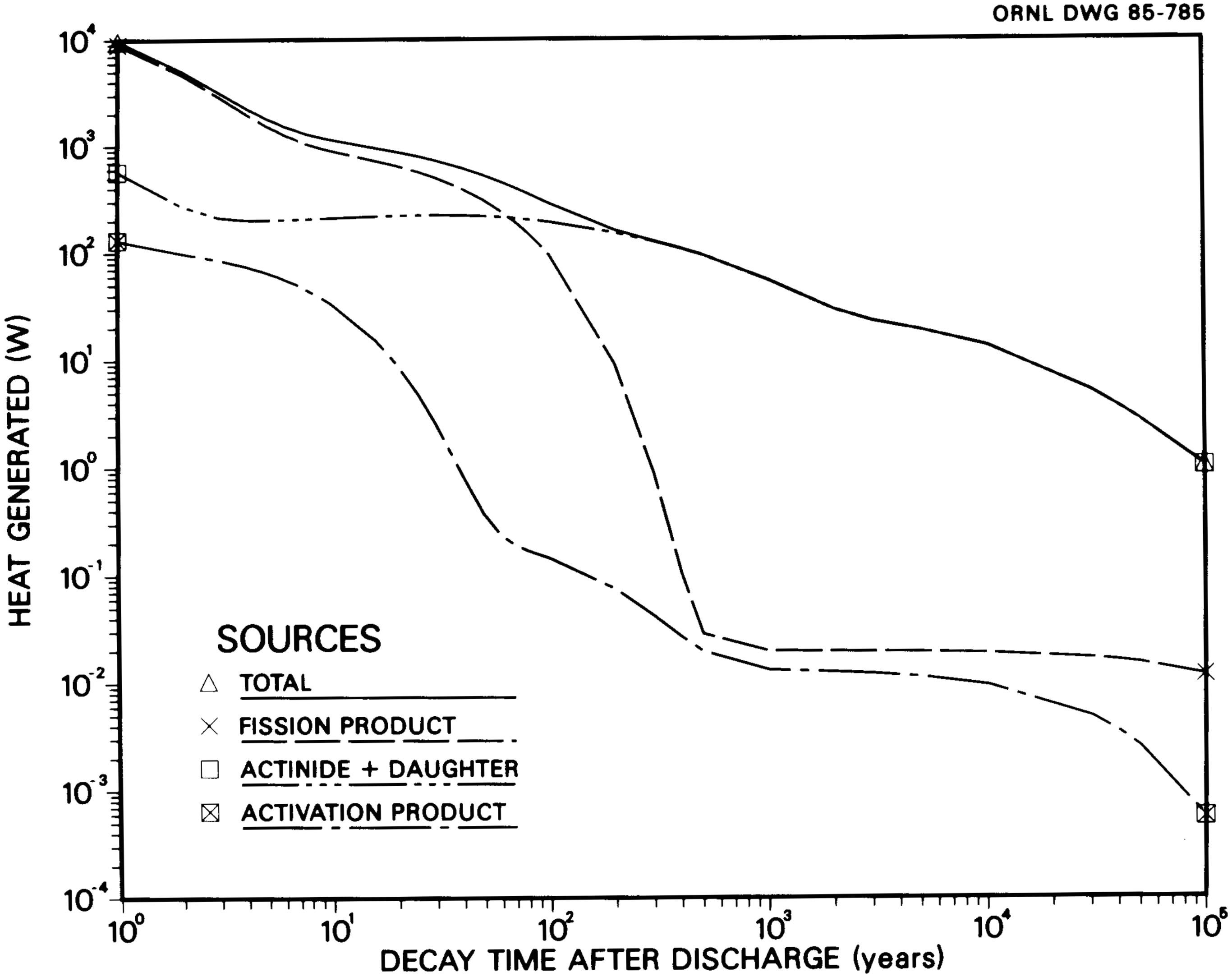


Fig. 3.7. Heat generated by 1 metric ton of initial heavy metal: PWR; 33,000 MWd.

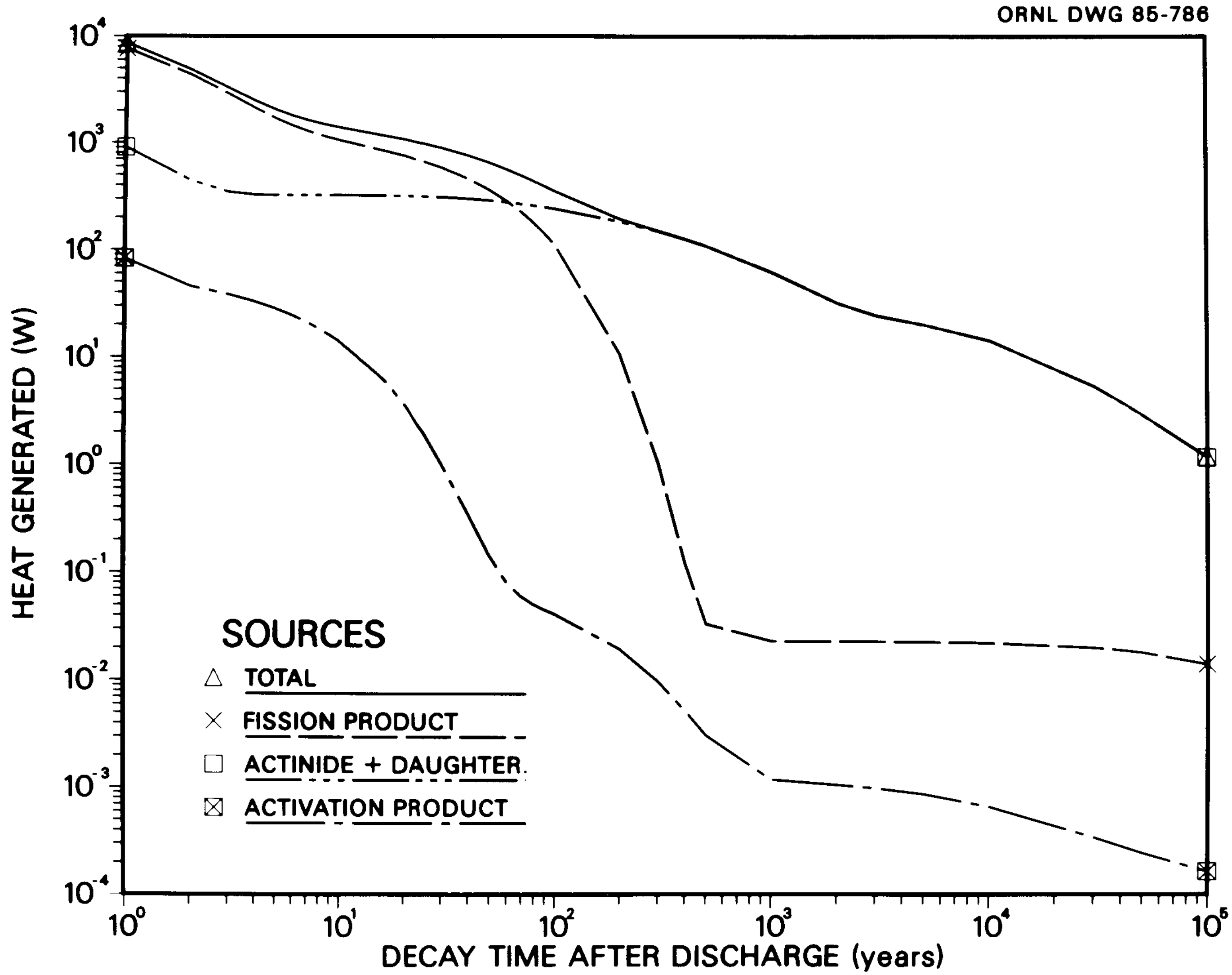


Fig. 3.8. Heat generated by 1 metric ton of initial heavy metal: BWR; 40,000 MWd.

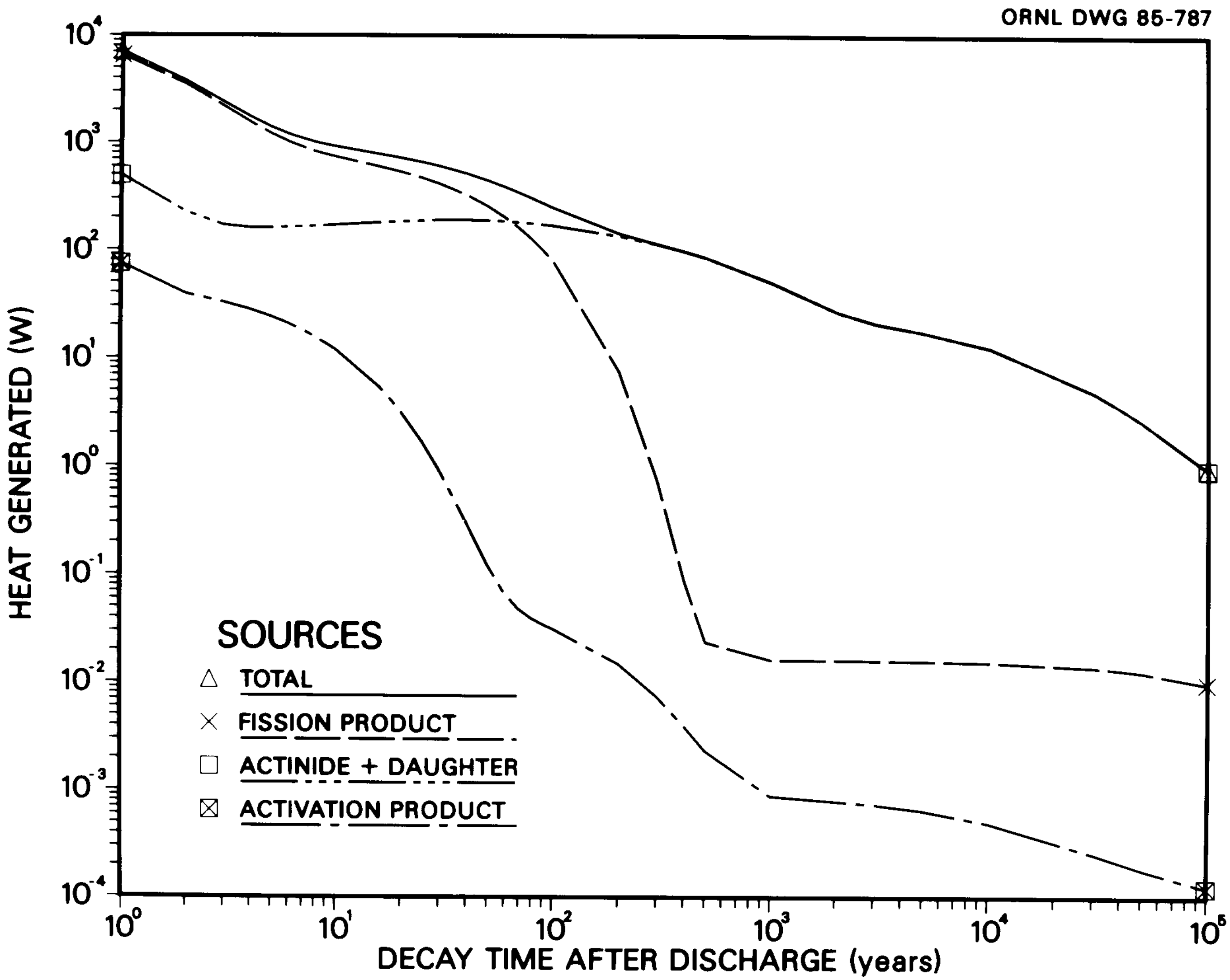


Fig. 3.9. Heat generated by 1 metric ton of initial heavy metal: BWR; 27,500 MWd.

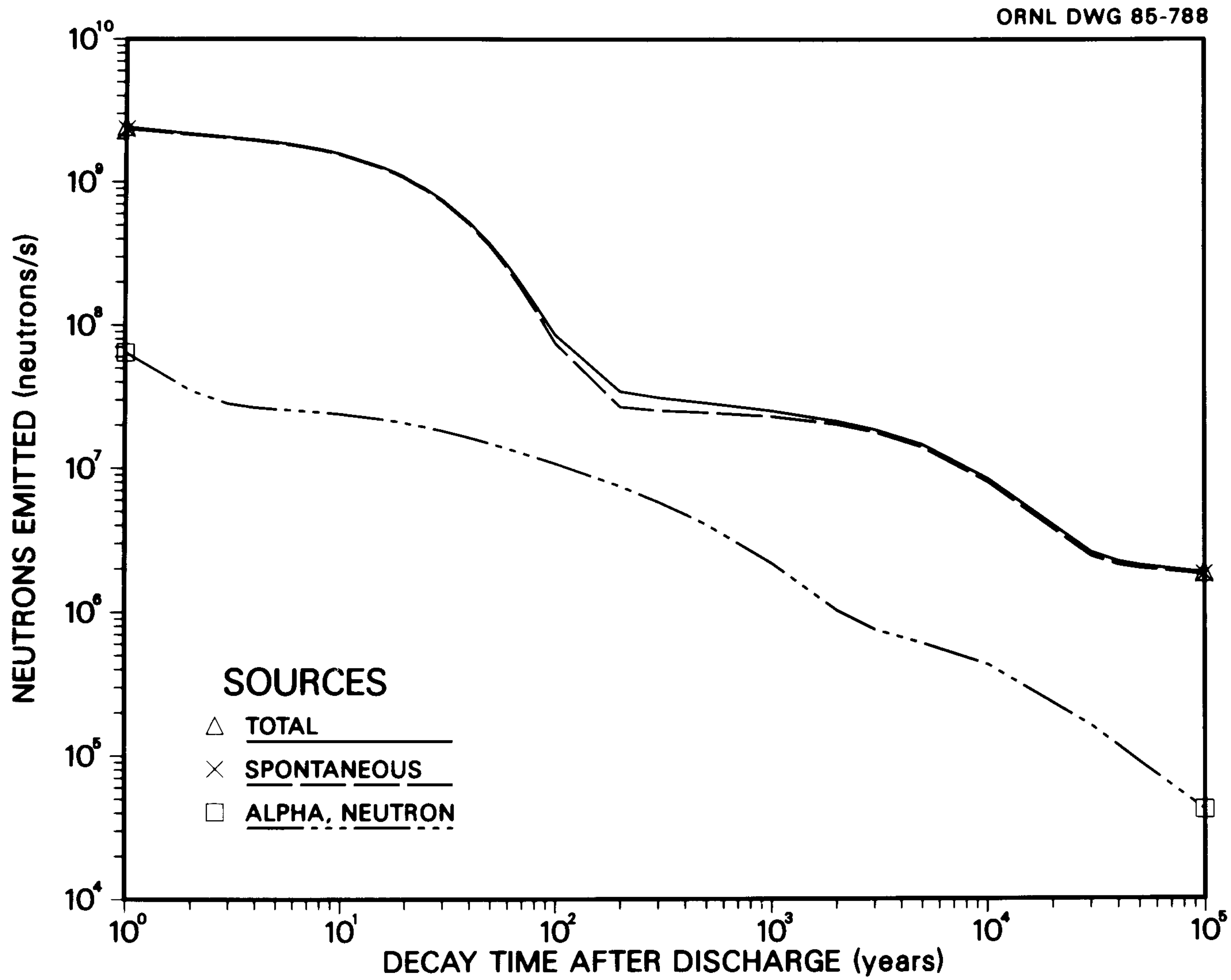


Fig. 3.10. Neutrons emitted by 1 metric ton of initial heavy metal: PWR; 60,000 MWd.

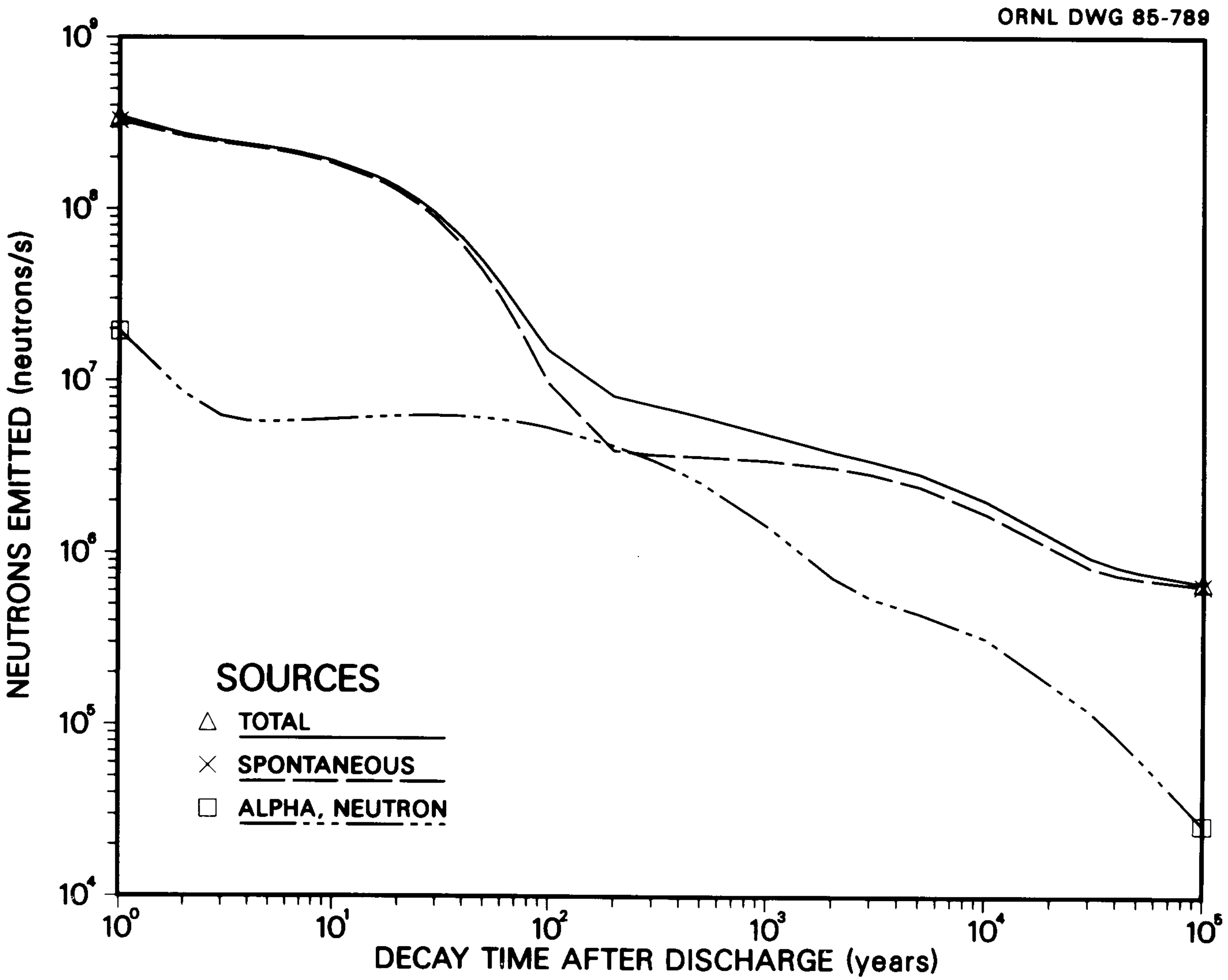


Fig. 3.11. Neutrons emitted by 1 metric ton of initial heavy metal: PWR; 33,000 MWd.

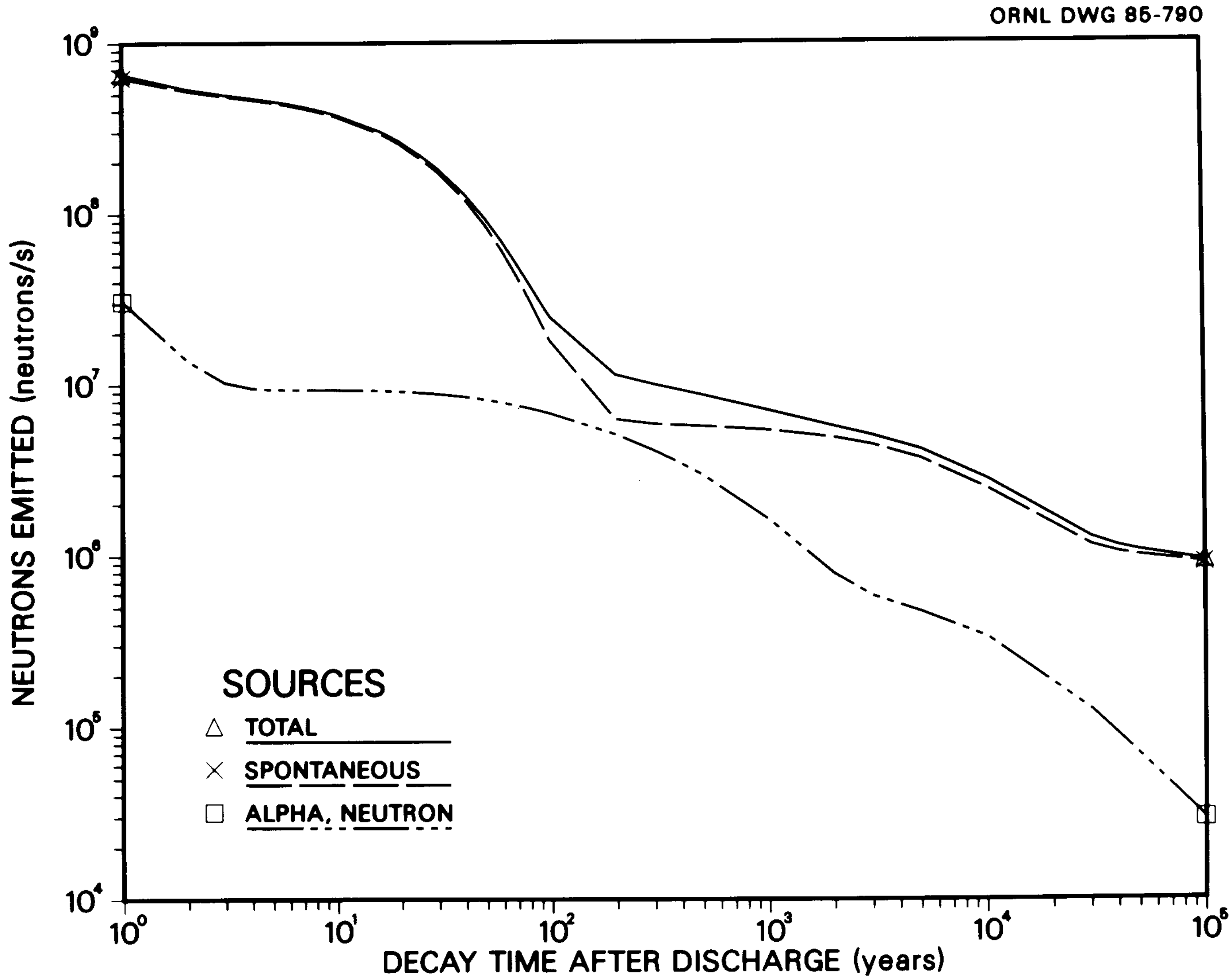


Fig. 3.12. Neutrons emitted by 1 metric ton of initial heavy metal: BWR; 40,000 MWd.

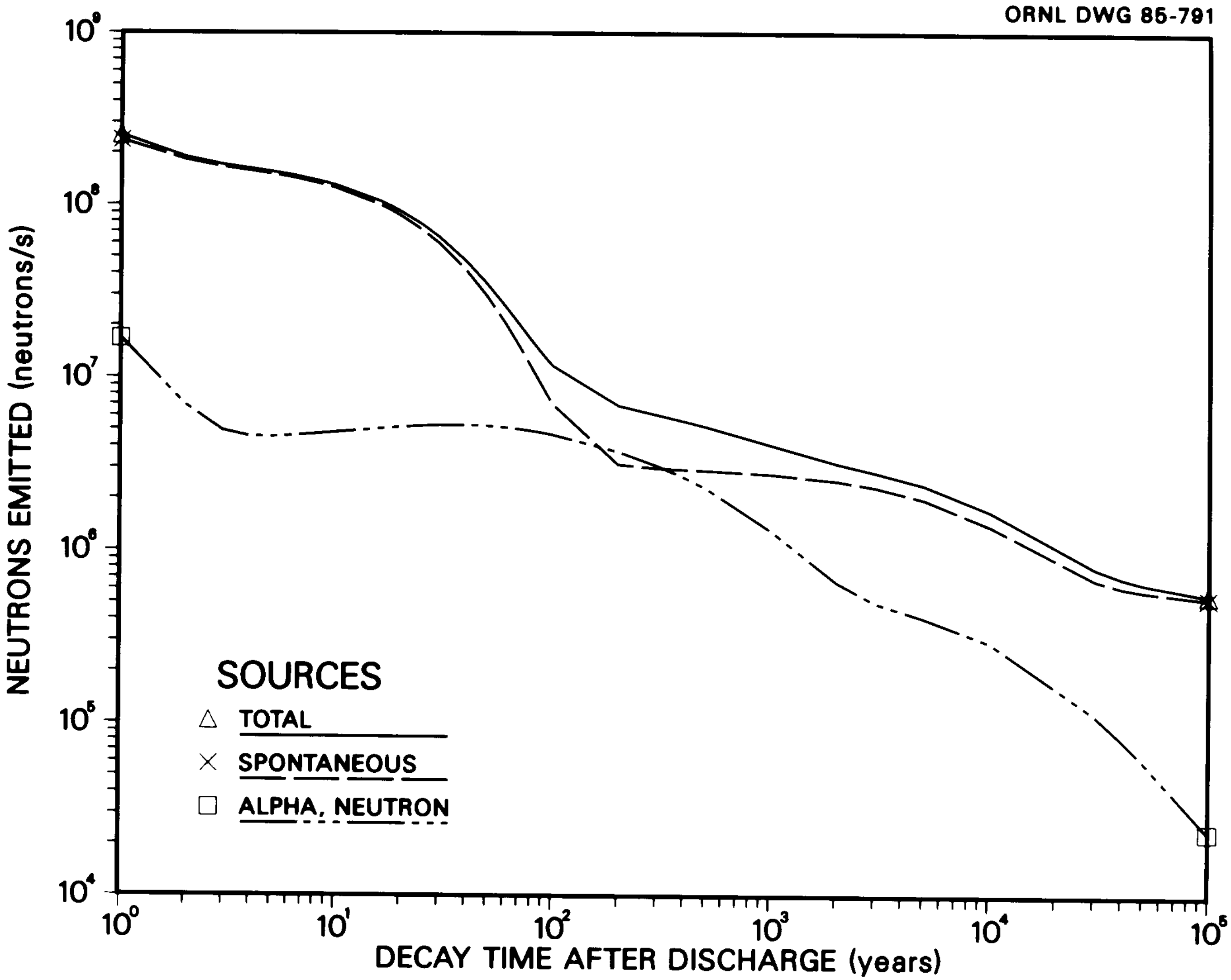


Fig. 3.13. Neutrons emitted by 1 metric ton of initial heavy metal: BWR; 27,500 MWd.

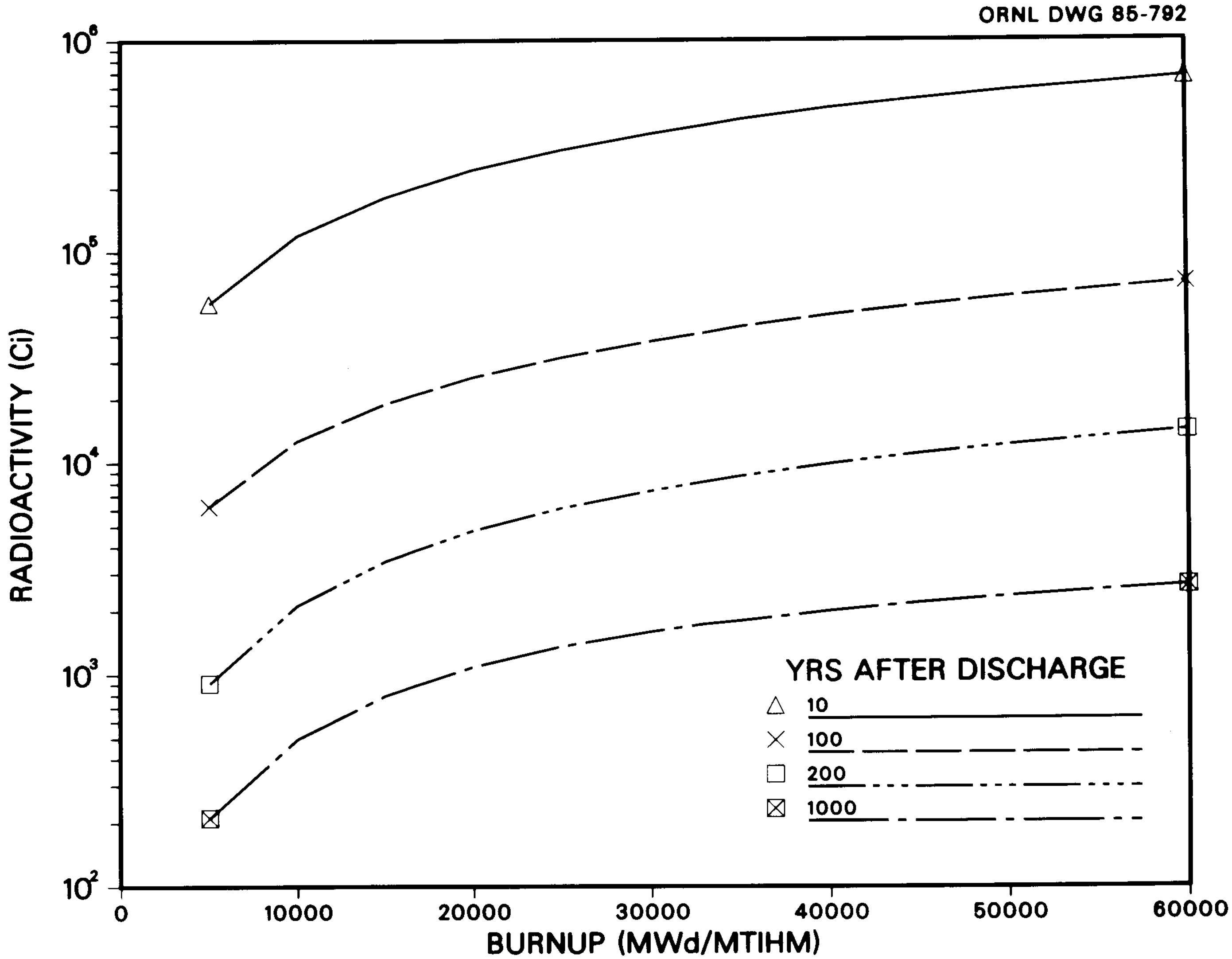


Fig. 3.14. Radioactivity produced by 1 metric ton of initial heavy metal for a PWR.

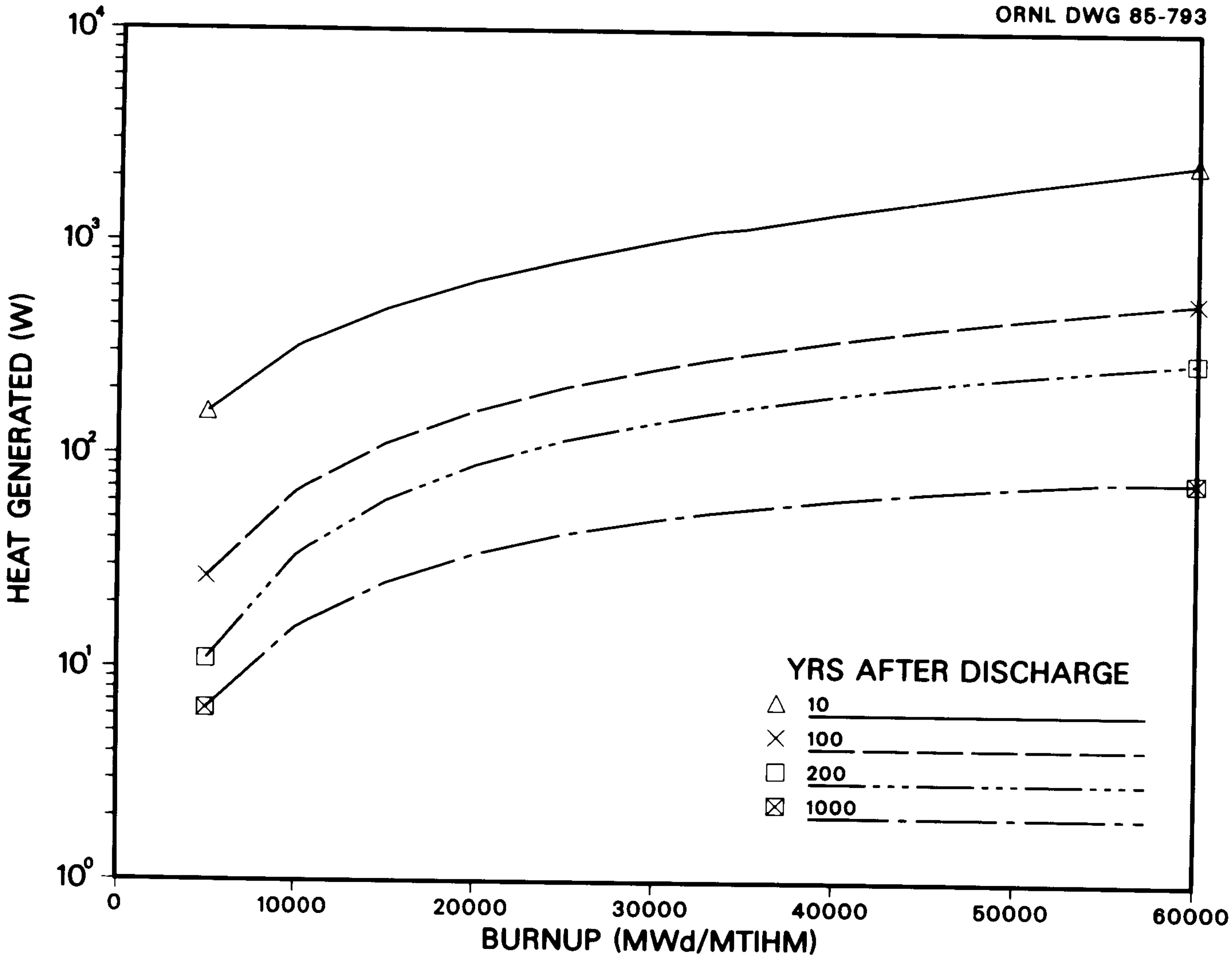


Fig. 3.15. Heat generated by 1 metric ton of initial heavy metal for a PWR.

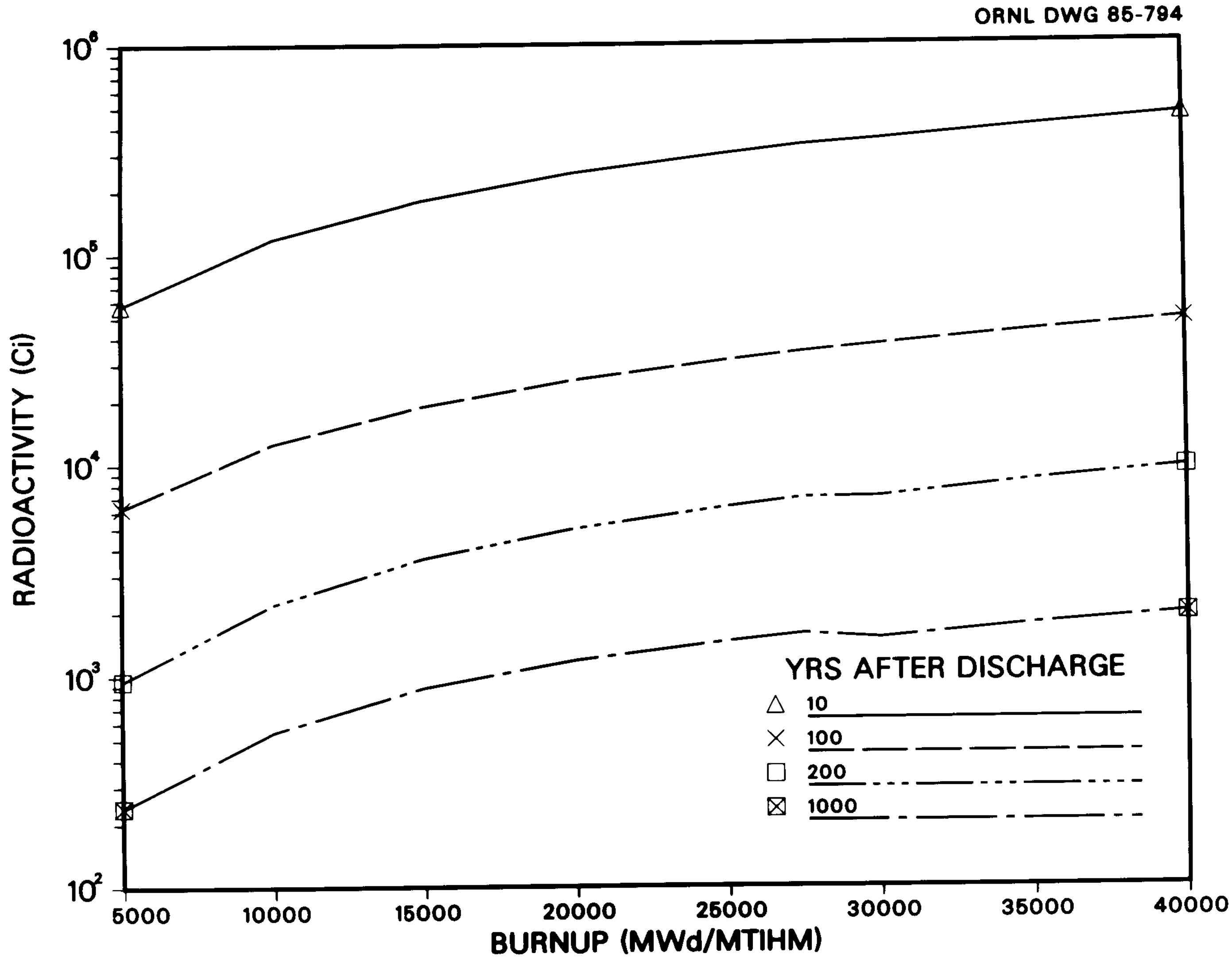


Fig. 3.16. Radioactivity produced by 1 metric ton of initial heavy metal for a BWR.

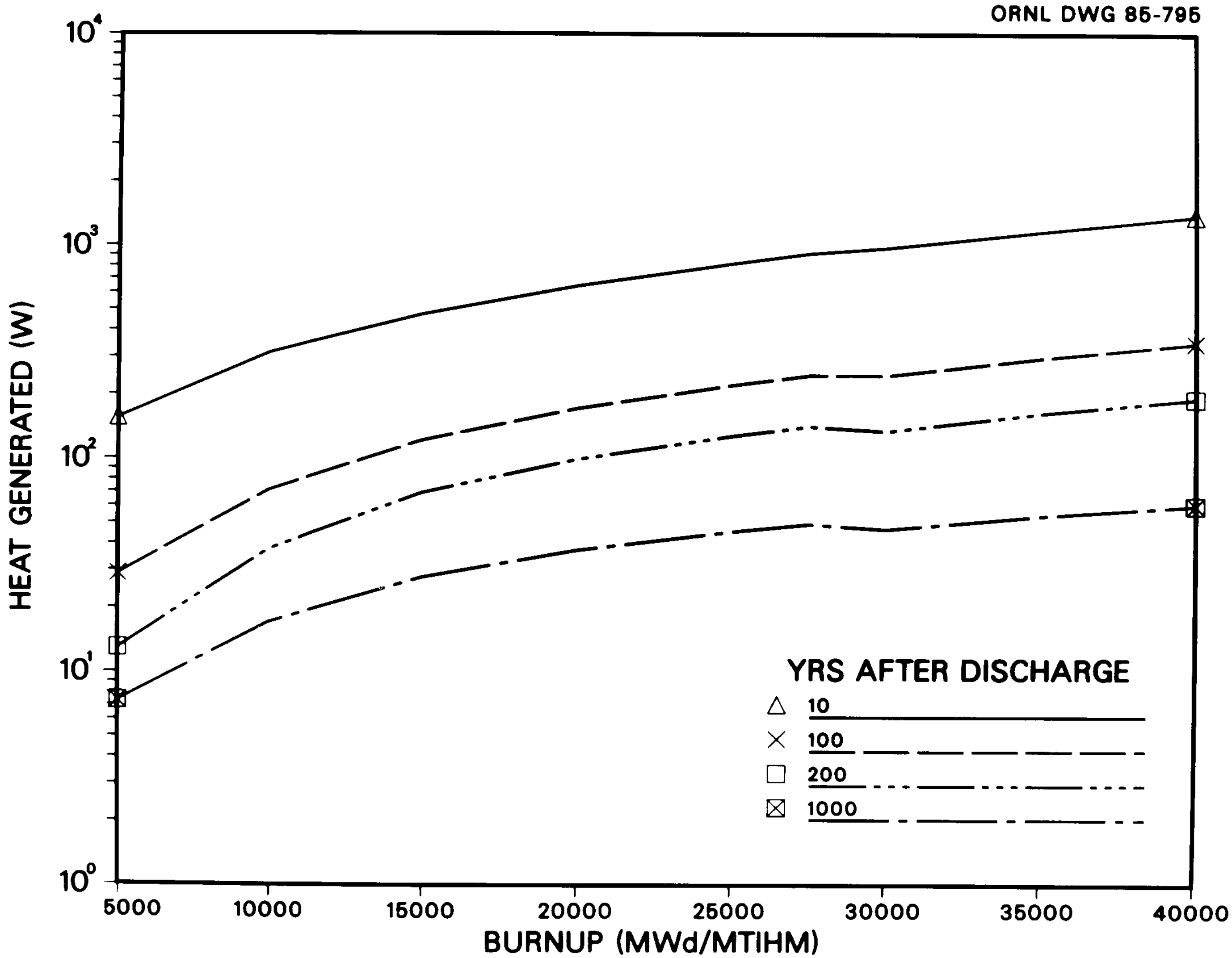


Fig. 3.17. Heat generated by 1 metric ton of initial heavy metal for a BWR.

Table 3.1. Physical characteristics of LWR fuel assemblies

	BWR[a]	PWR[b]
Overall assembly length, m	4.470	4.059
Cross section, cm	13.9 × 13.9	21.4 × 21.4
Fuel rod length, m	4.064	3.851
Active fuel height, m	3.759	3.658
Fuel rod OD, cm	1.252	0.950
Fuel rod array	8 × 8	17 × 17
Fuel rods per assembly	63	264
Assembly total weight, kg	319.9	657.9
Uranium/assembly, kg	183.3	461.4
UO_2/assembly, kg	208.0	523.4
Zircaloy/assembly, kg	103.3[c]	108.4[d]
Hardware/assembly, kg	8.6[e]	26.1[f]
Total metal/assembly, kg	111.9	134.5
Nominal volume/assembly, m^3	0.0864[g]	0.186[g]

[a]Source: General Electric Standard Safety Analysis Report, BWR/6, DOCKET STN 50-447, 1973.

[b]Source: Westinghouse Nuclear Energy Systems, RESAR-3, Reference Safety Analysis Report, DOCKET STN 50-480,1972.

[c]Includes Zircaloy fuel-rod spacers and fuel channel.

[d]Includes Zircaloy control-rod guide thimbles.

[e]Includes stainless steel tie-plates, Inconel springs, and plenum springs.

[f]Includes stainless steel nozzles and Inconel-718 grids.

[g]Based on overall outside dimension.

Table 3.2. Assumed fuel assembly structural material mass distribution

	PWR[a]			BWR[a]		
		Mass			Mass	
	Material	kg/MTHM	kg/assembly	Material	kg/MTHM	kg/assembly
<u>Fuel Zone</u>						
Cladding	Zircaloy-4	223.0	102.9	Zircaloy-2	279.5	51.2
Fuel channel[b]	--	--	--	Zircaloy-4	227.5	41.7
Grid spacers	Inconel 718	12.8	5.9	Zircaloy-4	10.6	1.9
Grid-spacer springs	Inconel 718			Inconel X-750	1.8	0.3
Grid-brazing material	Nicrobraze 50	2.6	1.2	--	--	--
Miscellaneous	SS 304[c]	9.9	4.6	--	--	--
<u>Fuel-gas plenum zone</u>						
Cladding	Zircaloy-4	12.0	5.5	Zircaloy-2	25.4	4.7
Fuel channel[b]	--	--	--	Zircaloy-4	20.7	3.8
Plenum spring	SS 302	4.2	1.9	SS 302	6.0	1.1
<u>End fitting zone</u>						
Top end fitting	SS 304	14.8	6.8	SS 304	10.9	2.0
Bottom end fitting	SS 304	12.4	5.7	SS 304	26.1	4.8
Expansion springs	--	--	--	Inconel X-750	2.1	0.4
Total		291.7	134.5		610.6	111.9

[a]<u>Source</u>: A. G. Croff, M. A. Bjerke, G. W. Morrison, and L. M. Petrie, <u>Revised Uranium — Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code</u>, ORNL/TM-6051, September 1978.

[b]Assumed to be discarded with fuel assembly, channels are often reused with fresh fuel.

[c]Distributed throughout the PWR core in sleeves and so forth.

Table 3.3. Assumed elemental compositions (g/ton of metal) of LWR fuel-assembly structural materials[a]

Element	Atomic number	Zircaloy-2	Zircaloy-4	Inconel-718	Inconel X-750	Stainless steel 302	Stainless steel 304	Nicrobraze 50
H	1	13	13	0	0	0	0	0
B	5	0.33	0.33	0	0	0	0	50
C	6	120	120	400	399	1,500	800	100
N	7	80	80	1,300	1,300	1,300	1,300	66
O	8	950	950	0	0	0	0	43
Al	13	24	24	5,992	7,982	0	0	100
Si	14	0	0	1,997	2,993	10,000	10,000	511
P	15	0	0	0	0	450	450	103,244
S	16	35	35	70	70	300	300	100
Ti	22	20	20	7,990	24,943	0	0	100
V	23	20	20	0	0	0	0	0
Cr	24	1,000	1,250	189,753	149,660	180,000	190,000	149,709
Mn[b]	25	20	20	1,997	6,984	20,000	20,000	100
Fe	26	1,500	2,250	179,766	67,846	697,740	688,440	471
Co[b]	27	10	10	4,694	6,485	800	800	381
Ni	28	500	20	519,625	721,861	89,200	89,200	744,438
Cu	29	20	20	999	499	0	0	0
Zr[b]	40	979,630	979,110	0	0	0	0	100
Nb	41	0	0	55,458	8,980	0	0	0
Mo	42	0	0	29,961	0	0	0	0
Cd	48	0.25	0.25	0	0	0	0	0
Sn	50	16,000	16,000	0	0	0	0	0
Hf	72	78	78	0	0	0	0	0
W	74	20	20	0	0	0	0	100
U	92	0.2	0.2	0	0	0	0	0
Density, g/cm^3	--	6.56	6.56	8.19	8.30	8.02	8.02	--

[a]Source: A. G. Croff, M. A. Bjerke, G. W. Morrison, and L. M. Petrie, Revised Uranium - Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code, ORNL/TM-6051, September 1978.

[b]Value used in ORIGEN should be less than this (actual) value if the materials are not in the active fuel zone.

Table 3.4. Assumed initial compositions of 1 metric ton of heavy metal in reference LWRs

Nonactinide composition of oxide fuel

Element	Atomic number	Quantity (g)	Element	Atomic number	Quantity (g)
Li	3	1.00E+0	Mn	25	1.70E+0
B	5	1.00E+0	Fe	26	1.80E+1
C	6	8.96E+1	Co	27	1.00E+0
N	7	2.50E+1	Ni	28	2.40E+1
O	8	1.34E+5	Cu	29	1.00E+0
F	9	1.07E+1	Zn	30	4.03E+1
Na	11	1.50E+1	Mo	42	1.00E+1
Mg	12	2.00E+1	Ag	47	1.00E-1
Al	13	1.67E+1	Cd	48	2.50E+1
Si	14	1.21E+1	In	49	2.00E+0
P	15	3.50E+1	Sn	50	4.00E+0
Cl	17	5.30E+0	Gd[a]	64	2.50E+0
Ca	20	2.00E+0	W	74	2.00E+0
Ti	22	1.00E+0	Pb	82	1.00E+0
V	23	3.00E+0	Bi	83	4.00E-1
Cr	24	4.00E+0			

Actinide composition of oxide fuel

Isotope	Burnup (MWd/MTIHM) PWR 33,000 Quantity (g)	PWR 60,000 Quantity (g)	BWR 27,500 Quantity (g)	BWR 40,000 Quantity (g)
^{234}U	2.90E+2	3.76E+2	2.47E+2	3.14E+2
^{235}U	3.20E+4	4.15E+4	2.75E+4	3.50E+4
^{238}U	9.68E+5	9.58E+5	9.72E+5	9.65E+5

[a]Average of 1.57E+3 g of gadolinium in BWR fuel rods as a burnable poison.

Table 3.5. Variation of radioactivity (Ci/MTIHM) for significant activation- and fission-product nuclides as a function of time since discharge from a 60,000-MWd/MTIHM PWR
(Includes all structural material)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
H-3[b]	1.17E+3	7.09E+2	4.54E+0	-	-	-
C-14[c]	2.44E+0	2.44E+0	2.41E+0	2.16E+0	7.27E-1	-
Mn-54[c]	4.59E+2	-	-	-	-	-
Fe-55[c]	5.24E+3	4.76E+2	-	-	-	-
Co-58[c]	2.13E+2	-	-	-	-	-
Co-60[c]	9.54E+3	2.92E+3	-	-	-	-
Ni-59[c]	6.40E+0	6.40E+0	6.39E+0	6.34E+0	5.87E+0	2.69E+0
Ni-63[c]	1.05E+3	9.83E+2	4.98E+2	-	-	-
Zn-65[c]	4.78E+1	-	-	-	-	-
Se-79	-	-	-	-	6.45E-1	2.47E-1
Kr-85	1.34E+4	7.48E+3	2.22E+1	-	-	-
Sr-89	4.53E+3	-	-	-	-	-
Sr-90	1.14E+5	9.16E+4	1.08E+4	-	-	-
Y-90	1.14E+5	9.16E+4	1.08E+4	-	-	-
Y-91	1.22E+4	-	-	-	-	-
Zr-93[b]	3.32E+0	3.32E+0	3.32E+0	3.32E+0	3.30E+0	3.17E+0
Zr-95[b]	2.93E+4	-	-	-	-	-
Nb-93m[b]	-	-	3.14E+0	3.15E+0	3.14E+0	3.01E+0
Nb-94[c]	-	-	-	2.18E+0	1.61E+0	7.43E-2
Nb-95[b]	6.59E+4	-	-	-	-	-
Tc-99	2.11E+1	2.11E+1	2.11E+1	2.10E+1	2.04E+1	1.52E+1
Ru-103	2.84E+3	-	-	-	-	-
Ru-106	3.84E+5	7.88E+2	-	-	-	-
Rh-106	3.84E+5	7.88E+2	-	-	-	-
Pd-107	-	-	-	2.43E-1	2.43E-1	2.41E-1
Ag-110m	3.72E+3	-	-	-	-	-
Sn-119m[b]	2.47E+3	-	-	-	-	-
Sn-126	1.47E+0	1.47E+0	1.47E+0	1.46E+0	1.37E+0	7.35E-1
Sb-125[b]	1.80E+4	1.89E+3	-	-	-	-
Sb-126	-	-	-	2.04E-1	1.92E-1	1.03E-1
Sb-126m	-	-	-	1.46E+0	1.37E+0	7.35E-1
Te-125m[b]	4.38E+3	4.62E+2	-	-	-	-
I-129	5.68E-2	5.68E-2	5.68E-2	5.68E-2	5.68E-2	5.66E-2
Cs-134	2.62E+5	1.27E+4	-	-	-	-
Cs-135	-	-	-	7.66E-1	7.64E-1	7.43E-1
Cs-137	1.78E+5	1.44E+5	1.80E+4	-	-	-
Ba-137m	1.68E+5	1.37E+5	1.71E+4	-	-	-
Ce-144	4.29E+5	1.42E+2	-	-	-	-
Pr-144	4.29E+5	1.42E+2	-	-	-	-
Pr-144m	5.14E+3	1.70E+0	-	-	-	-
Pm-147	9.39E+4	8.71E+3	-	-	-	-
Sm-151	5.30E+2	4.95E+2	2.47E+2	2.42E-1	-	-
Eu-154	2.33E+4	1.13E+4	7.99E+0	-	-	-
Eu-155	1.42E+4	4.05E+3	-	-	-	-
OTHER	7.55E+3	2.29E+2	1.22E+1	2.40E+0	9.89E-1	7.63E-2
SUBTOTAL						
A.P.[d]	2.59E+4	4.79E+3	5.11E+2	1.18E+1	8.71E+0	3.24E+0
F.P.[e]	2.75E+6	5.14E+5	5.70E+4	3.22E+1	3.10E+1	2.38E+1
TOTAL	2.79E+6	5.18E+5	5.75E+4	4.40E+1	3.98E+1	2.71E+1

[a]Nuclides contributing >0.1% are listed.

[b]Both activation and fission products contribute to this nuclide.

[c]Only activation products contribute to this nuclide.

[d]A.P. = Activation products.

[e]F.P. = Fission products.

Table 3.6. Variation of radioactivity (Ci/MTIHM) for significant activation- and fission-product nuclides as a function of time since discharge from a 33,000-MWd/MTIHM PWR
(Includes all structural material)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
H-3[b]	7.69E+2	4.64E+2	2.97E+0	-	-	-
C-14[c]	1.55E+0	1.55E+0	1.53E+0	1.38E+0	4.63E-1	-
Mn-54[c]	3.91E+2	-	-	-	-	-
Fe-55[c]	4.28E+3	3.89E+2	-	-	-	-
Co-58[c]	1.92E+2	-	-	-	-	-
Co-60[c]	6.97E+3	2.12E+3	-	-	-	-
Ni-59[c]	5.15E+0	5.15E+0	5.15E+0	5.11E+0	4.72E+0	2.17E+0
Ni-63[c]	6.97E+2	6.52E+2	3.31E+2	3.76E-1	-	-
Zn-65[c]	4.72E+1	-	-	-	-	-
Se-79	-	-	-	-	3.67E-1	1.41E-1
Kr-85	8.69E+3	4.85E+3	1.44E+1	-	-	-
Sr-89	5.72E+3	-	-	-	-	-
Sr-90	7.08E+4	5.72E+4	6.71E+3	-	-	-
Y-90	7.08E+4	5.72E+4	6.71E+3	-	-	-
Y-91	1.49E+4	-	-	-	-	-
Zr-93[b]	1.93E+0	1.93E+0	1.93E+0	1.93E+0	1.92E+0	1.84E+0
Zr-95[b]	3.14E+4	-	-	-	-	-
Nb-93m[b]	-	-	-	1.83E+0	1.83E+0	1.75E+0
Nb-94[c]	-	-	-	1.24E+0	9.10E-1	4.21E-2
Nb-95[b]	7.07E+4	-	-	-	-	-
Tc-99	1.31E+1	1.31E+1	1.30E+1	1.30E+1	1.26E+1	9.43E+0
Ru-103	2.59E+3	-	-	-	-	-
Ru-106	2.68E+5	5.50E+2	-	-	-	-
Rh-106	2.68E+5	5.50E+2	-	-	-	-
Pd-107	-	-	-	1.12E-1	1.12E-1	1.11E-1
Ag-110m	1.52E+3	-	-	-	-	-
Sn-119m[b]	2.14E+3	-	-	-	-	-
Sn-126	7.76E-1	7.76E-1	7.76E-1	7.71E-1	7.24E-1	3.88E-1
Sb-125[b]	1.22E+4	1.29E+3	-	-	-	-
Sb-126	-	-	-	1.08E-1	1.01E-1	5.44E-2
Sb-126m	-	-	-	7.71E-1	7.24E-1	3.88E-1
Te-125m[b]	2.98E+3	3.14E+2	-	-	-	-
I-129	3.15E-2	3.15E-2	3.15E-2	3.15E-2	3.15E-2	3.14E-2
Cs-134	1.08E+5	5.22E+3	-	-	-	-
Cs-135	-	-	-	3.45E-1	3.44E-1	3.35E-1
Cs-137	1.01E+5	8.21E+4	1.03E+4	-	-	-
Ba-137m	9.56E+4	7.77E+4	9.71E+3	-	-	-
Ce-144	4.51E+5	1.49E+2	-	-	-	-
Pr-144	4.51E+5	1.49E+2	-	-	-	-
Pr-144m	5.41E+3	1.79E+0	-	-	-	-
Pm-147	1.02E+5	9.48E+3	-	-	-	-
Sm-151	3.55E+2	3.31E+2	1.66E+2	1.62E-1	-	-
Eu-154	9.69E+3	4.69E+3	3.32E+0	-	-	-
Eu-155	5.62E+3	1.60E+3	-	-	-	-
OTHER	6.81E+3	3.80E+1	8.70E+0	9.90E-1	6.70E-2	5.60E-2
SUBTOTAL						
A.P.[d]	1.95E+4	3.48E+3	3.40E+2	8.38E+0	6.36E+0	2.46E+0
F.P.[e]	2.16E+6	3.04E+5	3.36E+4	1.92E+1	1.86E+1	1.42E+1
TOTAL	2.18E+6	3.07E+5	3.39E+4	2.76E+1	2.49E+1	1.67E+1

[a]Nuclides contributing >0.1% are listed.

[b]Both activation and fission products contribute to this nuclide.

[c]Only activation products contribute to this nuclide.

[d]A.P. = Activation products.

[e]F.P. = Fission products.

Table 3.7. Variation of radioactivity (Ci/MTIHM) for significant activation- and fission-product nuclides as a function of time since discharge from a 40,000-MWd/MTIHM BWR
(Includes all structural material)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
H-3[b]	8.43E+2[a]	5.09E+2	3.26E+0	-	-	-
C-14[c]	2.05E+0	2.05E+0	2.02E+0	1.82E+0	6.11E-1	-
Mn-54[c]	1.49E+2	-	-	-	-	-
Fe-55[c]	2.54E+3	2.31E+2	-	-	-	-
Co-58[c]	3.75E+1	-	-	-	-	-
Co-60[c]	2.62E+3	8.01E+2	-	-	-	-
Ni-59[c]	1.39E+0	1.39E+0	1.39E+0	1.38E+0	1.27E+0	5.84E-1
Ni-63[c]	2.08E+2	1.94E+2	9.84E+1	-	-	-
Zn-65[c]	3.56E+1	-	-	-	-	-
Se-79	-	-	-	4.80E-1	4.36E-1	1.67E-1
Kr-85	9.52E+3	5.32E+3	1.58E+1	-	-	-
Sr-89	3.59E+3	-	-	-	-	-
Sr-90	8.20E+4	6.62E+4	7.77E+3	-	-	-
Y-90	8.20E+4	6.62E+4	7.77E+3	-	-	-
Y-91	9.41E+3	-	-	-	-	-
Zr-93[b]	2.56E+0	2.56E+0	2.56E+0	2.56E+0	2.55E+0	2.45E+0
Zr-95[b]	2.18E+4	-	-	-	-	-
Nb-93m[b]	-	-	-	2.44E+0	2.43E+0	2.33E+0
Nb-95[b]	4.89E+4	-	-	-	-	-
Tc-99	1.56E+1	1.56E+1	1.56E+1	1.56E+1	1.51E+1	1.13E+1
Ru-103	1.86E+3	-	-	-	-	-
Ru-106	2.28E+5	4.67E+2	-	-	-	-
Rh-106	2.28E+5	4.67E+2	-	-	-	-
Pd-107	-	-	-	1.40E-1	1.40E-1	1.39E-1
Ag-110m	1.63E+3	-	-	-	-	-
Sn-119m[b]	3.83E+3	-	-	-	-	-
Sn-126	8.88E-1	8.88E-1	8.87E-1	8.82E-1	8.28E-1	4.44E-1
Sb-125[b]	1.25E+4	1.31E+3	-	-	-	-
Sb-126	-	-	-	1.24E-1	1.16E-1	6.22E-2
Sb-126m	-	-	-	8.82E-1	8.28E-1	4.44E-1
Te-125m[b]	3.04E+3	3.20E+2	-	-	-	-
I-129	3.73E-2	3.73E-2	3.73E-2	3.73E-2	3.73E-2	3.72E-2
Cs-134	1.27E+5	6.15E+3	-	-	-	-
Cs-135	-	-	-	5.66E-1	5.64E-1	5.49E-1
Cs-137	1.19E+5	9.66E+4	1.21E+4	-	-	-
Ba-137m	1.12E+5	9.14E+4	1.14E+4	-	-	-
Ce-144	3.06E+5	1.01E+2	-	-	-	-
Pr-144	3.06E+5	1.01E+2	-	-	-	-
Pr-144m	3.67E+3	-	-	-	-	-
Pm-147	8.80E+4	8.20E+3	-	-	-	-
Sm-151	3.80E+2	3.55E+2	1.78E+2	1.73E-1	-	-
Eu-154[b]	1.30E+4	6.31E+3	4.42E+0	-	-	-
Eu-155[b]	7.46E+3	2.12E+3	-	-	-	-
OTHER	4.95E+3	2.15E+1	3.52E+1	2.12E-1	8.14E-2	2.10E-2
SUBTOTAL						
A.P.[d]	1.94E+4	1.84E+3	1.04E+2	4.15E+0	2.71E+0	1.35E+0
F.P.[e]	1.81E+6	3.52E+5	3.93E+4	2.30E+1	2.22E+1	1.71E+1
TOTAL	1.83E+6	3.53E+5	3.94E+4	2.72E+1	2.50E+1	1.85E+1

[a]Nuclides contributing >0.1% are listed.
[b]Both activation and fission products contribute to this nuclide.
[c]Only activation products contribute to this nuclide.
[d]A.P. = Activation products.
[e]F.P. = Fission products.

Table 3.8. Variation of radioactivity (Ci/MTIHM) for significant activation- and fission-product nuclides as a function of time since discharge from a 27,500-MWd/MTIHM BWR (Includes all structural material)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
H-3[b]	6.63E+2	4.00E+2	2.56E+0	-	-	-
C-14[c]	1.53E+0	1.53E+0	1.52E+0	1.36E+0	4.57E-1	-
Mn-54[c]	1.45E+2	-	-	-	-	-
Fe-55[c]	2.23E+3	2.02E+2	-	-	-	-
Co-58[c]	3.71E+1	-	-	-	-	-
Co-60[c]	2.18E+3	6.66E+2	-	-	-	-
Ni-59[c]	1.07E+0	1.07E+0	1.07E+0	1.06E+0	9.82E-1	4.50E-1
Ni-63[c]	1.57E+2	1.47E+2	7.47E+1	-	-	-
Zn-65[c]	3.51E+1	-	-	-	-	-
Se-79	-	-	-	3.34E-1	3.04E-1	1.16E-1
Kr-85	7.02E+3	3.92E+3	1.16E+1	-	-	-
Sr-89	3.90E+3	-	-	-	-	-
Sr-90	5.82E+4	4.70E+4	5.52E+3	-	-	-
Y-90	5.82E+4	4.70E+4	5.52E+3	-	-	-
Y-91	1.01E+4	-	-	-	-	-
Zr-93[b]	1.80E+0	1.80E+0	1.80E+0	1.80E+0	1.80E+0	1.72E+0
Zr-95[b]	2.24E+4	-	-	-	-	-
Nb-93m[b]	-	-	-	1.71E+0	1.71E+0	1.64E+0
Nb-95[b]	5.04E+4	-	-	-	-	-
Tc-99	1.11E+1	1.11E+1	1.11E+1	1.11E+1	1.08E+1	8.04E+0
Ru-103	1.81E+3	-	-	-	-	-
Ru-106	1.97E+5	4.04E+2	-	-	-	-
Rh-106	1.97E+5	4.04E+2	-	-	-	-
Pd-107	-	-	-	9.46E-2	9.45E-2	9.36E-2
Ag-110m	1.05E+3	-	-	-	-	-
Sn-119m[b]	3.77E+3	-	-	-	-	-
Sn-126	6.25E-1	6.24E-1	6.24E-1	6.20E-1	5.83E-1	3.12E-1
Sb-125[b]	1.05E+4	1.10E+3	-	-	-	-
Sb-126	-	-	-	8.68E-2	8.16E-2	4.37E-2
Sb-126m	-	-	-	6.20E-1	5.83E-1	3.12E-1
Te-125m[b]	2.56E+3	2.69E+2	-	-	-	-
I-129	2.64E-2	2.64E-2	2.64E-2	2.64E-2	2.64E-2	2.63E-2
Cs-134	7.65E+4	3.71E+3	-	-	-	-
Cs-135	-	-	-	3.59E-1	3.58E-1	3.49E-1
Cs-137	8.37E+4	6.80E+4	8.49E+3	-	-	-
Ba-137m	7.91E+4	6.43E+4	8.03E+3	-	-	-
Ce-144	3.10E+5	1.02E+2	-	-	-	-
Pr-144	3.10E+5	1.02E+2	-	-	-	-
Pr-144m	3.72E+3	1.23E+0	-	-	-	-
Pm-147	8.68E+4	8.05E+3	-	-	-	-
Sm-151	3.20E+2	2.98E+2	1.49E+2	1.46E-1	-	-
Eu-154[b]	7.63E+3	3.70E+3	2.61E+0	-	-	-
Eu-155[b]	4.49E+3	1.28E+3	-	-	-	-
OTHER	5.82E+3	9.30E+1	-	1.53E-1	5.40E-2	4.16E-2
SUBTOTAL						
A.P.[d]	1.81E+4	1.58E+3	7.92E+1	3.14E+0	2.06E+0	1.02E+0
F.P.[e]	1.58E+6	2.50E+5	2.78E+4	1.63E+1	1.57E+1	1.21E+1
TOTAL	1.60E+6	2.51E+5	2.78E+4	1.94E+1	1.78E+1	1.31E+1

[a]Nuclides contributing >than 0.1% are listed.

[b]Both activation and fission products contribute to this nuclide.

[c]Only activation products contribute to this nuclide.

[d]A.P. = Activation products.

[e]F.P. = Fission products.

Table 3.9. Variation of radioactivity (Ci/MTIHM) for significant actinides as a function of time since discharge from a 60,000-MWd/MTIHM PWR
(Includes all structural material)

Isotope[a]	Time since discharge (years)					
	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Ra-226	-	-	3.32E-5	5.81E-3	2.68E-1	2.12E+0
U-234	-	-	-	4.08E+0	3.99E+0	3.16E+0
Np-237	-	-	-	1.74E+0	2.03E+0	1.97E+0
Np-239	7.22E+1	7.21E+1	7.15E+1	6.57E+1	2.82E+1	-
Pu-238	8.56E+3	8.10E+3	3.98E+3	3.60E+0	-	-
Pu-239	3.67E+2	3.67E+2	3.66E+2	3.59E+2	2.87E+2	2.24E+1
Pu-240	6.78E+2	6.90E+2	7.13E+2	6.49E+2	2.50E+2	-
Pu-241	1.88E+5	1.22E+5	1.61E+3	1.74E+0	-	-
Pu-242	-	-	-	4.53E+0	4.47E+0	3.80E+0
Am-241	5.77E+2	2.76E+3	5.98E+3	1.43E+3	-	-
Am-243	7.22E+1	7.21E+1	7.15E+1	6.57E+1	2.82E+1	-
Cm-242	2.75E+4	1.40E+1	9.25E+0	-	-	-
Cm-243	9.13E+1	7.34E+1	8.22E+0	-	-	-
Cm-244	1.55E+4	1.10E+4	3.51E+2	-	-	-
OTHER	6.47E+1	4.16E+1	3.03E+1	5.84E+0	-	3.07E+1[b]
TOTAL	2.42E+5	1.45E+5	1.32E+4	2.59E+3	6.13E+2	6.20E+1

[a]Nuclides contributing >0.1% are listed.

[b]The following isotopes contribute 2.12 Ci each: Pb-210, Pb-214, Bi-210, Bi-214, Po-210, Po-214, Po-218, and Rn-222. Others contributing 0.64 Ci each include: Pb-209, Bi-213, At-217, Fr-221, Ra-225, Ac-225, and Th-229.

Table 3.10. Variation of radioactivity (Ci/MTIHM) for significant actinides as a function of time since discharge from a 33,000-MWd/MTIHM PWR (Includes all structural material)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Ra-226	-	-	2.66E-5	3.12E-3	1.34E-1	1.07E+0
U-234	-	-	-	2.03E+0	1.99E+0	1.61E+0
Np-237	-	-	-	9.99E-1	1.18E+0	1.14E+0
Np-239	1.71E+1	1.71E+1	1.69E+1	1.56E+1	6.68E+0	-
Pu-238	2.45E+3	2.33E+3	1.15E+3	1.08E+0	-	-
Pu-239	3.13E+2	3.13E+2	3.12E+2	3.05E+2	2.37E+2	1.80E+1
Pu-240	5.26E+2	5.27E+2	5.26E+2	4.78E+2	1.84E+2	-
Pu-241	1.20E+5	7.76E+4	1.02E+3	-	-	-
Pu-242	-	-	-	1.72E+0	1.69E+0	1.44E+0
Am-241	3.08E+2	1.69E+3	3.75E+3	8.93E+2	-	-
Am-243	1.71E+1	1.71E+1	1.69E+1	1.56E+1	6.68E+0	-
Cm-242	1.04E+4	5.72E+0	3.78E+0	-	-	-
Cm-243	2.06E+1	1.66E+1	1.86E+0	-	-	-
Cm-244	1.86E+3	1.32E+3	4.21E+1	-	-	-
OTHER	2.74E+2	2.60E+1	1.56E+1	2.68E+0	4.30E+0	1.68E+1[b]
TOTAL	1.36E+5	8.39E+4	6.85E+3	1.72E+3	4.44E+2	3.90E+1

[a]Nuclides contributing >0.1% are listed.

[b]The following isotopes contribute 1.07 Ci each: Pb-210, Pb-214, Bi-210, Bi-214, Po-210, Po-214, Po-218, and Rn-222. Others contributing 0.37 Ci each include: Pb-209, Bi-213, At-217, Fr-221, Ra-225, Ac-225, and Th-229.

Table 3.11. Variation of radioactivity (Ci/MTIHM) for significant actinides as a function of time since discharge from a 40,000-MWd/MTIHM BWR (Includes all structural material)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Ra-226	-	-	2.94E-5	3.85E-3	1.70E-1	1.35E+0
U-234	-	-	-	2.58E+0	2.52E+0	2.02E+0
Np-237	-	-	-	1.21E+0	1.42E+0	1.38E+0
Np-239	2.83E+1	2.83E+1	2.80E+1	2.58E+1	1.11E+1	-
Pu-238	4.06E+3	3.85E+3	1.90E+3	1.82E+0	-	-
Pu-239	3.06E+2	3.06E+2	3.06E+2	2.98E+2	2.34E+2	1.79E+1
Pu-240	5.63E+2	5.65E+2	5.67E+2	5.16E+2	1.98E+2	-
Pu-241	1.37E+5	8.87E+4	1.17E+3	-	-	-
Pu-242	-	-	-	2.37E+0	2.33E+0	1.98E+0
Am-241	4.36E+2	2.02E+3	4.36E+3	1.04E+3	-	-
Am-243	2.83E+1	2.83E+1	2.80E+1	2.58E+1	1.11E+1	-
Cm-242	1.60E+4	1.09E+1	7.22E+0	-	-	-
Cm-243	3.64E+1	2.92E+1	3.28E+0	-	-	-
Cm-244	3.75E+3	2.66E+3	8.48E+1	-	-	-
OTHER	1.08E+2	6.23E+1	1.27E+1	3.56E+0	5.33E+0	2.06E+1[b]
TOTAL	1.62E+5	9.83E+4	8.47E+3	1.92E+3	4.66E+2	4.38E+1

[a]Nuclides contributing >0.1% are listed.

[b]The following isotopes contribute 1.35 Ci each: Pb-210, Pb-214, Bi-210, Bi-214, Po-210, Po-214, Po-218, and Rn-222. Others contributing 0.45 Ci each include: Pb-209, Bi-213, At-217, Fr-221, Ra-225, Ac-225, and Th-229.

Table 3.12. Variation of radioactivity (Ci/MTIHM) for significant actinides as a function of time since discharge from a 27,500-MWd/MTIHM BWR
(Includes all structural material)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Ra-226	-	-	2.32E-5	2.60E-3	1.11E-1	8.86E-1
U-234	-	-	-	1.68E+0	1.64E+0	1.34E+0
Np-237	-	-	-	8.64E-1	1.02E+0	9.95E-1
Np-239	1.29E+1	1.29E+1	1.28E+1	1.18E+1	5.06E+0	-
Pu-238	1.86E+3	1.78E+3	8.77E+2	8.87E-1	-	-
Pu-239	3.00E+2	3.00E+2	3.00E+2	2.92E+2	2.27E+2	1.72E+1
Pu-240	4.78E+2	4.78E+2	4.76E+2	4.33E+2	1.67E+2	-
Pu-241	1.07E+5	6.95E+4	9.13E+2	-	-	-
Pu-242	-	-	-	1.42E+0	1.39E+0	1.19E+0
Am-241	3.15E+2	1.56E+3	3.39E+3	8.07E+2	-	-
Am-243	1.29E+1	1.29E+1	1.28E+1	1.18E+1	5.06E+0	-
Cm-242	9.42E+3	6.87E+0	4.54E+0	-	-	-
Cm-243	1.67E+1	1.34E+1	1.50E+0	-	-	-
Cm-244	1.25E+3	8.86E+2	2.83E+1	-	-	-
OTHER	3.05E+1	2.29E+1	1.61E+1	2.00E+0	3.90E+0	1.44E+1[b]
TOTAL	1.21E+5	7.45E+4	6.03E+3	1.56E+3	4.12E+2	3.51E+1

[a]Nuclides contributing >0.1% are listed.

[b]The following isotopes contribute 0.89 Ci each: Pb-210, Pb-214, Bi-210, Bi-214, Po-210, Po-214, Po-218, and Rn-222. Others contributing 0.33 Ci each include: Pb-209, Bi-213, At-217, Fr-221, Ra-225, Ac-225, and Th-229.

Table 3.13. Variation in thermal power (W/MTIHM) for significant nuclides as a function of time since discharge from a 60,000-MWd/MTIHM PWR (Includes all structural material)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Co-60[b]	1.47E+2	4.50E+1	-	-	-	-
Kr-85	2.00E+1	1.12E+1	-	-	-	-
Sr-89	1.57E+1	-	-	-	-	-
Sr-90	1.32E+2	1.06E+2	-	-	-	-
Y-90	6.29E+2	5.08E+2	5.96E+1	-	-	-
Y-91	4.38E+1	-	-	-	-	-
Zr-95[c]	1.48E+2	-	-	-	-	-
Nb-95[c]	3.16E+2	-	-	-	-	-
Ru-106	2.28E+1	-	-	-	-	-
Rh-106	3.68E+3	7.56E+0	-	-	-	-
Ag-110m	6.21E+1	-	-	-	-	-
Sb-125[c]	5.63E+1	5.34E+0	-	-	-	-
Cs-134	2.66E+3	1.29E+2	-	-	-	-
Cs-137	1.97E+2	1.60E+2	2.00E+1	-	-	-
Ba-137m	6.60E+2	5.36E+2	6.71E+1	-	-	-
Ce-144	2.84E+2	-	-	-	-	-
Pr-144	3.15E+3	-	-	-	-	-
Pm-147	3.37E+1	3.12E+0	-	-	-	-
Eu-154[c]	2.09E+2	1.01E+2	-	-	-	-
U-233	-	-	-	-	-	2.05E-2
U-234	-	-	-	1.18E-1	1.15E-1	9.10E-2
U-236	-	-	-	-	-	1.55E-2
Np-237	-	-	-	-	-	6.02E-2
Pu-238	2.84E+2	2.68E+2	1.32E+2	-	-	-
Pu-239	1.13E+1	1.13E+1	1.13E+1	1.10E+1	8.84E+0	6.90E-1
Pu-240	2.11E+1	2.15E+1	2.22E+1	2.02E+1	7.78E+0	-
Pu-241	5.84E+0	3.79E+0	-	-	-	-
Pu-242	-	-	-	1.34E-1	1.32E-1	1.12E-1
Am-241	1.92E+1	9.16E+1	1.98E+2	4.74E+1	-	-
Am-243	2.32E+0	2.32E+0	2.30E+0	2.11E+0	9.07E-1	-
Cm-242	1.01E+3	-	-	-	-	-
Cm-243	3.35E+0	2.69E+0	-	-	-	-
Cm-244	5.44E+2	3.85E+2	1.23E+1	-	-	-
OTHER	7.25E+1	7.00E+0	8.50E+0	5.18E-1	3.42E-1	6.44E-1
SUBTOTAL						
A.P.[d]	1.80E+2	4.61E+1	2.23E-1	2.35E-2	1.69E-2	9.54E-4
F.P.[e]	1.23E+4	1.57E+3	1.59E+2	3.62E-2	3.43E-2	2.10E-2
A.+D.[f]	1.90E+3	7.88E+2	3.80E+2	8.14E+1	1.81E+1	1.61E+0
TOTAL	1.44E+4	2.41E+3	5.39E+2	8.15E+1	1.81E+1	1.63E+0

[a]Nuclides contributing >0.1% of total are listed.
[b]Only activation products contribute to this nuclide.
[c]Both activation and fission products contribute to this nuclide.
[d]A.P. = Activation products.
[e]F.P. = Fission products.
[f]A.+D. = Actinides plus daughters.

Table 3.14. Variation in thermal power (W/MTIHM) for significant nuclides as a function of time since discharge from a 33,000-MWd/MTIHM PWR (Includes all structural material)

Isotope[a]	Time since discharge (years) 1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Co-60[b]	1.07E+2	3.28E+1	-	-	-	-
Kr-85	1.30E+1	7.27E+1	-	-	-	-
Sr-89	1.98E+1	-	-	-	-	-
Sr-90	8.22E+1	6.63E+1	7.79E+0	-	-	-
Y-90	3.93E+2	3.17E+2	3.72E+1	-	-	-
Y-91	5.34E+1	-	-	-	-	-
Zr-95[c]	1.59E+2	-	-	-	-	-
Nb-95[c]	3.39E+2	-	-	-	-	-
Ru-106	1.60E+1	-	-	-	-	-
Rh-106	2.57E+3	5.28E+0	-	-	-	-
Ag-110m	2.54E+1	-	-	-	-	-
Sb-125[c]	3.82E+1	4.02E+0	-	-	-	-
Cs-134	1.10E+3	5.31E+1	-	-	-	-
Cs-137	1.12E+2	9.08E+1	1.14E+1	-	-	-
Ba-137m	3.76E+2	3.05E+2	3.81E+1	-	-	-
Ce-144	2.99E+2	-	-	-	-	-
Pr-144	3.31E+3	-	-	-	-	-
Pm-147	3.67E+1	3.40E+0	-	-	-	-
Eu-154[c]	8.67E+1	4.20E+1	-	-	-	-
U-233	-	-	-	-	-	1.19E-2
U-234	-	-	-	5.84E-2	5.72E-2	4.64E-2
U-236	-	-	-	-	-	1.09E-2
Np-237	-	-	-	-	-	3.49E-2
Pu-238	8.13E+1	7.74E+1	3.71E+1	-	-	-
Pu-239	9.65E+0	9.64E+0	9.62E+0	9.39E+0	7.32E+0	5.54E-1
Pu-240	1.64E+1	1.64E+1	1.64E+1	1.49E+1	5.73E+0	-
Pu-241	3.71E+0	2.41E+0	-	-	-	-
Pu-242	-	-	-	5.08E-2	5.00E-2	4.25E-2
Am-241	1.02E+1	5.63E+1	1.24E+2	2.97E+1	-	-
Am-243	5.49E-1	5.49E-1	5.44E-1	5.00E-1	2.15E-1	-
Cm-242	3.83E+2	-	-	-	-	-
Cm-243	7.56E-1	6.08E-1	-	-	-	-
Cm-244	6.51E+1	4.62E+1	1.47E+0	-	-	-
OTHER	4.96E+1	4.70E+0	1.60E+0	1.65E-1	1.40E-1	3.57E-1
SUBTOTAL						
A.P.[d]	1.30E+2	3.35E+1	1.46E-1	1.34E-2	9.66E-3	5.64E-4
F.P.[e]	9.04E+3	8.96E+2	9.46E+1	2.01E-2	1.91E-2	1.18E-2
A.+D.[f]	5.71E+2	2.10E+2	1.91E+2	5.47E+1	1.35E+1	1.03E+0
TOTAL	9.74E+3	1.14E+3	2.86E+2	5.47E+1	1.35E+1	1.05E+0

[a]Nuclides contributing >0.1% of total are listed.
[b]Only activation products contribute to this nuclide.
[c]Both activation and fission products contribute to this nuclide.
[d]A.P. = Activation products.
[e]F.P. = Fission products.
[f]A.+D. = Actinides plus daughters.

Table 3.15. Variation in thermal power (W/MTIHM) for significant nuclides as a function of time since discharge from a 40,000-MWd/MTIHM BWR (Includes all structural material)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Co-60[b]	4.04E+1	1.24E+1	-	-	-	-
Kr-85	1.43E+1	7.97E+0	-	-	-	-
Sr-89	1.24E+1	-	-	-	-	-
Sr-90	9.51E+1	7.68E+1	9.01E+0	-	-	-
Y-90	4.54E+2	3.67E+2	4.30E+1	-	-	-
Y-91	3.38E+1	-	-	-	-	-
Zr-95[c]	1.10E+2	-	-	-	-	-
Nb-95[c]	2.35E+2	-	-	-	-	-
Ru-106	1.35E+1	-	-	-	-	-
Rh-106	2.18E+3	4.48E+0	-	-	-	-
Ag-110m	2.72E+1	-	-	-	-	-
Sb-125[c]	3.90E+1	4.10E+0	-	-	-	-
Cs-134	1.29E+3	6.26E+1	-	-	-	-
Cs-137	1.32E+2	1.07E+2	1.34E+1	-	-	-
Ba-137m	4.42E+2	3.59E+2	4.49E+1	-	-	-
Ce-144	2.03E+2	-	-	-	-	-
Pr-144	2.25E+3	-	-	-	-	-
Pm-147	3.17E+1	2.94E+0	-	-	-	-
Eu-154[c]	1.17E+2	5.64E+1	-	-	-	-
U-233	-	-	-	-	-	1.44E-2
U-234	-	-	-	7.43E-2	7.26E-2	5.83E-2
U-236	-	-	-	-	-	1.23E-2
Np-237	-	-	-	-	-	4.22E-2
Pu-238	1.34E+2	1.28E+2	6.29E+1	-	-	-
Pu-239	9.44E+0	9.44E+0	9.41E+0	9.20E+0	7.22E+0	5.51E-1
Pu-240	1.75E+1	1.76E+1	1.76E+1	1.60E+1	6.18E+0	-
Pu-241	4.24E+0	2.75E+0	-	-	-	-
Pu-242	-	-	-	6.99E-2	6.88E-2	5.85E-2
Am-241	1.45E+1	6.71E+1	1.45E+2	3.45E+1	-	-
Am-243	9.10E-1	9.09E-1	9.02E-1	8.28E-1	3.56E-1	-
Cm-242	5.91E+2	-	-	-	-	-
Cm-243	1.34E+0	1.07E+0	-	-	-	-
Cm-244	1.31E+2	9.30E+1	2.97E+0	-	-	-
OTHER	1.24E+1	7.85E+0	8.00E-1	2.96E-1	1.75E-1	4.25E-1
SUBTOTAL						
A.P.[d]	8.28E+1	1.40E+1	4.18E-2	1.20E-3	6.64E-4	1.64E-4
F.P.[e]	7.66E+3	1.05E+3	1.10E+2	2.34E-2	2.22E-2	1.38E-2
A.+D.[f]	9.05E+2	3.20E+2	2.39E+2	6.09E+1	1.40E+1	1.15E+0
TOTAL	8.65E+3	1.38E+3	3.50E+2	6.09E+1	1.41E+1	1.16E+0

[a]Nuclides contributing >0.1% of total are listed.
[b]Only activation products contribute to this nuclide.
[c]Both activation and fission products contribute to this nuclide.
[d]A.P. = Activation products.
[e]F.P. = Fission products.
[f]A.+D. = Actinides plus daughters.

Table 3.16. Variation in thermal power (W/MTIHM) for significant nuclides as a function of time since discharge from a 27,500-MWd/MTIHM BWR (Includes all structural material)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Co-60[b]	3.36E+1	1.03E+1	-	-	-	-
Kr-85	1.05E+1	5.88E+0	-	-	-	-
Sr-89	1.35E+1	-	-	-	-	-
Sr-90	6.76E+1	5.45E+1	6.40E+0	-	-	-
Y-90	3.23E+2	2.60E+2	3.06E+1	-	-	-
Y-91	3.63E+1	-	-	-	-	-
Zr-95[c]	1.14E+2	-	-	-	-	-
Nb-95[c]	2.42E+2	-	-	-	-	-
Ru-106	1.17E+1	-	-	-	-	-
Rh-106	1.89E+3	3.87E+0	-	-	-	-
Ag-110m	1.76E+1	-	-	-	-	-
Sb-125[c]	3.28E+1	3.45E+0	-	-	-	-
Cs-134	7.78E+2	3.78E+2	-	-	-	-
Cs-137	9.25E+1	7.52E+1	9.40E+0	-	-	-
Ba-137m	3.11E+2	2.52E+2	3.16E+1	-	-	-
Ce-144	2.06E+2	-	-	-	-	-
Pr-144	2.28E+3	-	-	-	-	-
Pm-147	3.12E+1	2.89E+0	-	-	-	-
Eu-154[c]	6.83E+1	3.31E+1	-	-	-	-
U-233	-	-	-	-	-	1.04E-2
U-234	-	-	-	4.83E-2	4.73E-2	3.87E-2
U-236	-	-	-	-	-	9.42E-3
Np-237	-	-	-	-	-	3.04E-2
Pu-238	6.18E+1	5.90E+1	2.91E+1	-	-	-
Pu-239	9.26E+0	9.26E+0	9.23E+0	9.01E+0	7.00E+0	5.29E-1
Pu-240	1.49E+1	1.49E+1	1.48E+1	1.35E+1	5.19E+0	-
Pu-241	3.32E+0	2.15E+0	-	-	-	-
Pu-242	-	-	-	4.18E-2	4.12E-2	3.50E-2
Am-241	1.05E+1	5.17E+1	1.12E+2	2.68E+1	-	-
Am-243	4.16E-1	4.15E-1	4.12E-1	3.78E-1	1.62E-1	-
Cm-242	3.47E+2	-	-	-	-	-
Cm-243	6.12E-1	4.92E-1	-	-	-	-
Cm-244	4.37E+1	3.10E+1	9.89E-1	-	-	-
OTHER	2.47E+1	6.32E+0	6.00E-1	1.25E-1	1.14E-1	2.92E-1
SUBTOTAL						
A.P.[d]	7.42E+2	1.19E+1	3.18E-2	8.92E-4	5.02E-4	1.24E-4
F.P.[e]	6.50E+3	7.30E+2	7.80E+1	1.65E-2	1.57E-2	9.78E-3
A.+D.[f]	4.92E+2	1.69E+2	1.68E+2	4.99E+1	1.25E+1	9.35E-1
TOTAL	7.07E+3	9.11E+2	2.46E+2	4.99E+1	1.26E+1	9.45E-1

[a]Nuclides contributing >0.1% of total are listed.
[b]Only activation products contribute to this nuclide.
[c]Both activation and fission products contribute to this nuclide.
[d]A.P. = Activation products.
[e]F.P. = Fission products.
[f]A.+D. = Actinides plus daughters.

Table 3.17. Variation in neutron production (neutrons/s•MTIHM) by spontaneous fission as a function of time since discharge from a 60,000-MWd/MTIHM PWR
(Includes all structural material)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
U-238	-	-	1.16E+4	1.16E+4	1.16E+4	1.16E+4
Pu-238	-	-	6.18E+5	5.59E+2	-	-
Pu-240	2.71E+6	2.76E+6	2.85E+6	2.59E+6	9.98E+5	7.17E+1
Pu-242	2.00E+6	2.00E+6	2.00E+6	2.00E+6	1.97E+6	1.68E+6
Cm-242	1.79E+8	9.11E+4	6.02E+4	-	-	-
Cm-244	2.14E+9	1.51E+9	4.83E+7	-	-	-
Cm-246	2.11E+7	2.11E+7	2.08E+7	1.82E+7	4.88E+7	9.15E+0
Cm-248	-	-	1.62E+5	1.62E+5	1.59E+5	1.32E+5
Cf-252	9.45E+6	8.88E+5	-	-	-	-
TOTAL	2.35E+9	1.54E+9	7.48E+7	2.30E+7	8.02E+6	1.82E+6

[a]Nuclides contributing >0.1% are listed.

Table 3.18. Variation in neutron production (neutrons/s•MTIHM) by spontaneous fission as a function of time since discharge from a 33,000-MWd/MTIHM PWR
(Includes all structural material)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
U-238	-	-	-	1.20E+4	1.20E+4	1.20E+4
Pu-238	3.80E+5	3.62E+5	1.78E+5	1.68E+2	-	-
Pu-240	2.10E+6	2.10E+6	2.10E+6	1.91E+6	7.35E+5	5.27E+1
Pu-242	7.60E+5	7.60E+5	7.60E+5	7.59E+5	7.47E+5	6.36E+5
Cm-242	6.78E+7	3.72E+4	2.46E+4	-	-	-
Cm-244	2.56E+8	1.81E+8	5.79E+6	-	-	-
Cm-246	9.06E+5	9.04E+5	8.92E+5	7.82E+5	2.09E+5	-
Cm-248	-	-	-	1.93E+3	1.89E+3	1.57E+3
TOTAL	3.28E+8	1.86E+8	9.76E+6	3.46E+6	1.70E+6	6.49E+5

[a]Nuclides contributing >0.1% are listed.

Table 3.19. Variation in neutron production (neutrons/s•MTIHM) by spontaneous fission as a function of time since discharge from a 40,000-MWd/MTIHM BWR

(Includes all structural material)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
U-238	-	-	1.19E+4	1.19E+4	1.19E+4	1.19E+4
Pu-238	6.29E+5	5.98E+5	2.94E+5	2.82E+2	-	-
Pu-240	2.25E+6	2.26E+6	2.26E+6	2.06E+6	7.93E+5	5.70E+1
Pu-242	1.04E+6	1.04E+6	1.04E+6	1.04E+6	1.03E+6	8.75E+5
Cm-242	1.04E+8	7.11E+4	4.70E+4	7.76E+2	-	-
Cm-244	5.15E+8	3.65E+8	1.16E+7	-	-	-
Cm-246	2.58E+6	2.58E+6	2.55E+6	2.32E+6	5.97E+5	-
Cm-248	-	-	8.58E+3	8.56E+3	8.41E+3	7.00E+3
TOTAL	6.27E+8	3.72E+8	1.79E+7	5.36E+6	2.44E+6	8.94E+5

[a]Nuclides contributing >0.1% are listed.

Table 3.20. Variation in neutron production (neutrons/s•MTIHM) by spontaneous fission as a function of time since discharge from a 27,500-MWd/MTIHM BWR

(Includes all structural material)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
U-238	-	-	1.21E+4	1.21E+4	1.21E+4	1.21E+4
Pu-238	2.89E+5	2.76E+5	1.36E+5	1.38E+2	-	-
Pu-240	1.91E+6	1.91E+6	1.90E+6	1.73E+6	6.66E+5	4.77E+1
Pu-242	6.26E+5	6.26E+5	6.26E+5	6.25E+5	6.15E+5	5.24E+5
Cm-242	6.14E+7	4.47E+4	2.96E+4	4.89E+2	-	-
Cm-244	1.72E+8	1.22E+8	3.88E+6	-	-	-
Cm-246	5.01E+5	5.01E+5	4.94E+5	4.33E+5	1.16E+5	-
Cm-248	-	-	-	8.70E+2	8.54E+2	7.10E+2
TOTAL	2.36E+8	1.25E+8	7.08E+6	2.80E+6	1.41E+6	5.37E+5

[a]Nuclides contributing >0.1% are listed.

Table 3.21. Variation in neutron production (neutrons/s•MTIHM) by the alpha-neutron reaction as a function of time since discharge from a 60,000-MWd/MTIHM PWR (Includes all structural material)

Isotope[a]	Time since discharge (years)					
	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Po-210	-	-	-	-	2.26E+2	1.79E+3
Po-213	-	-	-	-	-	1.72E+3
Po-214	-	-	-	-	6.33E+2	5.00E+3
Po-218	-	-	-	-	3.42E+2	2.70E+3
At-217	-	-	-	-	-	1.28E+3
Rn-222	-	-	-	-	2.55E+2	2.01E+3
Fr-221	-	-	-	-	-	9.88E+2
Ra-226	-	-	-	-	1.50E+2	1.19E+3
Ac-225	-	-	-	-	-	7.32E+2
Th-229	-	-	-	-	-	4.56E+2
Th-230	-	-	-	-	1.78E+2	1.08E+3
U-233	-	-	-	-	-	4.07E+2
U-234	-	-	-	2.27E+3	2.22E+3	1.76E+3
U-236	-	-	-	-	2.14E+2	2.43E+2
U-238	-	-	-	-	-	9.67E+1
Np-237	-	-	-	1.23E+3	1.43E+3	1.39E+3
Pu-238	8.14E+6	7.71E+6	3.79E+6	3.43E+3	-	-
Pu-239	2.67E+5	2.67E+5	2.67E+5	2.61E+5	2.09E+5	1.63E+4
Pu-240	5.14E+5	5.23E+5	5.40E+5	4.92E+5	1.89E+5	-
Pu-242	-	-	-	2.79E+3	2.75E+3	2.34E+3
Am-241	5.53E+5	2.65E+6	5.73E+6	1.37E+6	8.03E+2	-
Am-243	6.16E+4	6.15E+4	6.10E+4	5.60E+4	2.41E+4	-
Cm-242	3.69E+7	1.88E+4	1.24E+4	-	-	-
Cm-243	1.22E+5	9.77E+4	1.10E+4	-	-	-
Cm-244	1.77E+7	1.26E+7	4.01E+5	1.66E+3	7.98E+2	-
TOTAL	6.43E+7	2.39E+7	1.08E+7	2.19E+6	4.33E+5	4.17E+4

[a]Nuclides contributing >0.1% are listed.

Table 3.22. Variation in neutron production (neutrons/s•MTIHM) by the alpha-neutron reaction as a function of time since discharge from a 33,000-MWd/MTIHM PWR (Includes all structural material)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Po-210	-	-	-	-	4.40E+1	9.00E+2
Po-213	-	-	-	-	-	9.99E+2
Po-214	-	-	-	-	3.16E+2	2.52E+3
Po-218	-	-	-	-	1.71E+2	1.36E+3
At-217	-	-	-	-	-	7.40E+2
Rn-222	-	-	-	-	1.27E+2	1.01E+3
Fr-221	-	-	-	-	-	5.73E+2
Ra-226	-	-	-	-	7.52E+1	5.98E+2
Ac-225	-	-	-	-	-	4.24E+2
Th-229	-	-	-	-	-	2.64E+2
Th-230	-	-	-	-	8.87E+1	5.45E+2
U-233	-	-	-	-	-	2.36E+2
U-234	-	-	-	1.13E+3	1.10E+3	8.94E+2
U-236	-	-	-	-	1.50E+2	1.72E+2
U-238	-	-	-	-	-	9.99E+1
Np-237	-	-	-	7.04E+2	8.30E+2	8.06E+2
Pu-238	2.33E+6	2.12E+6	1.09E+6	1.03E+3	-	-
Pu-239	2.28E+5	2.28E+5	2.27E+5	2.22E+5	1.73E+5	1.31E+4
Pu-240	3.99E+5	4.00E+5	3.98E+5	3.62E+5	1.40E+5	-
Pu-242	-	-	-	1.06E+3	1.04E+3	8.86E+2
Am-241	2.95E+5	2.23E+6	3.59E+6	8.57E+5	-	-
Am-243	1.46E+4	1.46E+4	1.44E+4	1.33E+4	5.70E+3	-
Cm-242	1.40E+7	7.45E+3	5.08E+3	-	-	-
Cm-243	2.74E+4	2.20E+4	2.47E+3	-	-	-
Cm-244	2.12E+6	1.51E+6	4.81E+4	-	-	-
TOTAL	1.94E+7	6.03E+6	5.38E+6	1.46E+6	3.22E+5	2.63E+4

[a]Nuclides contributing >0.1% are listed.

Table 3.23. Variation in neutron production (neutrons/s•MTIHM) by the alpha-neutron reaction as a function of time since discharge from a 40,000-MWd/MTIHM BWR (Includes all structural material)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Po-210	-	-	-	-	1.43E+2	1.14E+3
Po-213	-	-	-	-	-	1.21E+3
Po-214	-	-	-	-	4.01E+2	3.18E+3
Po-218	-	-	-	-	2.17E+2	1.72E+3
At-217	-	-	-	-	-	8.93E+2
Rn-222	-	-	-	-	1.61E+2	1.28E+3
Fr-221	-	-	-	-	-	6.92E+2
Ra-226	-	-	-	-	9.53E+1	7.56E+2
Ac-225	-	-	-	-	-	5.12E+2
Th-229	-	-	-	-	-	3.19E+2
Th-230	-	-	-	-	1.12E+2	6.88E+2
U-233	-	-	-	-	-	2.85E+2
U-234	-	-	-	1.43E+3	1.40E+3	1.12E+3
U-236	-	-	-	-	1.71E+2	1.94E+2
U-238	-	-	-	-	-	9.92E+1
Np-237	-	-	-	8.56E+2	1.00E+3	9.73E+2
Pu-238	3.86E+6	3.67E+6	1.80E+6	1.73E+3	-	-
Pu-239	2.23E+5	2.23E+5	2.22E+5	2.17E+5	1.70E+5	1.30E+4
Pu-240	4.26E+5	4.28E+5	4.30E+5	3.91E+5	1.50E+5	-
Pu-242	-	-	-	1.46E+3	1.43E+3	1.22E+3
Am-241	4.19E+5	1.94E+6	4.18E+6	9.96E+5	1.42E+2	-
Am-243	2.42E+4	2.41E+4	2.39E+4	2.20E+4	9.44E+3	-
Cm-242	2.15E+7	1.46E+4	9.69E+3	-	-	-
Cm-243	4.84E+4	3.89E+4	4.36E+3	-	-	-
Cm-244	4.28E+6	3.03E+6	9.68E+4	-	-	-
TOTAL	3.08E+7	9.37E+6	6.77E+6	1.63E+6	3.36E+5	2.95E+4

[a]Nuclides contributing >0.1% are listed.

Table 3.24. Variation in neutron production (neutrons/s•MTIHM) by the alpha-neutron reaction as a function of time since discharge from a 27,500-MWd/MTIHM BWR
(Includes all structural material)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Po-210	-	-	-	-	9.35E+1	7.48E+2
Po-213	-	-	-	-	-	8.70E+2
Po-214	-	-	-	-	2.62E+2	2.09E+3
Po-218	-	-	-	-	1.41E+2	1.13E+3
At-217	-	-	-	-	-	6.44E+2
Rn-222	-	-	-	-	1.05E+2	8.42E+2
Fr-221	-	-	-	-	-	4.99E+2
Ra-226	-	-	-	-	6.21E+1	4.97E+2
Ac-225	-	-	-	-	-	3.70E+2
Th-229	-	-	-	-	-	2.30E+2
Th-230	-	-	-	-	7.34E+1	4.53E+2
U-233	-	-	-	-	-	2.06E+2
U-234	-	-	-	9.31E+2	9.12E+2	7.47E+2
U-236	-	-	-	-	1.29E+2	1.48E+2
U-238	-	-	-	-	1.01E+2	1.01E+2
Np-237	-	-	-	6.10E+2	7.23E+2	7.02E+2
Pu-238	1.77E+6	1.69E+6	8.35E+5	8.44E+2	-	-
Pu-239	2.19E+5	2.19E+5	2.18E+5	2.13E+5	1.66E+5	1.25E+4
Pu-240	3.62E+5	3.62E+5	3.61E+5	3.28E+5	1.26E+5	-
Pu-242	-	-	-	8.71E+2	8.57E+2	7.30E+2
Am-241	3.02E+5	1.49E+6	3.25E+6	7.74E+5	3.98E+1	-
Am-243	1.10E+4	1.10E+4	1.09E+4	1.00E+4	4.31E+3	-
Cm-242	1.26E+7	9.22E+3	6.10E+3	-	-	-
Cm-243	2.22E+4	1.78E+4	2.00E+3	-	-	-
Cm-244	1.43E+6	1.01E+6	3.22E+4	-	-	-
TOTAL	1.68E+7	4.82E+6	4.72E+6	1.33E+6	3.00E+5	2.37E+4

[a]Nuclides contributing >0.1% are listed.

Table 3.25. Variation in photon production (photons/s•MTIHM) as a function of time since discharge from a 60,000-MWd/MTIHM PWR (Includes all structural material)

	Time since discharge (years)					
Emean[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
1.00E-2	2.97E+16	3.21E+15	4.22E+14	1.92E+13	3.50E+12	4.09E+11
2.50E-2	6.75E+15	6.71E+14	7.50E+13	1.41E+12	5.83E+10	4.35E+10
3.75E-2	6.72E+15	8.75E+14	8.68E+13	2.97E+11	8.72E+10	2.47E+10
5.75E-2	6.15E+15	6.33E+14	1.46E+14	1.97E+13	5.84E+10	2.71E+10
8.50E-1	4.31E+15	3.84E+14	3.97E+13	1.91E+12	8.58E+11	9.75E+10
1.25E-1	4.83E+15	4.09E+14	2.55E+13	1.22E+12	5.26E+11	1.61E+10
2.25E-1	3.78E+15	3.12E+14	3.20E+13	8.12E+11	3.56E+11	4.55E+10
3.75E-1	2.10E+15	1.45E+14	1.33E+13	1.64E+11	1.24E+11	9.78E+10
5.75E-1	2.50E+16	5.94E+15	6.58E+14	1.32E+11	1.29E+11	1.10E+11
8.50E-1	1.28E+16	6.32E+14	2.40E+12	1.57E+11	1.18E+11	1.90E+10
1.25E+0	2.28E+15	4.70E+14	8.49E+11	1.78E+09	4.97E+09	2.77E+10
1.75E+0	1.15E+14	7.21E+12	5.85E+10	7.84E+07	2.77E+09	2.21E+10
2.25E+0	1.42E+14	8.42E+10	2.09E+07	2.28E+07	8.45E+08	6.66E+09
2.75E+0	3.16E+12	7.76E+09	6.47E+08	3.07E+06	1.57E+07	1.16E+08
3.50E+0	3.99E+11	9.77E+08	7.72E+06	2.41E+06	3.62E+06	2.20E+07
5.00E+0	1.04E+08	6.76E+07	3.30E+06	1.01E+06	3.72E+05	1.00E+05
7.00E+0	1.20E+07	7.80E+06	3.79E+05	1.16E+05	4.28E+04	1.16E+04
9.50E+0	1.37E+06	8.96E+05	4.35E+04	1.33E+04	4.92E+03	1.33E+03
TOTAL	1.05E+17	1.37E+16	1.50E+15	4.50E+13	5.82E+12	9.47E+11

[a]Energy is given in MeV and covers a range which is equal distance between the preceeding and following value.

Table 3.26. Variation in photon production (photons/s•MTIHM) as a function of time since discharge from a 33,000-MWd/MTIHM PWR
(Includes all structural material)

	Time since discharge (years)					
Emean[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
1.00E-2	2.46E+16	1.91E+15	2.50E+14	1.14E+13	1.96E+12	2.41E+11
2.50E-2	5.55E+15	4.13E+14	4.61E+13	8.68E+11	2.86E+10	2.45E+10
3.75E-2	5.69E+15	4.95E+14	5.14E+13	1.17E+11	2.47E+10	1.41E+10
5.75E-2	5.11E+15	3.82E+14	9.07E+13	1.23E+13	2.14E+10	1.53E+10
8.50E-1	3.61E+15	2.24E+14	2.37E+13	4.76E+11	2.26E+11	5.51E+10
1.25E-1	4.26E+15	2.12E+14	1.52E+13	2.95E+11	1.27E+11	9.33E+09
2.25E-1	3.15E+15	1.86E+14	1.95E+13	1.94E+11	8.77E+10	2.46E+10
3.75E-1	1.72E+15	9.09E+13	8.22E+12	6.58E+10	5.76E+10	5.32E+10
5.75E-1	1.33E+16	3.29E+15	3.74E+14	7.00E+10	6.82E+10	5.69E+10
8.50E-1	7.55E+15	2.65E+14	1.46E+12	8.85E+10	6.64E+10	1.00E+10
1.25E+0	1.36E+15	2.64E+14	4.97E+11	9.65E+08	2.56E+09	1.40E+10
1.75E+0	7.70E+13	3.13E+12	3.55E+10	4.49E+07	1.39E+09	1.12E+10
2.25E+0	1.42E+14	7.27E+10	5.96E+06	1.07E+07	4.22E+08	3.36E+09
2.75E+0	2.25E+12	4.78E+09	1.76E+08	6.96E+05	7.58E+06	5.85E+07
3.50E+0	2.79E+11	5.91E+08	1.12E+06	4.68E+05	1.60E+06	1.10E+07
5.00E+0	1.46E+07	8.19E+06	4.74E+05	1.86E+05	9.35E+04	3.67E+04
7.00E+0	1.69E+06	9.44E+05	5.40E+04	2.12E+04	1.07E+04	4.22E+03
9.50E+0	1.94E+05	1.08E+05	6.17E+03	2.43E+03	1.23E+03	4.86E+02
TOTAL	7.61E+16	7.73E+15	8.80E+14	2.59E+13	2.67E+12	5.33E+11

[a]Energy is given in MeV and covers a range which is equal distance between the preceding and following value.

Table 3.27. Variation in photon production (photons/s•MTIHM) as a function of time since discharge from a 40,000-MWd/MTIHM BWR
(Includes all structural material)

	Time since discharge (years)					
Emean[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
1.00E-2	1.95E+16	2.22E+15	2.94E+14	1.33E+13	2.26E+12	2.86E+11
2.50E-2	4.48E+15	4.73E+14	5.35E+13	1.01E+12	3.41E+10	2.90E+10
3.75E-2	4.48E+15	5.85E+14	6.00E+13	1.56E+11	3.72E+10	1.69E+10
5.75E-2	4.04E+15	4.44E+14	1.05E+14	1.43E+13	2.75E+10	1.83E+10
8.50E-1	2.83E+15	2.63E+14	2.77E+13	7.68E+11	3.55E+11	6.61E+10
1.25E-1	3.23E+15	2.60E+14	1.78E+13	4.82E+11	2.08E+11	1.12E+10
2.25E-1	2.48E+15	2.17E+14	2.27E+13	3.20E+11	1.42E+11	3.01E+10
3.75E-1	1.38E+15	1.03E+14	9.53E+12	8.56E+10	7.12E+10	6.39E+10
5.75E-1	1.40E+16	3.86E+15	4.40E+14	8.01E+10	7.83E+10	6.77E+10
8.50E-1	7.11E+15	3.25E+14	1.60E+12	7.18E+09	6.70E+09	8.87E+09
1.25E+0	1.06E+15	2.00E+14	5.87E+11	1.10E+09	3.13E+09	1.77E+10
1.75E+0	6.70E+13	4.11E+12	4.14E+10	5.29E+07	1.76E+09	1.41E+10
2.25E+0	9.85E+13	5.42E+10	8.17E+06	1.33E+07	5.35E+08	4.24E+09
2.75E+0	1.89E+12	4.26E+09	2.32E+08	9.64E+05	9.64E+06	7.39E+07
3.50E+0	2.37E+11	5.24E+08	1.96E+06	6.72E+05	2.05E+06	1.40E+07
5.00E+0	2.79E+07	1.64E+07	8.33E+05	2.70E+05	1.28E+05	5.04E+04
7.00E+0	3.21E+06	1.89E+06	9.54E+04	3.09E+04	1.48E+04	5.80E+03
9.50E+0	3.69E+05	2.17E+05	1.09E+04	3.54E+03	1.70E+03	6.67E+02
TOTAL	6.48E+16	8.96E+15	1.03E+15	3.05E+13	3.23E+12	6.34E+11

[a]Energy is given in MeV and covers a range which is equal distance between the preceding and following value.

Table 3.28. Variation in photon production (photons/s•MTIHM) as a function of time since discharge from a 27,500-MWd/MTIHM BWR (Includes all structural material)

	Time since discharge (years)					
Emean[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
1.00E-2	1.76E+16	1.56E+15	2.08E+14	1.02E+13	1.75E+12	2.12E+11
2.50E-2	4.06E+15	3.40E+14	3.82E+13	7.81E+11	2.23E+10	2.07E+10
3.75E-2	4.08E+15	4.06E+14	4.24E+13	9.89E+10	1.92E+10	1.21E+10
5.75E-2	3.67E+15	3.15E+14	7.88E+13	1.11E+13	1.86E+10	1.31E+10
8.50E-1	2.58E+15	1.83E+14	1.95E+13	3.65E+11	1.75E+11	4.72E+10
1.25E-1	3.01E+15	1.71E+14	1.25E+13	2.25E+11	9.72E+10	8.10E+09
2.25E-1	2.26E+15	1.52E+14	1.60E+13	1.47E+11	6.69E+10	2.09E+10
3.75E-1	1.25E+15	7.51E+13	6.76E+12	5.32E+10	4.74E+10	4.48E+10
5.75E-1	9.95E+15	2.69E+15	3.10E+14	5.64E+10	5.49E+10	4.64E+10
8.50E-1	5.39E+15	1.96E+14	1.13E+12	5.15E+09	4.82E+09	5.99E+09
1.25E+0	7.79E+14	1.33E+14	4.06E+11	7.94E+08	2.12E+09	1.17E+10
1.75E+0	5.57E+13	2.47E+12	2.91E+10	3.94E+07	1.15E+09	9.28E+09
2.25E+0	9.80E+13	5.11E+10	4.74E+06	8.88E+06	3.49E+08	2.79E+09
2.75E+0	1.64E+12	3.46E+09	1.02E+08	5.69E+05	6.27E+06	4.86E+07
3.50E+0	2.04E+11	4.32E+08	8.31E+05	3.88E+05	1.32E+06	9.18E+06
5.00E+0	1.06E+07	5.53E+06	3.52E+05	1.54E+05	7.84E+04	3.04E+04
7.00E+0	1.22E+06	6.37E+05	4.00E+04	1.76E+04	8.98E+03	3.49E+03
9.50E+0	1.41E+05	7.32E+04	4.57E+03	2.01E+03	1.03E+03	4.02E+02
TOTAL	5.48E+16	6.23E+15	7.33E+14	2.30E+13	2.26E+12	4.55E+11

[a]Energy is given in MeV and covers a range which is equal distance between the preceding and following value.

4. CONSEQUENCES OF FUEL CONSOLIDATION

Spent fuel storage capacities at several commercial LWRs are inadequate to handle projected fuel discharges.[1] To alleviate this problem, many organizations are investigating methods for increasing storage capacity by consolidating the fuel pins from several fuel assemblies.[2] Such an operation will produce two separate categories of radioactive material that must be considered during waste disposal. The first group comprises the zircaloy tubes that contain residual fuel, the fission products, and the actinides and their daughters. The second category consists of the remaining structural material that holds the fuel rods in a rigid configuration. Instrument tubes, guide tubes, and other materials of construction may be found in this waste. An additional item that is used in BWR fuel assemblies is the fuel channel, which is a box that provides a barrier to separate coolant flow paths, to guide the control rods, and to provide rigidity and protection for the fuel bundle during handling. Nearly all of the activation products will be found in the second category.

4.1 CONSOLIDATED FUEL PINS

By far, the fuel pins in a spent fuel assembly contain the major quantity of radioactivity and, in this discussion, are assumed to be consolidated and placed in specially designed containers. The latest of several configurations that have been considered for consolidated spent fuel is a cylinder that is divided into six pie-shaped sections, each of which will contain pins from either two PWR assemblies (the reference is a 17 × 17 Westinghouse assembly) or five BWR assemblies (the reference is an 8 × 8 G.E. assembly).[3] The container has been designed to store fuel that is at least 10 years old and to dissipate heat that does not exceed 5700 W for consolidated BWR pins or 6600 W for consolidated PWR pins. Using these criteria and the quantity of heavy metal contained in LWR fuel assemblies (see Table 3.1, Sect. 3), one can convert the values presented for MTIHM in the tables in Sect. 3 to results that can be applied to the reference design fuel container. These conversion factors are 5.50 (30 assemblies × 0.1833 MTIHM/assembly) for the BWR case and 5.54 (12 assemblies × 0.4614 MTIHM/assembly) for the PWR case. The present system is adequate to store model LWR fuel with respect to thermal power as 30 assemblies of 10-year-old BWR fuel (burnup = 27,500 MWd/MTIHM) generate 5010 W, and 12 assemblies of comparable PWR fuel (burnup = 33,000 MWd/MTIHM) will generate 6320 W. However, extended-burnup fuel of the same age can not be placed safely in the current canister design since its heat dissipation properties will be exceeded by both BWR (7600 W) and PWR (13,350 W) consolidated fuel with burnups of 40,000 and 60,000 MWd/MTIHM, respectively.

4.2 NONFUEL COMPONENTS

In any fuel consolidation operation, the top and bottom of the fuel assembly, which are structural members, must be cut free so that the fuel rods are exposed for removal. The rods may then be extracted from the remaining assembly by any one of several processes now being considered. The structural remains may then be combined with the end fittings, compacted, and stored in a suitable manner. These items, as well as control spiders, burnable poison rod assemblies, control rod elements, thimble plugs, fission chambers, and primary and secondary neutron sources, are discussed in Appendix E of the proposed contractual agreement between DOE and owners and generators of spent fuel.[4] The contract states that these materials may be included as part of spent nuclear fuel and would be acceptable by DOE for disposal in a repository.

The material remaining after the removal of fuel pins from the Westinghouse 17 × 17 PWR design includes upper and lower end fittings, leaf springs, grid spacers, and other miscellaneous hardware. The alloys used in their manufacture are SS-304, Inconel-718, Zircaloy-2, Zircaloy-4, and Nicrobraze-50 (see Tables 3.2-3.4, Sect. 3). The total weight of these alloys is 23.4 kg or 52.6 kg/MTIHM. The results of an ORIGEN2 decay calculation for the quantity of radioactivity (Fig. 4.1) and heat produced (Fig. 4.2) show that the activity decreases by a factor of 30 and the heat output by a factor of 100 after a century of storage. The major contributing isotopes (all activation products) are ^{59}Ni (1.5%) and ^{63}Ni (98%) for radioactivity and ^{63}Ni (91%) and ^{94}Nb (9%) for thermal power. For decay periods exceeding 1000 years, ^{59}Ni and ^{94}Nb present the greatest hazard.

Disassembly of the standard 8 × 8 G.E. BWR fuel assembly produces two distinct classes of waste, the structural material and the fuel channels. The first group contains the upper and lower end fittings and tie plates, grid spacers, and various springs. Materials of construction are similar to those used for PWRs, and the total weight is ~9.5 kg per assembly (52 kg/MTIHM). ORIGEN2 calculations for selected decay periods produce similar shaped curves (Figs. 4.3 and 4.4) as were obtained for the PWR cases, although displaying somewhat lower radioactivity and heat output.

The fuel channel, weighing ~42 kg, forms the second group of BWR nonfuel components. In most cases, it remains with a single fuel assembly throughout its lifetime (assumed mode in this study). However, there have been studies to examine the feasibility of using fuel channels for two fuel assembly lifetimes which would increase the activation products contained therein.[5] Utilization of Zircaloy-4 as construction material decreases the radioactivity (Fig. 4.5) after 2 years of decay and decreases it dramatically after 20 years when compared to that obtained for the miscellaneous fuel assembly parts. The plots (Fig. 4.6) for thermal output exhibit similar behavior. The major contributor to the thermal output for decay periods >1000 years is the parent-daughter combination of nuclides, $^{93}Zr \rightarrow {}^{93m}Nb$.

4.3 CONCLUSIONS

The nonfuel components may be considered to be on the borderline as to the need for their placement in a geologic repository. As discussed by Luksic and McKee,[6] the primary concern in the decision for geologic disposal of nonfuel components is the concentration of various impurities in the structural material. For example, if the concentration of natural niobium is in the range of 20 to 30 ppm for components in the active core or 50 to 100 ppm for those outside of the active core, the ^{94}Nb concentration will exceed that specified in 10 CFR Part 61 for land disposal.[7] Since these concentrations are given in Ci/m^3, the compaction or volume reduction possible for these components will have a strong influence on the possibility of placing them in a low-level burial ground.

4.4 REFERENCES FOR SECT. 4

1. R. A. Libby and G. M. Holter, "Projected Commercial Spent Nuclear Fuel Discharges and Additional Storage Requirements," Trans. Am. Nucl. Soc. 49, 94-95 (1985).
2. W. J. Bailey and G. H. Beeman, "Status of Consolidation of LWR Spent Fuel," Trans. Am. Nucl. Soc. 49, 95-96 (1985).
3. J. Cavanaugh, Battelle, Columbus, Ohio, telephone call to J. W. Roddy, Oak Ridge National Laboratory, Oak Ridge, Tenn., June 13, 1985.
4. Code of Federal Regulations, Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste, 10 CFR Pt. 961, pp. 521-38 (January 1985).
5. J. A. Gorman and G. W. Lipsey, Assessment of BWR Fuel-Channel Lifetimes. Final Report, EPRI-NP-2483, July 1982.
6. A. T. Luksic and R. W. McKee, Non-Fuel Hardware Characterization: Disposal Concerns, PNL-SA-12545, February 1985.
7. Code of Federal Regulations, Licensing Requirements for Land Disposal of Radioactive Waste, 10 CFR Pt. 61, pp. 593-617 (January 1985).

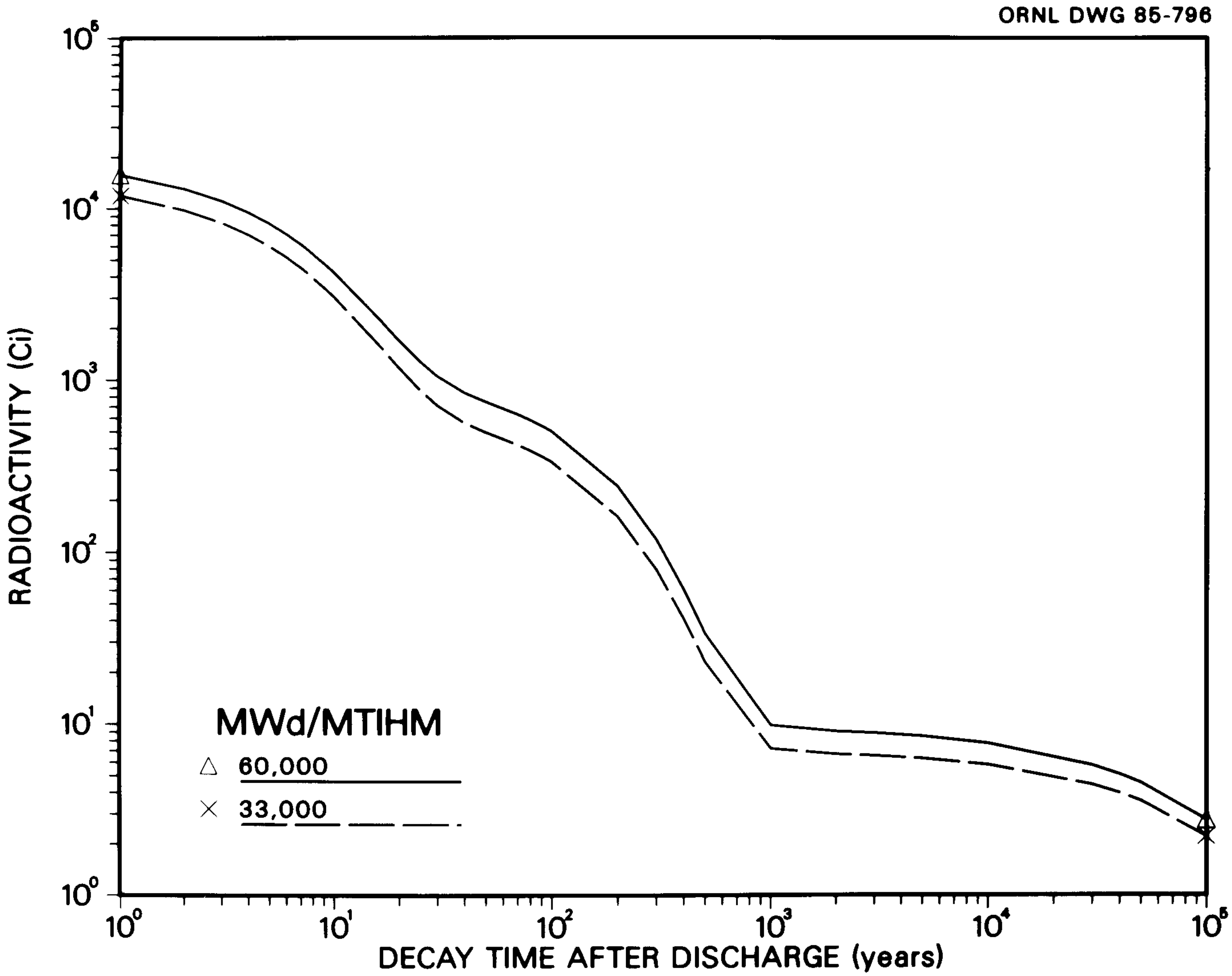


Fig. 4.1. Radioactivity from miscellaneous PWR hardware associated with 1 metric ton of initial heavy metal.

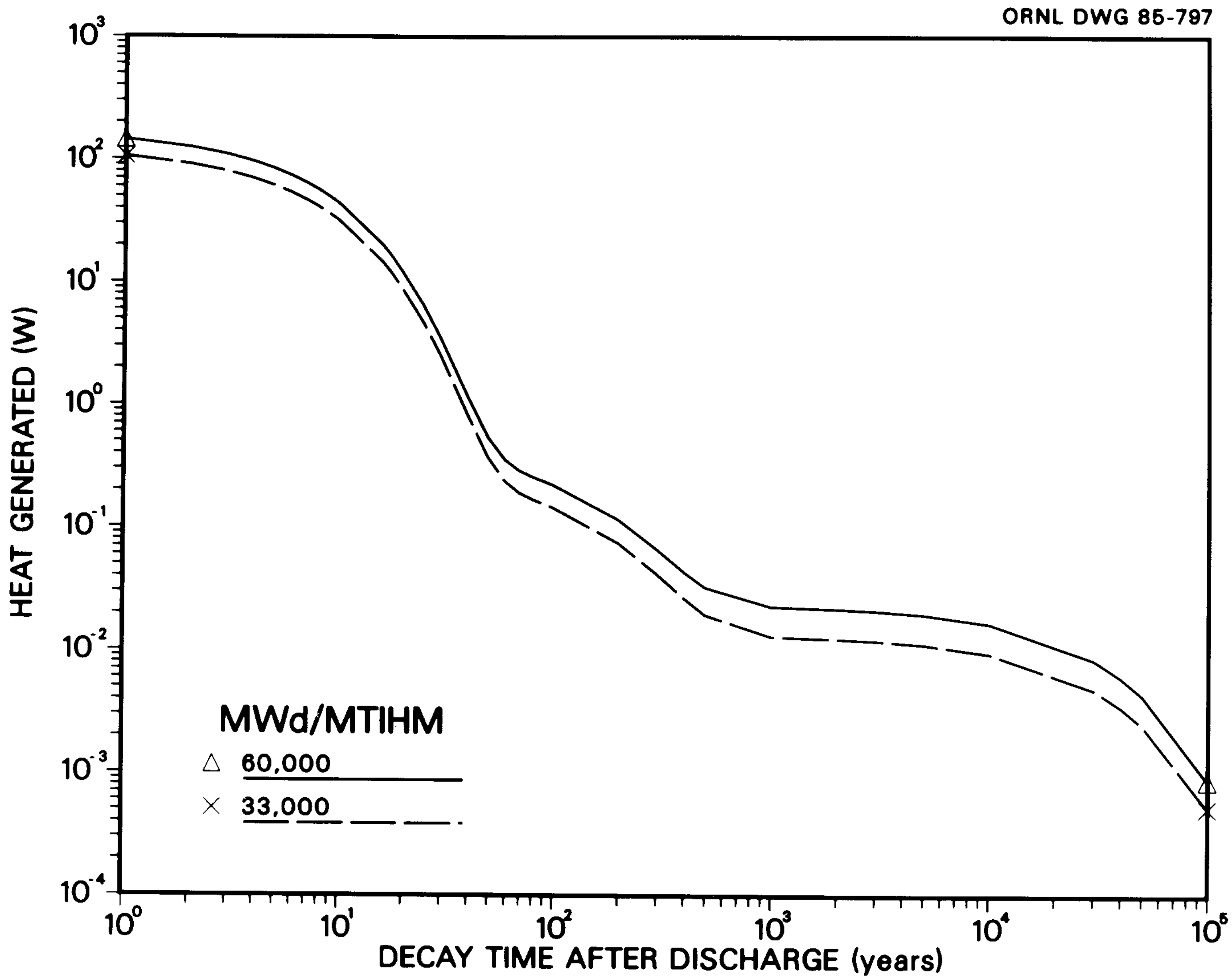


Fig. 4.2. Heat generated from miscellaneous PWR hardware associated with 1 metric ton of initial heavy metal.

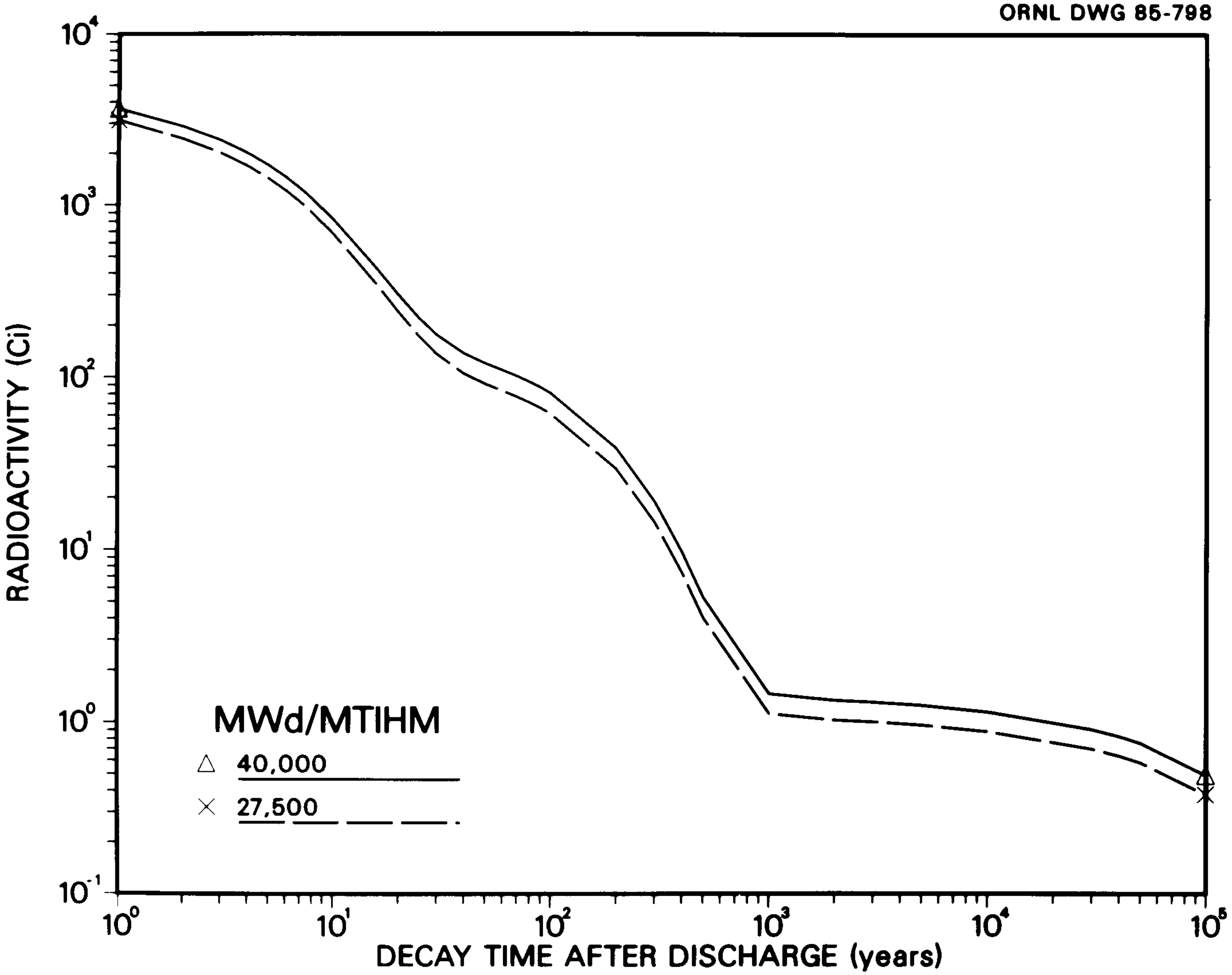


Fig. 4.3. Radioactivity from miscellaneous BWR hardware (excluding fuel channels) associated with 1 metric ton of initial heavy metal.

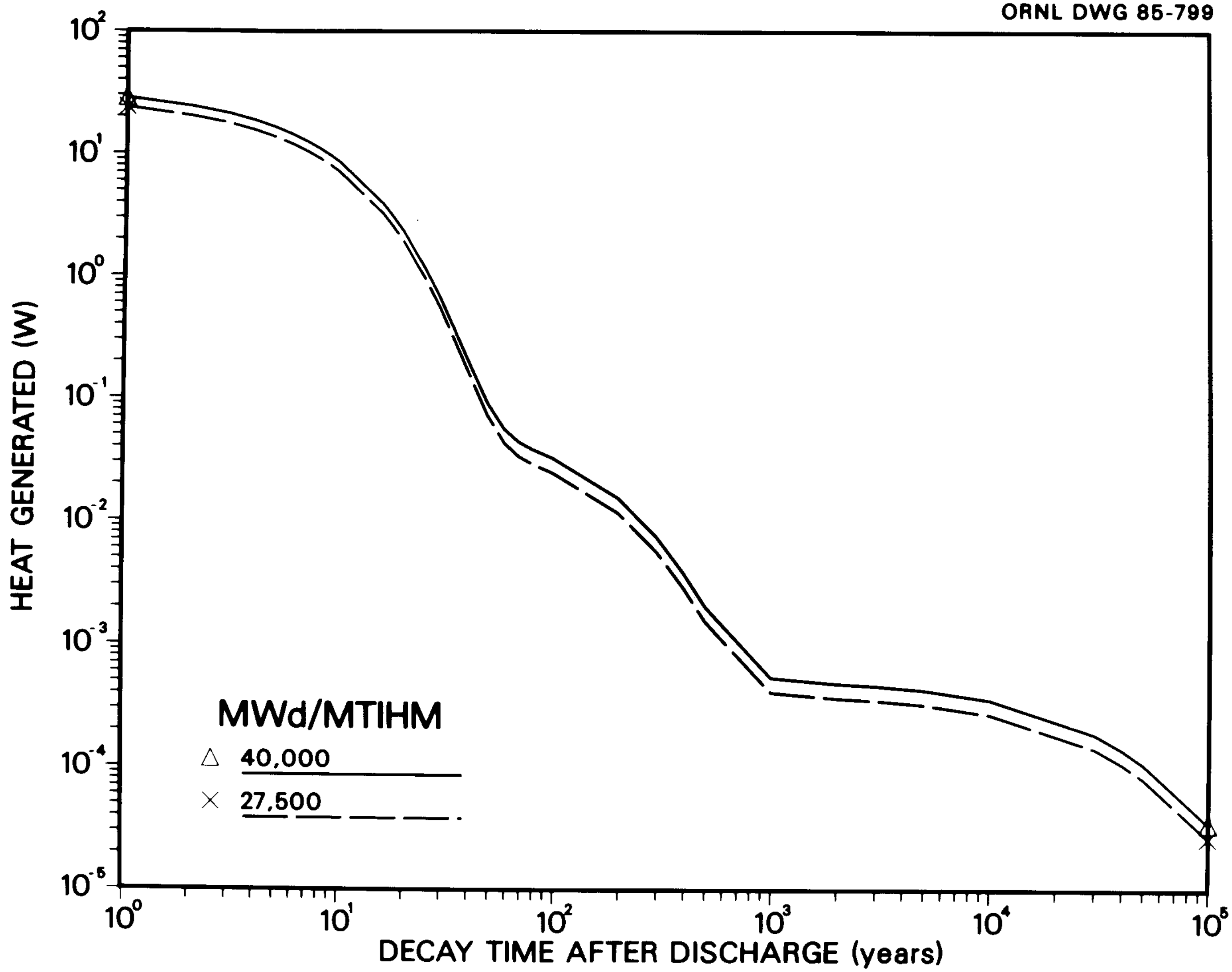


Fig. 4.4. Heat generated from miscellaneous BWR hardware (excluding fuel channels) associated with 1 metric ton of initial heavy metal.

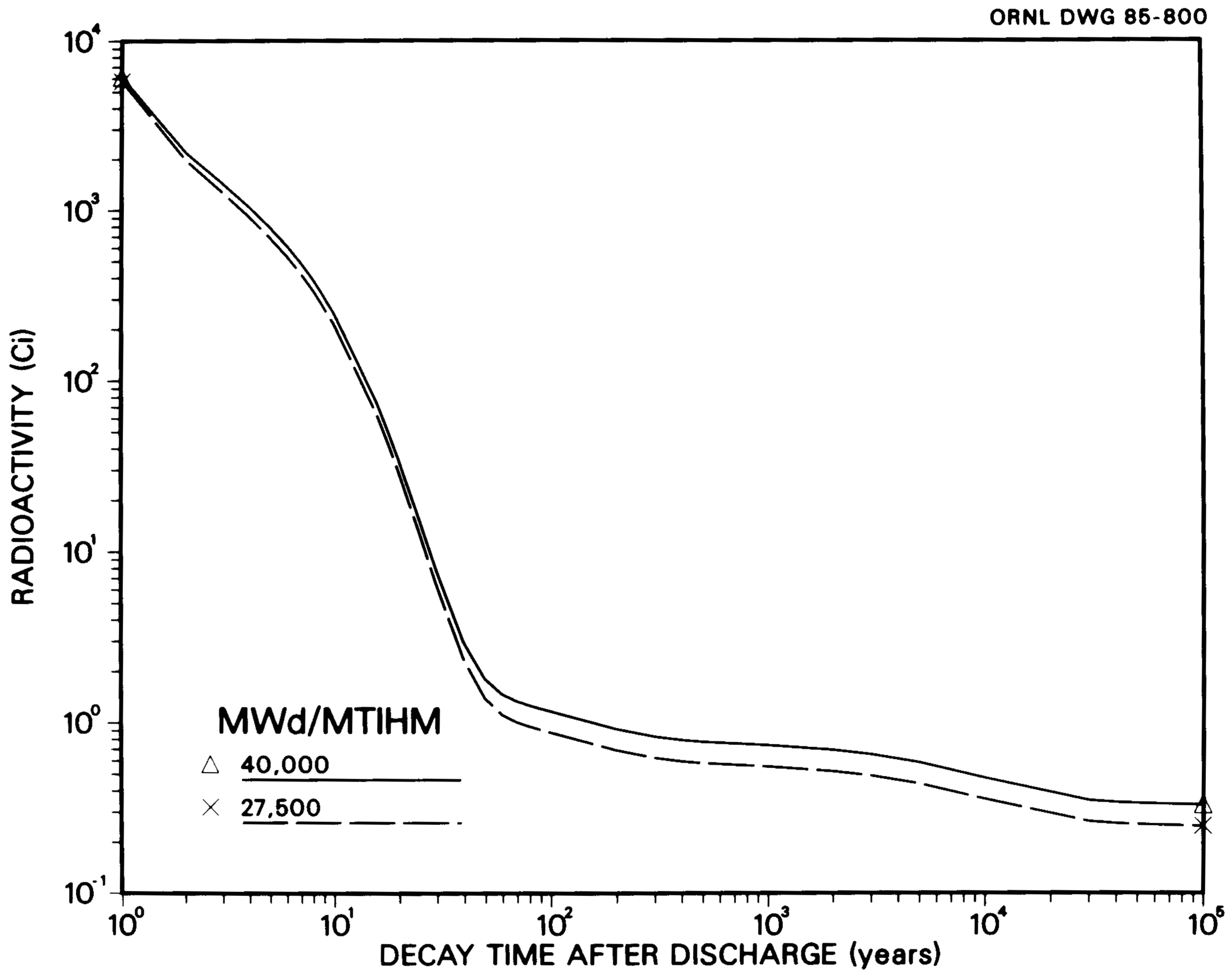


Fig. 4.5. Radioactivity from BWR fuel channels associated with 1 metric ton of initial heavy metal.

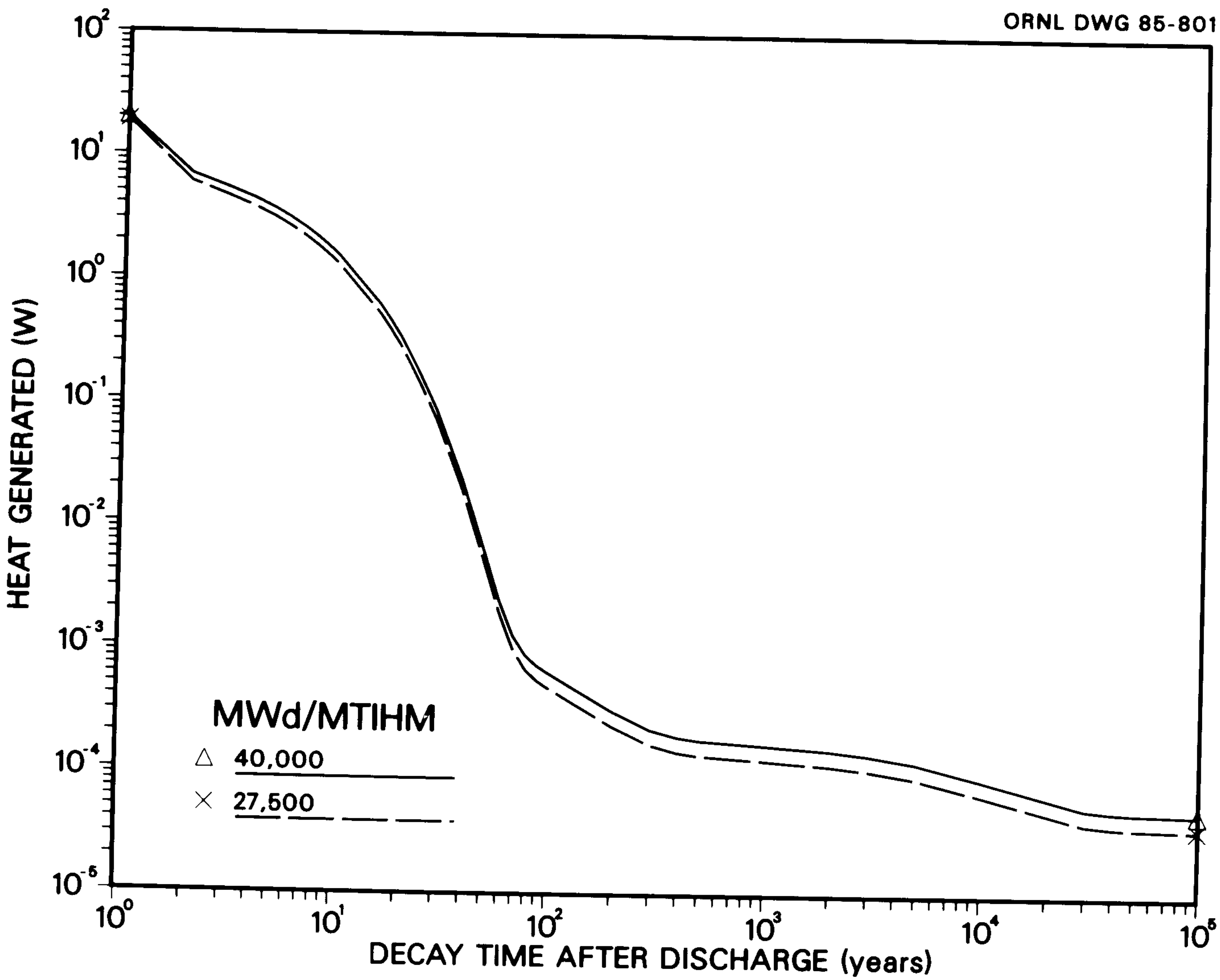


Fig. 4.6. Heat generated from BWR fuel channels associated with 1 metric ton of initial heavy metal.

APPENDIX A. SAMPLE DATA BASE SEARCH

The National Waste Terminal Storage Data Base resides on 24 diskettes which consist of two program disks (labeled Program Disk 1 and Program Disk 2) and 22 data disks (13 for PWRs and 9 for BWRs). The software has been designed to function in conjunction with any IBM PC containing at least 512,000 bytes of memory and either two floppy disks or one floppy- and one hard-disk drive. In the following section, an attempt has been made to illustrate the steps required to perform a typical search and the output that will be obtained. Explanatory comments are given in brackets ([ ]) following most of the statements to help clarify their meaning. Depression of the **enter** key (↵) is usually required after an input command. A back arrow (←) has been placed on certain lines of the text to highlight the searcher's input. Two sets of instructions have been included in this sample search. The user should select the set applicable to his/her IBM PC.

A.1 INSTRUCTIONS FOR A TWO-FLOPPY-DISK-DRIVE SYSTEM

1. Set the internal switches in the PC for either three or four disk drives. This is accomplished by setting switches **7** and **8** on switch box **1** to the OFF and ON positions, respectively, for a three-drive system or both to the OFF position for a four-drive system.

2. Place the NWTSP program disk in drive **A** and activate the computer.

3. The system will ask for the date (e.g., 9-10-85) and the time (e.g., 14:37).

4. After entering the date and time, a pictorial representation of an IBM PC will be displayed with "ORNL Microcomputer Applications" flashing on the screen. Pressing any key will display the installation menu.

5. If you are a first-time user, you may wish to scan the Help File by depressing **F9**. Otherwise, select **F7** which will load the information from the program into the PC memory.

6. Replace program disk No. 1 in drive **A** with program disk No. 2 when prompted to do so, and insert the desired PWR or BWR data disk into drive **B**. Press **enter** and skip to Sect. A.3 for an example of how to perform a search.

A.2 INSTRUCTIONS FOR A HARD-DISK-DRIVE SYSTEM

1. Place the NWTSP program disk No. 1 into drive **A** and activate the computer.

2. The system will ask for the date (e.g., 9-10-85) and the time (e.g., 14:37).

3. After entering the date and time, a pictorial representation of an IBM PC will be displayed with "ORNL Microcomputer Applications" flashing on the screen. Pressing any key will now display the installation menu.

4. If you are a first-time user, you may wish to access the Help File by selecting **F9**. Otherwise, press **F8** to copy the program files onto the hard disk from the two floppy disks provided.

5. The installation program will ask the location of the hard disk; the appropriate selection (C, D, or E) varies with the configuration of the user's system. Type in the drive letter and press **enter**. The program will then ask if all data disks are to be installed on the hard disk. If you wish to copy the 22 data floppy disks onto the hard disk of your system, type **y**; otherwise enter **n**.

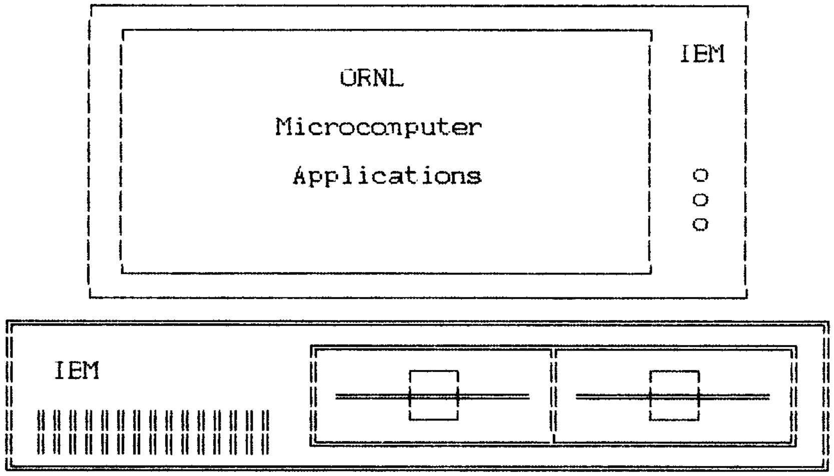
6. The installation routine will now make a subdirectory for the NWTSP system on the user's hard disk and begin copying files to it. When all files needed from program disk No. 1 have been copied, you will be requested to insert program disk No. 2; if data disks are to be copied onto the hard disk, the system will instruct when to insert the PWR disks and the BWR disks, one at a time, into drive **A**.

7. Whenever you wish to run the NWTSP system, make the current directory the root directory, if it is not already your root directory, and type **NWTSP** followed by **enter**.

8. If you ever wish to delete the NWTSP files from the hard disk, select **F10** and the system will be erased (see Help File by pressing **F9**).

A.3 OUTPUT FROM SAMPLE SEARCH OF THE NWTSP DATA BASE

Although this example is brief and portrays only a small portion of the information stored in the data base, it gives the user a basic understanding of the system's operation and capability. This output has been copied verbatim from an actual search.

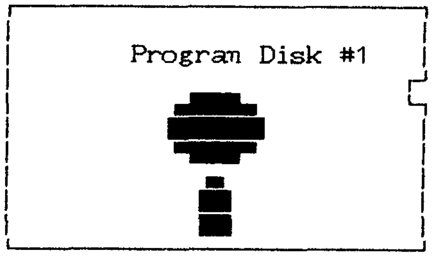


Strike a key when ready . . .

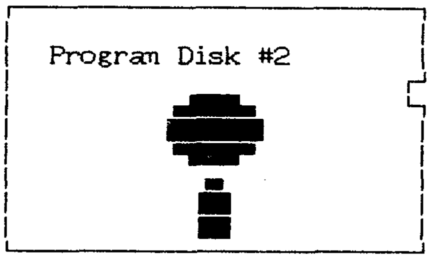
```
Data Ease Installation Menu - Microcomputer Applications

                         F1   F2
                         F3   F4
                         F5   F6
     Start PC System     F7   F8    Install on Hard Disk
                Help     F9   F10   De-install on Hard Disk
```

A>

```
--------------------------------------------------------------
     Setting up NWTSP System . . .
          1. Copying dBase Programs
          2. Copying Picture Files
          3. Copying Totals Data Ease Files
          4. Copying Totals Index Files
--------------------------------------------------------------
```



Remove your PROGRAM DISK #1 from drive A
Strike a key when ready . . .



Insert your PROGRAM DISK #2 in drive A
Strike a key when ready . . .

Insert either a PWR data disk or a EWR data disk in drive B
Strike a key when ready . . .

```
Welcome to the National Waste Terminal Storage
   Program Personal Computer Inquiry System

Developed by: Oak Ridge National Laboratory, Oak Ridge, TN.

This data base was generated by the Integrated Data Base
Program/Repository Waste Characteristics, which is
sponsored by the Office of Civilian Radioactive Waste
Management Office. The information was generated using
the ORIGEN2 computer code and was based on one metric
ton of initial heavy metal including all structural
material in a fuel assembly. A companion report,
"Repository Waste Characteristics I. Commercial LWR
Spent Fuel", ORNL/TM-9591, may be obtained by contacting:
J. W. Roddy, ORNL, P.O. Box X, Oak Ridge, TN. 37831;
telephone: (615)576-8348.

          Press <Spacebar> to Begin . . .
```

```
National Waste Terminal Storage Program Interactive Inquiry System

These are your selections:

A. Type of Reactor:
B. Output Desired:
C. Decay Time:
D. Accuracy Desired:
E. Output will go to:  Screen

Your choices are: A-E (Change indicated item.)
                  S   Sequence thru questions A thru D.
                  R   Run the query you have defined.
                  X   eXit Fuel Repository System.
Enter Your Selection:
```

```
(A.) Select Type of Reactor:
 1. PWR  5,000      7. PWR 33,000     13. PWR 60,000     19. BWR 27,500
 2. PWR 10,000      8. PWR 35,000     14. BWR  5,000     20. BWR 30,000
 3. PWR 15,000      9. PWR 40,000     15. BWR 10,000     21. BWR 35,000
 4. PWR 20,000     10. PWR 45,000     16. BWR 15,000     22. BWR 40,000
 5. PWR 25,000     11. PWR 50,000     17. BWR 20,000
 6. PWR 30,000     12. PWR 55,000     18. BWR 25,000
Which Type of Reactor is desired (1-22):  0
```

[The user must select the appropriate data disk and place it into drive **B** if his system uses two floppy disks.]

```
      National Waste Terminal Storage Program Interactive Inquiry System
 These are your selections:

 A. Type of Reactor:   |
 B. Output Desired:    |
 C. Decay Time:        |
 D. Accuracy Desired:  |
 E. Output will go to: | Screen

 Your choices are: A-E (Change indicated item.)
                   S   Sequence thru questions A thru D.
                   R   Run the query you have defined.
                   X   eXit Fuel Repository System.
 Enter Your Selection:

(A.) Select Type of Reactor:
 1. PWR  5,000       7. PWR 33,000      13. PWR 60,000      19. BWR 27,500
 2. PWR 10,000       8. PWR 35,000      14. BWR  5,000      20. BWR 30,000
 3. PWR 15,000       9. PWR 40,000      15. BWR 10,000      21. BWR 35,000
 4. PWR 20,000      10. PWR 45,000      16. BWR 15,000      22. BWR 40,000
 5. PWR 25,000      11. PWR 50,000      17. BWR 20,000
 6. PWR 30,000      12. PWR 55,000      18. BWR 25,000
Which Type of Reactor is desired (1-22): 7←
```

[The user has selected a PWR of 33,000 MWd/MTIHM.]

```
      National Waste Terminal Storage Program Interactive Inquiry System
 These are your selections:

 A. Type of Reactor:   | PWR 33,000
 B. Output Desired:    |
 C. Decay Time:        |
 D. Accuracy Desired:  |
 E. Output will go to: | Screen

 Your choices are: A-E (Change indicated item.)
                   S   Sequence thru questions A thru D.
                   R   Run the query you have defined.
                   X   eXit Fuel Repository System.
 Enter Your Selection:

(B.) Select Measure of Output Desired:
      1. Grams
      2. Curies
      3. Watts

Which Output is desired (1-3): 0
```

[The user is asked to select either composition (grams), radioactivity (curies), or thermal power (watts) as his output.]

```
   National Waste Terminal Storage Program Interactive Inquiry System
 These are your selections:

 A. Type of Reactor:   | PWR 33,000
 B. Output Desired:    |
 C. Decay Time:        |
 D. Accuracy Desired:  |
 E. Output will go to: | Screen

 Your choices are: A-E (Change indicated item.)
                   S   Sequence thru questions A thru D.
                   R   Run the query you have defined.
                   X   eXit Fuel Repository System.
 Enter Your Selection:
(B.) Select Measure of Output Desired:
      1. Grams
      2. Curies
      3. Watts

Which Output is desired (1-3): 2 <-
```

[The user has selected radioactivity (curies).]

```
   National Waste Terminal Storage Program Interactive Inquiry System
 These are your selections:

 A. Type of Reactor:   | PWR 33,000
 B. Output Desired:    | Curies
 C. Decay Time:        |
 D. Accuracy Desired:  |
 E. Output will go to: | Screen

 Your choices are: A-E (Change indicated item.)
                   S   Sequence thru questions A thru D.
                   R   Run the query you have defined.
                   X   eXit Fuel Repository System.
 Enter Your Selection:
(C.) Select number of years of Decay, which is desired (1-38):  0
  1. 1        8.  8      15. 30     22.  100     29.   3000   36.  200000
  2. 2        9.  9      16. 40     23.  200     30.   5000   37.  500000
  3. 3       10. 10      17. 50     24.  300     31.  10000   38. 1000000
  4. 4       11. 16      18. 60     25.  400     32.  30000
  5. 5       12. 18      19. 70     26.  500     33.  40000
  6. 6       13. 20      20. 80     27. 1000     34.  50000
  7. 7       14. 25      21. 90     28. 2000     35. 100000
```

[The user is asked to select the number of years the fuel has been discharged from the reactor.]

```
     National Waste Terminal Storage Program Interactive Inquiry System
  These are your selections:

  A. Type of Reactor:   | PWR 33,000
  B. Output Desired:    | Curies
  C. Decay Time:        |
  D. Accuracy Desired:  |
  E. Output will go to: | Screen

  Your choices are: A-E (Change indicated item.)
                    S   Sequence thru questions A thru D.
                    R   Run the query you have defined.
                    X   eXit Fuel Repository System.
  Enter Your Selection:
(C.) Select number of years of Decay, which is desired (1-38): 27 <-
  1. 1       8.   8     15. 30     22.  100     29.   3000   36.  200000
  2. 2       9.   9     16. 40     23.  200     30.   5000   37.  500000
  3. 3      10.  10     17. 50     24.  300     31.  10000   38. 1000000
  4. 4      11.  16     18. 60     25.  400     32.  30000
  5. 5      12.  18     19. 70     26.  500     33.  40000
  6. 6      13.  20     20. 80     27. 1000     34.  50000
  7. 7      14.  25     21. 90     28. 2000     35. 100000
```

[The user selects the 1000 year option.]

```
     National Waste Terminal Storage Program Interactive Inquiry System
  These are your selections:

  A. Type of Reactor:   | PWR 33,000
  B. Output Desired:    | Curies
  C. Decay Time:        | 1000 years
  D. Accuracy Desired:  |
  E. Output will go to: | Screen

  Your choices are: A-E (Change indicated item.)
                    S   Sequence thru questions A thru D.
                    R   Run the query you have defined.
                    X   eXit Fuel Repository System.
  Enter Your Selection:
(D.) Select Accuracy Level:
     1. > 1.000 % of Total
     2. > 0.100 % of Total
     3. > 0.010 % of Total
     4. > 0.001 % of Total
     5. All isotopes

Which Level of Accuracy is desired (1-5):  0
```

[The user has some choice as to the number of nuclides displayed by his selection of the accuracy level desired. Low accuracies (1 or 2) are recommended for the initial search. The selection of Option 5 for composite (grams) and short decay times will generate lengthy tables.]

```
     National Waste Terminal Storage Program Interactive Inquiry System
|| These are your selections:                                            ||
||                        ______________________________________________ ||
|| A. Type of Reactor:   | PWR 33,000                                  | ||
|| B. Output Desired:    | Curies                                      | ||
|| C. Decay Time:        | 1000 years                                  | ||
|| D. Accuracy Desired:  |                                             | ||
|| E. Output will go to: | Screen                                      | ||
||                        ---------------------------------------------- ||
|| Your choices are: A-E (Change indicated item.)                        ||
||                   S   Sequence thru questions A thru D.               ||
||                   R   Run the query you have defined.                 ||
||                   X   eXit Fuel Repository System.                    ||
|| Enter Your Selection:                                                 ||
 (D.) Select Accuracy Level:
      1. > 1.000 % of Total
      2. > 0.100 % of Total
      3. > 0.010 % of Total
      4. > 0.001 % of Total
      5. All isotopes

 Which Level of Accuracy is desired (1-5):  2<-
```

[The user has asked for a tabulation of all isotopes which contribute >0.1% to the total radioactivity.]

```
     National Waste Terminal Storage Program Interactive Inquiry System
|| These are your selections:                                            ||
||                        ______________________________________________ ||
|| A. Type of Reactor:   | PWR 33,000                                  | ||
|| B. Output Desired:    | Curies                                      | ||
|| C. Decay Time:        | 1000 years                                  | ||
|| D. Accuracy Desired:  | > 0.100 % of Total                          | ||
|| E. Output will go to: | Screen                                      | ||
||                        ---------------------------------------------- ||
|| Your choices are: A-E (Change indicated item.)                        ||
||                   S   Sequence thru questions A thru D.               ||
||                   R   Run the query you have defined.                 ||
||                   X   eXit Fuel Repository System.                    ||
|| Enter Your Selection: R<-                                             ||
```

[R designates the system will begin the tabulation upon the user pressing the **enter** key. The data will appear on the screen, but the information may be diverted to a printer by depressing **5** and **enter.** Of course, the user's system must be equipped with the necessary hardware.]

Spent Fuel Repository Characteristics Data Base
Developed by: Oak Ridge National Laboratory, Oak Ridge, TN.

Type of Reactor: PWR 33,000
Elapsed Decay (years): 1000
All isotopes representing: > 0.100 % of Total

Isotope	Curies	Percentage of Total
NI 59	5.11E+00	0.293 %
ZR 93	1.93E+00	0.110 %
NB 93M	1.83E+00	0.105 %
TC 99	1.30E+01	0.747 %
U234	2.03E+00	0.116 %
NP239	1.56E+01	0.896 %
PU239	3.05E+02	17.528 %
PU240	4.78E+02	27.471 %
AM241	8.93E+02	51.321 %
AM243	1.56E+01	0.896 %
Subtotal Curies =	1.73E+03	99.488 %
Total all isotopes =	1.74E+03	

Press any key to continue...

[The table lists the nuclides, their contribution to the radioactivity and to the total. The subtotal applies to only those isotopes given in the table.]

National Waste Terminal Storage Program Interactive Inquiry System

```
These are your selections:

A. Type of Reactor:    | PWR 33,000
B. Output Desired:     | Curies
C. Decay Time:         | 1000 years
D. Accuracy Desired:   | > 0.100 % of Total
E. Output will go to:  | Screen

Your choices are: A-E (Change indicated item.)
                  S   Sequence thru questions A thru D.
                  R   Run the query you have defined.
                  X   eXit Fuel Repository System.
Enter Your Selection: R
```

[User may select any options listed as choices (A-E, S, R, or X).]

```
     National Waste Terminal Storage Program Interactive Inquiry System
  These are your selections:

  A. Type of Reactor:   | PWR 33,000
  B. Output Desired:    | Curies
  C. Decay Time:        | 1000 years
  D. Accuracy Desired:  | > 0.100 % of Total
  E. Output will go to: | Screen

  Your choices are: A-E (Change indicated item.)
                    S   Sequence thru questions A thru D.
                    R   Run the query you have defined.
                    X   eXit Fuel Repository System.
  Enter Your Selection: D<-
```

[The user wishes to change the accuracy of the search.]

```
     National Waste Terminal Storage Program Interactive Inquiry System
  These are your selections:

  A. Type of Reactor:   | PWR 33,000
  B. Output Desired:    | Curies
  C. Decay Time:        | 1000 years
  D. Accuracy Desired:  | > 0.100 % of Total
  E. Output will go to: | Screen

  Your choices are: A-E (Change indicated item.)
                    S   Sequence thru questions A thru D.
                    R   Run the query you have defined.
                    X   eXit Fuel Repository System.
  Enter Your Selection: D
(D.) Select Accuracy Level:
      1. > 1.000 % of Total
      2. > 0.100 % of Total
      3. > 0.010 % of Total
      4. > 0.001 % of Total
      5. All isotopes

Which Level of Accuracy is desired (1-5):  0
```

[User is asked to select accuracy desired.]

```
     National Waste Terminal Storage Program Interactive Inquiry System
 These are your selections:

 A. Type of Reactor:     | PWR 33,000
 B. Output Desired:      | Curies
 C. Decay Time:          | 1000 years
 D. Accuracy Desired:    | > 0.100 % of Total
 E. Output will go to:   | Screen

 Your choices are: A-E (Change indicated item.)
                   S   Sequence thru questions A thru D.
                   R   Run the query you have defined.
                   X   eXit Fuel Repository System.
 Enter Your Selection: D
 (D.) Select Accuracy Level:
      1. > 1.000 % of Total
      2. > 0.100 % of Total
      3. > 0.010 % of Total
      4. > 0.001 % of Total
      5. All isotopes

 Which Level of Accuracy is desired (1-5):  3<-
```

[The user wishes to increase the accuracy by 10-fold by selecting 0.01%.]

```
     National Waste Terminal Storage Program Interactive Inquiry System
 These are your selections:

 A. Type of Reactor:     | PWR 33,000
 B. Output Desired:      | Curies
 C. Decay Time:          | 1000 years
 D. Accuracy Desired:    | > 0.010 % of Total
 E. Output will go to:   | Screen

 Your choices are: A-E (Change indicated item.)
                   S   Sequence thru questions A thru D.
                   R   Run the query you have defined.
                   X   eXit Fuel Repository System.
 Enter Your Selection: R
```

[The user wishes the tabulation.]

Spent Fuel Repository Characteristics Data Base
Developed by: Oak Ridge National Laboratory, Oak Ridge, TN.

Type of Reactor: PWR 33,000
Elapsed Decay (years): 1000
All isotopes representing: > 0.010 % of Total

Isotope	Curies	Percentage of Total
C 14	1.37E+00	0.078 %
NI 59	5.11E+00	0.293 %
NI 63	3.76E-01	0.021 %
SE 79	4.04E-01	0.023 %
ZR 93	1.93E+00	0.110 %
NB 93M	1.83E+00	0.105 %
NB 94	1.24E+00	0.071 %
TC 99	1.30E+01	0.747 %
SN126	7.71E-01	0.044 %
SB126M	7.71E-01	0.044 %
CS135	3.45E-01	0.019 %

(Press Q to Quit)

Press any key to continue...

[Tabulation is not yet complete. User may depress Q to terminate output, but the user in this example wishes to continue the listing.]

Spent Fuel Repository Characteristics Data Base
Developed by: Oak Ridge National Laboratory, Oak Ridge, TN.

Type of Reactor: PWR 33,000
Elapsed Decay (years): 1000
All isotopes representing: > 0.010 % of Total

Isotope	Curies	Percentage of Total
TH234	3.18E-01	0.018 %
PA233	9.99E-01	0.057 %
PA234M	3.18E-01	0.018 %
U234	2.03E+00	0.116 %
U236	2.71E-01	0.015 %
U238	3.18E-01	0.018 %
NP237	9.99E-01	0.057 %
NP239	1.56E+01	0.896 %
PU238	1.08E+00	0.062 %
PU239	3.05E+02	17.528 %
PU240	4.78E+02	27.471 %

(Press Q to Quit)

Press any key to continue...

[Still not complete.]

```
            Spent Fuel Repository Characteristics Data Base
   Developed by: Oak Ridge National Laboratory, Oak Ridge, TN.

            Type of Reactor: PWR 33,000
            Elapsed Decay (years):    1000
            All isotopes representing: > 0.010 % of Total

                                                       Percentage
           Isotope                Curies                of Total
=================================================================
            PU242                1.72E+00                0.098 %
            AM241                8.93E+02               51.321 %
            AM243                1.56E+01                0.896 %
=================================================================
        Subtotal Curies    =     1.74E+03              100.137 %
     Total all isotopes =        1.74E+03
Press any key to continue...
```

[All nuclides meeting criteria have been listed.]

```
     National Waste Terminal Storage Program Interactive Inquiry System
 These are your selections:

 A. Type of Reactor:   | PWR 33,000
 B. Output Desired:    | Curies
 C. Decay Time:        | 1000 years
 D. Accuracy Desired:  | > 0.010 % of Total
 E. Output will go to: | Screen

 Your choices are: A-E (Change indicated item.)
                   S   Sequence thru questions A thru D.
                   R   Run the query you have defined.
                   X   eXit Fuel Repository System.
 Enter Your Selection: R
```

[User may select any options listed as choices (A-E, S, R, or X).]

```
     National Waste Terminal Storage Program Interactive Inquiry System
 These are your selections:

 A. Type of Reactor:   | PWR 33,000
 B. Output Desired:    | Curies
 C. Decay Time:        | 1000 years
 D. Accuracy Desired:  | > 0.010 % of Total
 E. Output will go to: | Screen

 Your choices are: A-E (Change indicated item.)
                   S   Sequence thru questions A thru D.
                   R   Run the query you have defined.
                   X   eXit Fuel Repository System.
 Enter Your Selection: X<--
```

[User wishes to terminate the search by typing **X**.]

```
   National Waste Terminal Storage Program Interactive Inquiry System
  These are your selections:

  A. Type of Reactor:   PWR 33,000
  B. Output Desired:    Curies
  C. Decay Time:        1000 years
  D. Accuracy Desired:  > 0.010 % of Total
  E. Output will go to: Screen

  Your choices are: A-E (Change indicated item.)
                    S   Sequence thru questions A thru D.
                    R   Run the query you have defined.
                    X   eXit Fuel Repository System.
  Enter Your Selection: X<-
```

EXIT SYSTEM SELECTED !

Do you wish to end this session (Y/N)? N<-

[User decides to continue search and types **N.**]

```
   National Waste Terminal Storage Program Interactive Inquiry System
  These are your selections:

  A. Type of Reactor:   PWR 33,000
  B. Output Desired:    Curies
  C. Decay Time:        1000 years
  D. Accuracy Desired:  > 0.010 % of Total
  E. Output will go to: Screen

  Your choices are: A-E (Change indicated item.)
                    S   Sequence thru questions A thru D.
                    R   Run the query you have defined.
                    X   eXit Fuel Repository System.
  Enter Your Selection: X
```

EXIT SYSTEM SELECTED !

Do you wish to end this session (Y/N)? Y<-

[User gets a second chance and reverses his decision by typing **Y** and pressing **enter.**]

APPENDIX B. DIAGRAMS OF RADIOACTIVE DECAY CHAINS

This appendix contains diagrams[1] of the decay chains that involve two or more of the radionuclides that may be of importance in their long-term disposal in a geologic repository. The half-life, modes of decay, and decay branching ratios for each radionuclide in the decay chain are shown. The branching ratios for spontaneous fission and those with ratios <0.1% are omitted. Symbols are defined in Sect. B.1.

$$^{28}\mathrm{Mg}\ (20.91\ \mathrm{h}) \xrightarrow{\beta^-} {}^{28}\mathrm{Al}\ (2.240\ \mathrm{m}) \xrightarrow{\beta^-} {}^{28}\mathrm{Si}$$

$$^{32}\mathrm{Si}\ (3.3\mathrm{E}2\ \mathrm{y}) \xrightarrow{\beta^-} {}^{32}\mathrm{P}\ (14.29\ \mathrm{d}) \xrightarrow{\beta^-} {}^{32}\mathrm{S}$$

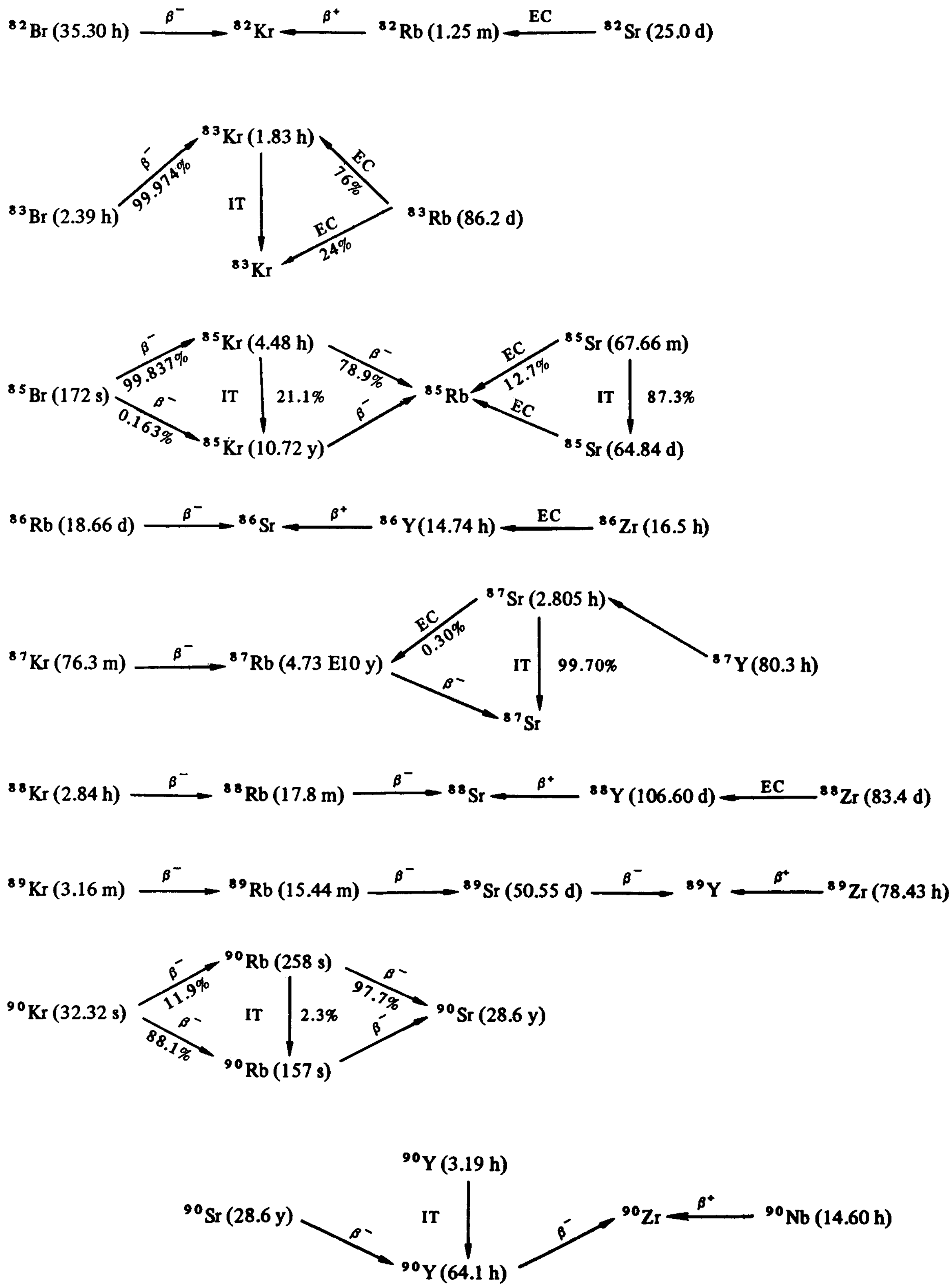
$$^{44}\mathrm{Ca} \xleftarrow{\beta^+} {}^{44}\mathrm{Sc}\ (3.927\ \mathrm{h}) \xleftarrow{\mathrm{EC}} {}^{44}\mathrm{Ti}\ (47.3\ \mathrm{y})$$

$$\begin{tikzcd}
{}^{46}\mathrm{Sc}\ (18.67\ \mathrm{s}) \arrow[d, "\mathrm{IT}"'] & {}^{46}\mathrm{Ti} \\
{}^{46}\mathrm{Sc}\ (83.83\ \mathrm{d}) \arrow[ur, "\beta^-"] &
\end{tikzcd}$$

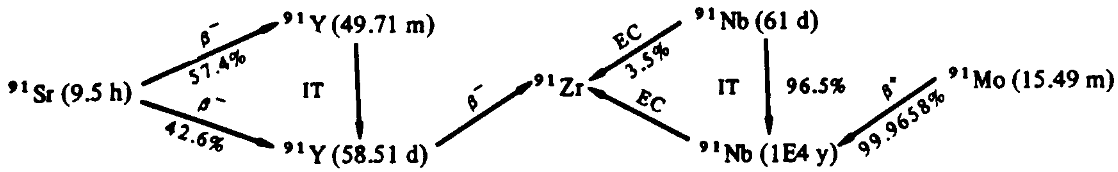
$$^{47}\mathrm{Ca}\ (4.536\ \mathrm{d}) \xrightarrow{\beta^-} {}^{47}\mathrm{Sc}\ (3.422\ \mathrm{d}) \xrightarrow{\beta^-} {}^{47}\mathrm{Ti}$$

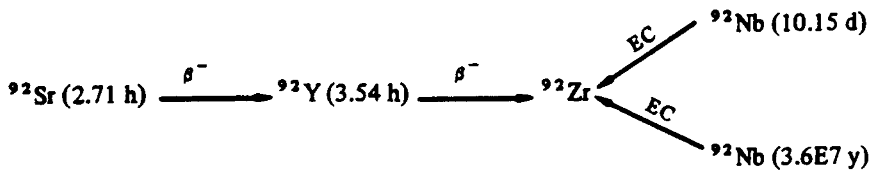
$$^{49}\mathrm{Ca}\ (8.719\ \mathrm{m}) \xrightarrow{\beta^-} {}^{49}\mathrm{Sc}\ (57.4\ \mathrm{m}) \xrightarrow{\beta^-} {}^{49}\mathrm{Ti} \xleftarrow{\mathrm{EC}} {}^{49}\mathrm{V}\ (330\ \mathrm{d}) \xleftarrow{\beta^+} {}^{49}\mathrm{Cr}\ (42.09\ \mathrm{m})$$

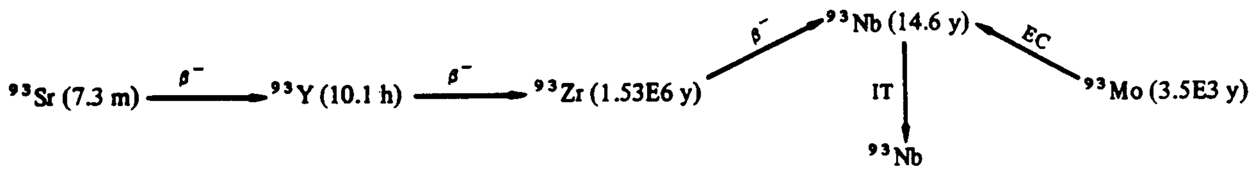
$$\begin{tikzcd}
 & & {}^{52}\mathrm{Mn}\ (21.4\ \mathrm{m}) \arrow[dl, "\beta^+\ 98.25\%"'] \arrow[dd, "\mathrm{IT}\ 1.75\%"] & \\
{}^{52}\mathrm{V}\ (3.75\ \mathrm{m}) \arrow[r, "\beta^-"] & {}^{52}\mathrm{Cr} & & {}^{52}\mathrm{Fe}\ (8.275\ \mathrm{h}) \arrow[ul, "\beta^+"'] \\
 & & {}^{52}\mathrm{Mn}\ (5.591\ \mathrm{d}) \arrow[ul, "\beta^+"] &
\end{tikzcd}$$

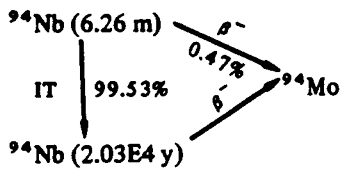


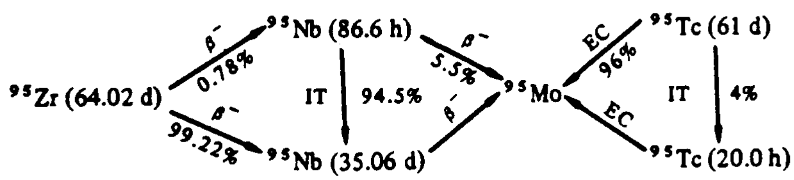
⁸²Br (35.30 h)
β⁻
⁸²Kr
β⁺
⁸²Rb (1.25 m)
EC
⁸²Sr (25.0 d)
⁸³Kr (1.83 h)
β⁻
99.974%
EC
76%
⁸³Br (2.39 h)
IT
⁸³Rb (86.2 d)
EC
24%
⁸³Kr
⁸⁵Kr (4.48 h)
β⁻
99.837%
β⁻
78.9%
EC
12.7%
⁸⁵Sr (67.66 m)
⁸⁵Br (172 s)
β⁻
IT
21.1%
⁸⁵Rb
IT
87.3%
0.163%
β⁻
EC
⁸⁵Kr (10.72 y)
⁸⁵Sr (64.84 d)
⁸⁶Rb (18.66 d)
β⁻
⁸⁶Sr
β⁺
⁸⁶Y (14.74 h)
EC
⁸⁶Zr (16.5 h)
⁸⁷Sr (2.805 h)
EC
0.30%
⁸⁷Kr (76.3 m)
β⁻
⁸⁷Rb (4.73 E10 y)
IT
99.70%
⁸⁷Y (80.3 h)
β⁻
⁸⁷Sr
⁸⁸Kr (2.84 h)
β⁻
⁸⁸Rb (17.8 m)
β⁻
⁸⁸Sr
β⁺
⁸⁸Y (106.60 d)
EC
⁸⁸Zr (83.4 d)
⁸⁹Kr (3.16 m)
β⁻
⁸⁹Rb (15.44 m)
β⁻
⁸⁹Sr (50.55 d)
β⁻
⁸⁹Y
β⁺
⁸⁹Zr (78.43 h)
⁹⁰Rb (258 s)
β⁻
11.9%
β⁻
97.7%
⁹⁰Kr (32.32 s)
β⁻
IT
2.3%
β⁻
⁹⁰Sr (28.6 y)
88.1%
⁹⁰Rb (157 s)
⁹⁰Y (3.19 h)
⁹⁰Sr (28.6 y)
IT
β⁻
⁹⁰Zr
β⁺
⁹⁰Nb (14.60 h)
β⁻
⁹⁰Y (64.1 h)

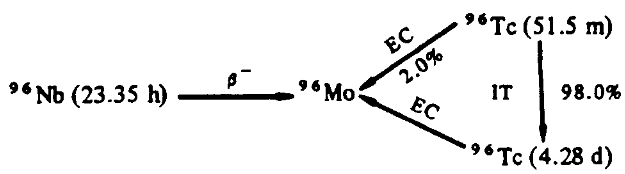

^{91}Sr (9.5 h)
β^- 57.4%
β^- 42.6%
^{91}Y (49.71 m)
IT
^{91}Y (58.51 d)
β^-
^{91}Zr
EC 3.5%
EC
^{91}Nb (61 d)
IT
96.5%
^{91}Nb (1E4 y)
β^+ 99.9658%
^{91}Mo (15.49 m)

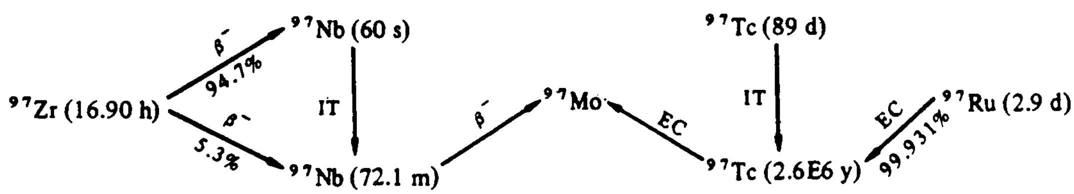

^{92}Sr (2.71 h)
β^-
^{92}Y (3.54 h)
β^-
^{92}Zr
EC
^{92}Nb (10.15 d)
EC
^{92}Nb (3.6E7 y)

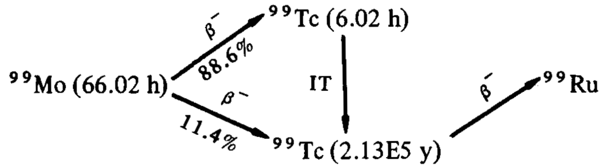

^{93}Sr (7.3 m)
β^-
^{93}Y (10.1 h)
β^-
^{93}Zr (1.53E6 y)
β^-
^{93}Nb (14.6 y)
EC
^{93}Mo (3.5E3 y)
IT
^{93}Nb


^{94}Nb (6.26 m)
β^- 0.47%
^{94}Mo
IT
99.53%
β^-
^{94}Nb (2.03E4 y)

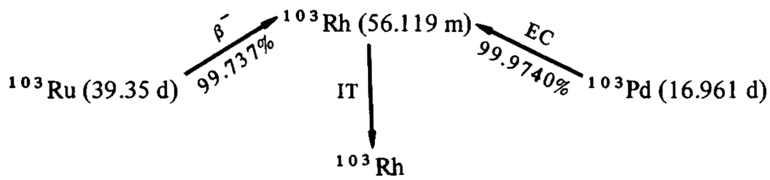

^{95}Zr (64.02 d)
β^- 0.78%
β^- 99.22%
^{95}Nb (86.6 h)
IT
94.5%
^{95}Nb (35.06 d)
β^- 5.5%
β^-
^{95}Mo
EC 96%
EC
^{95}Tc (61 d)
IT
4%
^{95}Tc (20.0 h)

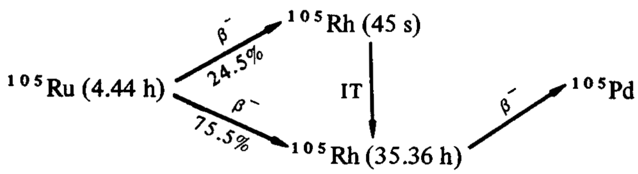

^{96}Nb (23.35 h)
β^-
^{96}Mo
EC 2.0%
EC
^{96}Tc (51.5 m)
IT
98.0%
^{96}Tc (4.28 d)


^{97}Zr (16.90 h)
β^- 94.7%
β^- 5.3%
^{97}Nb (60 s)
IT
^{97}Nb (72.1 m)
β^-
^{97}Mo
EC
^{97}Tc (89 d)
IT
^{97}Tc (2.6E6 y)
EC 99.931%
^{97}Ru (2.9 d)

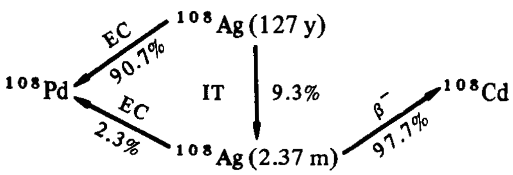


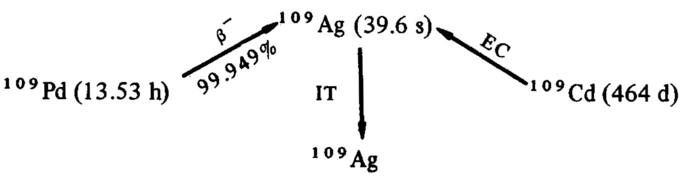
$$^{101}\mathrm{Mo}\ (14.61\ \mathrm{m}) \xrightarrow{\beta^-} {}^{101}\mathrm{Tc}\ (14.2\ \mathrm{m}) \xrightarrow{\beta^-} {}^{101}\mathrm{Ru}$$

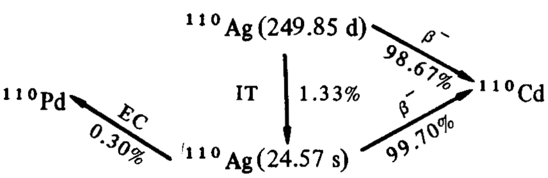


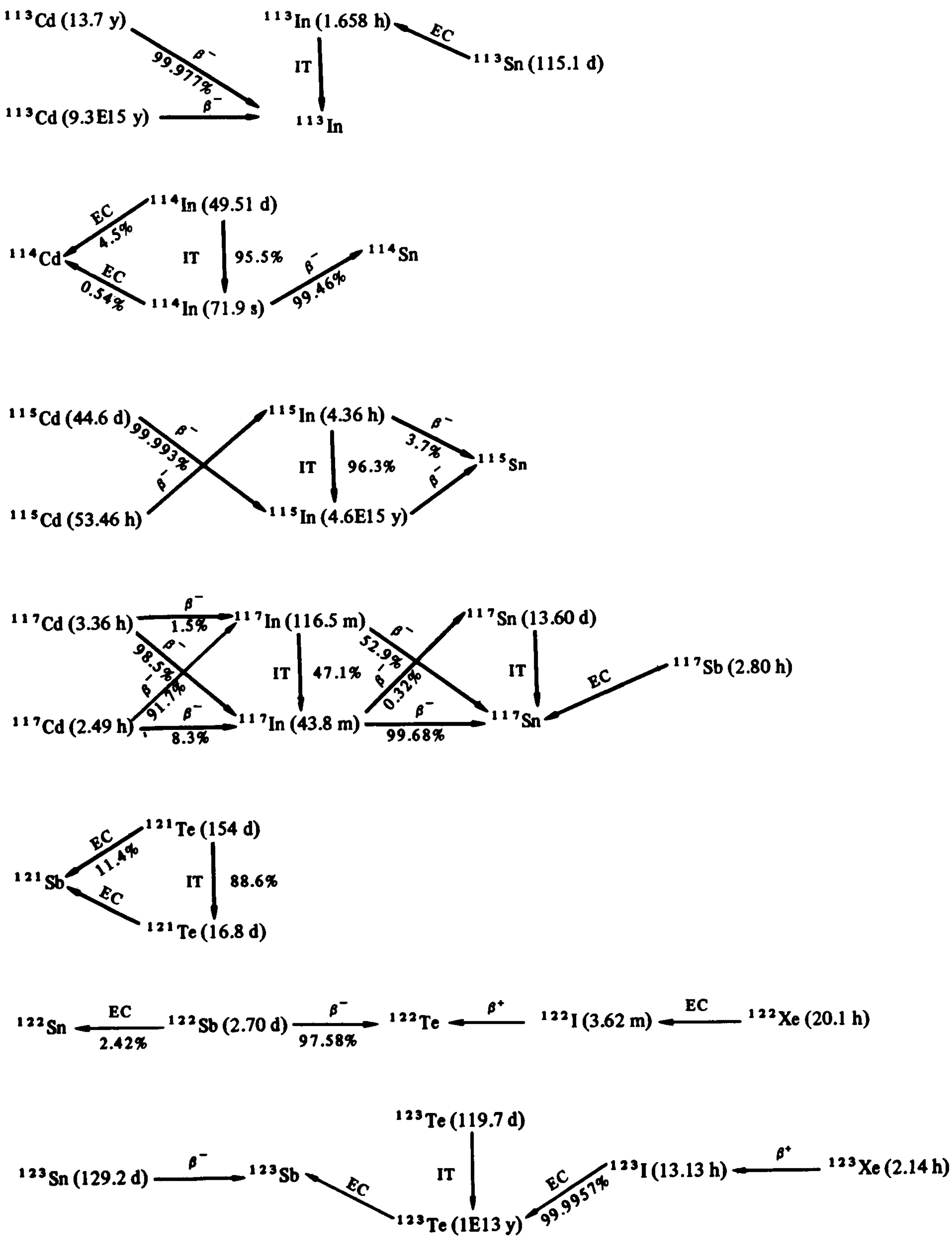


$$^{106}\mathrm{Ru}\ (368.2\ \mathrm{d}) \xrightarrow{\beta^-} {}^{106}\mathrm{Rh}\ (29.92\ \mathrm{s}) \xrightarrow{\beta^-} {}^{106}\mathrm{Pd} \xleftarrow{\mathrm{EC}} {}^{106}\mathrm{Ag}\ (8.46\ \mathrm{d})$$

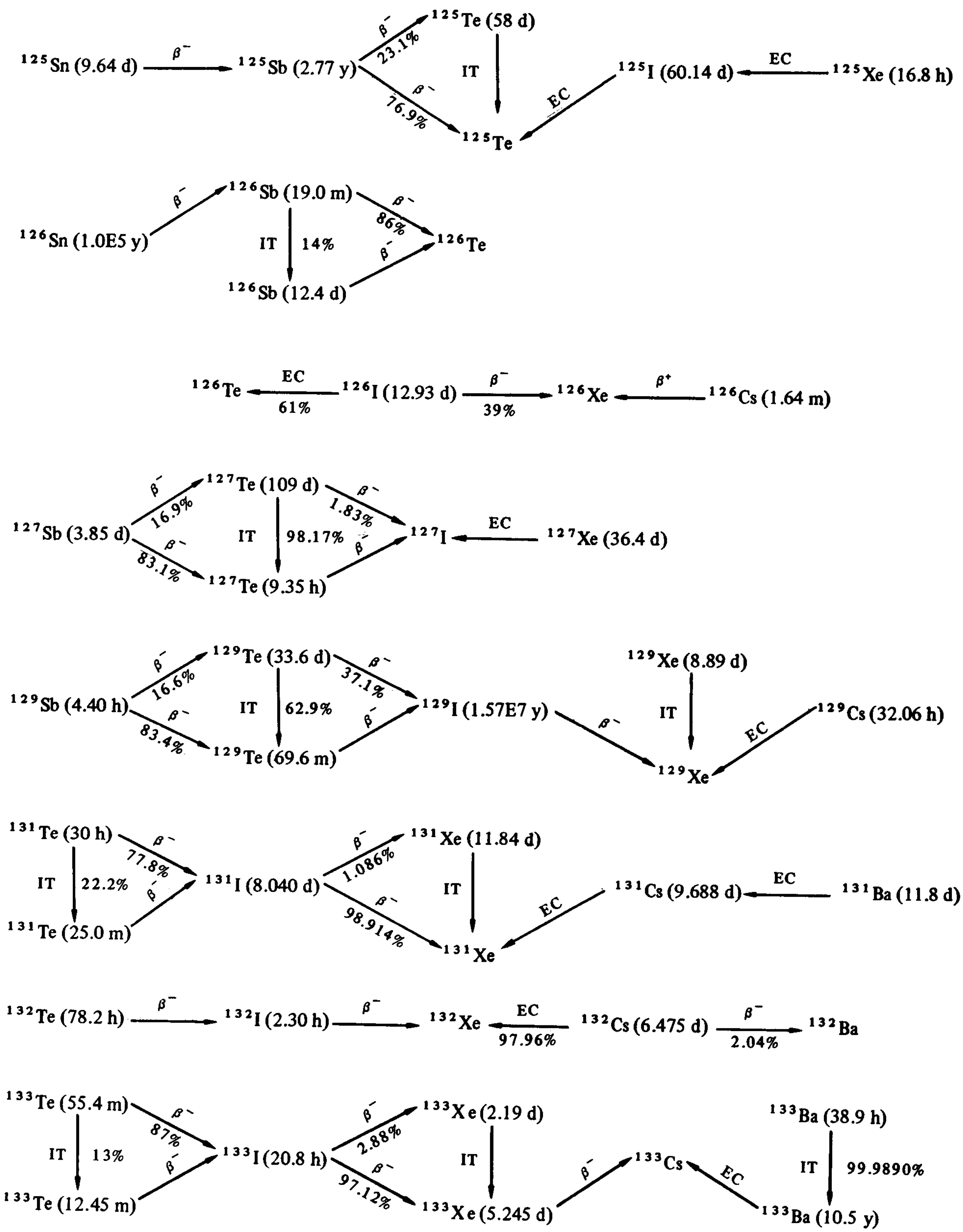








¹¹³Cd (13.7 y)
β⁻ 99.977%
¹¹³Cd (9.3E15 y)
β⁻
¹¹³In (1.658 h)
IT
¹¹³In
EC
¹¹³Sn (115.1 d)
¹¹⁴In (49.51 d)
EC 4.5%
¹¹⁴Cd
EC 0.54%
IT 95.5%
¹¹⁴In (71.9 s)
β⁻ 99.46%
¹¹⁴Sn
¹¹⁵Cd (44.6 d)
β⁻ 99.993%
¹¹⁵Cd (53.46 h)
β⁻
¹¹⁵In (4.36 h)
IT 96.3%
¹¹⁵In (4.6E15 y)
β⁻ 3.7%
β⁻
¹¹⁵Sn
¹¹⁷Cd (3.36 h)
β⁻ 1.5%
β⁻ 98.5%
¹¹⁷Cd (2.49 h)
β⁻ 91.7%
β⁻ 8.3%
¹¹⁷In (116.5 m)
IT 47.1%
¹¹⁷In (43.8 m)
β⁻ 52.9%
β⁻ 0.32%
β⁻ 99.68%
¹¹⁷Sn (13.60 d)
IT
¹¹⁷Sn
EC
¹¹⁷Sb (2.80 h)
¹²¹Te (154 d)
EC 11.4%
¹²¹Sb
EC
IT 88.6%
¹²¹Te (16.8 d)
¹²²Sn
EC 2.42%
¹²²Sb (2.70 d)
β⁻ 97.58%
¹²²Te
β⁺
¹²²I (3.62 m)
EC
¹²²Xe (20.1 h)
¹²³Te (119.7 d)
¹²³Sn (129.2 d)
β⁻
¹²³Sb
EC
IT
¹²³Te (1E13 y)
EC 99.9957%
¹²³I (13.13 h)
β⁺
¹²³Xe (2.14 h)



¹²⁵Sn (9.64 d)
β⁻
¹²⁵Sb (2.77 y)
β⁻ 23.1%
¹²⁵Te (58 d)
IT
β⁻ 76.9%
¹²⁵Te
EC
¹²⁵I (60.14 d)
EC
¹²⁵Xe (16.8 h)
¹²⁶Sn (1.0E5 y)
β⁻
¹²⁶Sb (19.0 m)
IT 14%
β⁻ 86%
¹²⁶Te
β⁻
¹²⁶Sb (12.4 d)
¹²⁶Te
EC 61%
¹²⁶I (12.93 d)
β⁻ 39%
¹²⁶Xe
β⁺
¹²⁶Cs (1.64 m)
¹²⁷Sb (3.85 d)
β⁻ 16.9%
¹²⁷Te (109 d)
IT 98.17%
β⁻ 1.83%
β⁻ 83.1%
¹²⁷Te (9.35 h)
β⁻
¹²⁷I
EC
¹²⁷Xe (36.4 d)
¹²⁹Sb (4.40 h)
β⁻ 16.6%
¹²⁹Te (33.6 d)
IT 62.9%
β⁻ 37.1%
β⁻ 83.4%
¹²⁹Te (69.6 m)
β⁻
¹²⁹I (1.57E7 y)
β⁻
¹²⁹Xe (8.89 d)
IT
¹²⁹Xe
EC
¹²⁹Cs (32.06 h)
¹³¹Te (30 h)
IT 22.2%
β⁻ 77.8%
¹³¹Te (25.0 m)
β⁻
¹³¹I (8.040 d)
β⁻ 1.086%
¹³¹Xe (11.84 d)
IT
β⁻ 98.914%
¹³¹Xe
EC
¹³¹Cs (9.688 d)
EC
¹³¹Ba (11.8 d)
¹³²Te (78.2 h)
β⁻
¹³²I (2.30 h)
β⁻
¹³²Xe
EC 97.96%
¹³²Cs (6.475 d)
β⁻ 2.04%
¹³²Ba
¹³³Te (55.4 m)
IT 13%
β⁻ 87%
¹³³Te (12.45 m)
β⁻
¹³³I (20.8 h)
β⁻ 2.88%
¹³³Xe (2.19 d)
IT
β⁻ 97.12%
¹³³Xe (5.245 d)
β⁻
¹³³Cs
EC
¹³³Ba (38.9 h)
IT 99.9890%
¹³³Ba (10.5 y)

$$^{134}\mathrm{Te}\ (41.8\ \mathrm{m}) \xrightarrow{\beta^-} {}^{134}\mathrm{I}\ (52.6\ \mathrm{m}) \xrightarrow{\beta^-} {}^{134}\mathrm{Xe}$$

$$^{134}\mathrm{Cs}\ (2.90\ \mathrm{h}) \xrightarrow{\mathrm{IT}} {}^{134}\mathrm{Cs}\ (2.062\ \mathrm{y}) \xrightarrow{\beta^-} {}^{134}\mathrm{Ba}$$

$$^{135}\mathrm{I}\ (6.61\ \mathrm{h}) \xrightarrow[16.5\%]{\beta^-} {}^{135}\mathrm{Xe}\ (15.36\ \mathrm{m})$$

$$^{135}\mathrm{I}\ (6.61\ \mathrm{h}) \xrightarrow[83.5\%]{\beta^-} {}^{135}\mathrm{Xe}\ (9.11\ \mathrm{h})$$

$$^{135}\mathrm{Xe}\ (15.36\ \mathrm{m}) \xrightarrow[99.9964\%]{\mathrm{IT}} {}^{135}\mathrm{Xe}\ (9.11\ \mathrm{h}) \xrightarrow{\beta^-} {}^{135}\mathrm{Cs}\ (2.3\mathrm{E}6\ \mathrm{y}) \xrightarrow{\beta^-} {}^{135}\mathrm{Ba}$$

$$^{135}\mathrm{Ba}\ (28.7\ \mathrm{h}) \xrightarrow{\mathrm{IT}} {}^{135}\mathrm{Ba}$$

$$^{137}\mathrm{Xe}\ (3.83\ \mathrm{m}) \xrightarrow{\beta^-} {}^{137}\mathrm{Cs}\ (30.17\ \mathrm{y}) \xrightarrow[94.6\%]{\beta^-} {}^{137}\mathrm{Ba}\ (2.522\ \mathrm{m}) \xrightarrow{\mathrm{IT}} {}^{137}\mathrm{Ba}$$

$$^{137}\mathrm{Cs}\ (30.17\ \mathrm{y}) \xrightarrow[5.4\%]{\beta^-} {}^{137}\mathrm{Ba}$$

$$^{138}\mathrm{Xe}\ (14.13\ \mathrm{m}) \xrightarrow{\beta^-} {}^{138}\mathrm{Cs}\ (32.2\ \mathrm{m}) \xrightarrow{\beta^-} {}^{138}\mathrm{Ba}$$

$$^{139}\mathrm{Cs}\ (9.40\ \mathrm{m}) \xrightarrow{\beta^-} {}^{139}\mathrm{Ba}\ (83.1\ \mathrm{m}) \xrightarrow{\beta^-} {}^{139}\mathrm{La} \xleftarrow{\mathrm{EC}} {}^{139}\mathrm{Ce}\ (137.66\ \mathrm{d})$$

$$^{140}\mathrm{Ba}\ (12.789\ \mathrm{d}) \xrightarrow{\beta^-} {}^{140}\mathrm{La}\ (40.22\ \mathrm{h}) \xrightarrow{\beta^-} {}^{140}\mathrm{Ce}$$

$$^{141}\mathrm{Ba}\ (18.27\ \mathrm{m}) \xrightarrow{\beta^-} {}^{141}\mathrm{La}\ (3.94\ \mathrm{h}) \xrightarrow{\beta^-} {}^{141}\mathrm{Ce}\ (32.50\ \mathrm{d}) \xrightarrow{\beta^-} {}^{141}\mathrm{Pr}$$

$$^{142}\mathrm{Ba}\ (10.70\ \mathrm{m}) \xrightarrow{\beta^-} {}^{142}\mathrm{La}\ (95.4\ \mathrm{m}) \xrightarrow{\beta^-} {}^{142}\mathrm{Ce}$$

$$^{142}\mathrm{Pr}\ (19.13\ \mathrm{h}) \xrightarrow[99.9836\%]{\beta^-} {}^{142}\mathrm{Nd}$$

$$^{143}\mathrm{Ce}\ (33.0\ \mathrm{h}) \xrightarrow{\beta^-} {}^{143}\mathrm{Pr}\ (13.56\ \mathrm{d}) \xrightarrow{\beta^-} {}^{143}\mathrm{Nd} \xleftarrow{\mathrm{EC}} {}^{143}\mathrm{Pm}\ (265\ \mathrm{d})$$

$^{144}\mathrm{Ce}\ (284.3\ \mathrm{d}) \xrightarrow[1.43\%]{\beta^-} {}^{144}\mathrm{Pr}\ (7.2\ \mathrm{m}) \xrightarrow[99.94\%]{\mathrm{IT}} {}^{144}\mathrm{Pr}\ (17.28\ \mathrm{m})$

$^{144}\mathrm{Ce}\ (284.3\ \mathrm{d}) \xrightarrow[98.57\%]{\beta^-} {}^{144}\mathrm{Pr}\ (17.28\ \mathrm{m}) \xrightarrow{\beta^-} {}^{144}\mathrm{Nd} \xleftarrow{\mathrm{EC}} {}^{144}\mathrm{Pm}\ (363\ \mathrm{d})$

$^{147}\mathrm{Nd}\ (10.98\ \mathrm{d}) \xrightarrow{\beta^-} {}^{147}\mathrm{Pm}\ (2.6234\ \mathrm{y}) \xrightarrow{\beta^-} {}^{147}\mathrm{Sm}\ (1.06\mathrm{E}11\ \mathrm{y}) \xrightarrow{\alpha} {}^{143}\mathrm{Nd}$

$^{148}\mathrm{Pm}\ (41.3\ \mathrm{d}) \xrightarrow[95.8\%]{\beta^-} {}^{148}\mathrm{Sm}$

$^{148}\mathrm{Pm}\ (41.3\ \mathrm{d}) \xrightarrow[4.2\%]{\mathrm{IT}} {}^{148}\mathrm{Pm}\ (5.37\ \mathrm{d}) \xrightarrow{\beta^-} {}^{148}\mathrm{Sm}$

$^{149}\mathrm{Nd}\ (1.73\ \mathrm{h}) \xrightarrow{\beta^-} {}^{149}\mathrm{Pm}\ (53.08\ \mathrm{h}) \xrightarrow{\beta^-} {}^{149}\mathrm{Sm}$

$^{151}\mathrm{Pm}\ (28.40\ \mathrm{h}) \xrightarrow{\beta^-} {}^{151}\mathrm{Sm}\ (90\ \mathrm{y}) \xrightarrow{\beta^-} {}^{151}\mathrm{Eu}$

$^{152}\mathrm{Sm} \xleftarrow[29\%]{\mathrm{EC}} {}^{152}\mathrm{Eu}\ (9.32\ \mathrm{h}) \xrightarrow[71\%]{\beta^-} {}^{152}\mathrm{Gd}\ (1.1\mathrm{E}14\ \mathrm{y})$

$^{152}\mathrm{Sm} \xleftarrow[72.2\%]{\mathrm{EC}} {}^{152}\mathrm{Eu}\ (13.6\ \mathrm{y}) \xrightarrow[27.8\%]{\beta^-} {}^{152}\mathrm{Gd}\ (1.1\mathrm{E}14\ \mathrm{y}) \xrightarrow{\alpha} {}^{148}\mathrm{Sm}$

$^{157}\mathrm{Gd} \xleftarrow{\mathrm{EC}} {}^{157}\mathrm{Tb}\ (150\ \mathrm{y}) \xleftarrow{\mathrm{EC}} {}^{157}\mathrm{Dy}\ (8.06\ \mathrm{h})$

$^{162}\mathrm{Gd}\ (9.7\ \mathrm{m}) \xrightarrow{\beta^-} {}^{162}\mathrm{Tb}\ (7.76\ \mathrm{m}) \xrightarrow{\beta^-} {}^{162}\mathrm{Dy}$

^{166}Ho (1200 y) $\xrightarrow{\beta^-}$ ^{166}Er

^{166}Dy (81.6 h) $\xrightarrow{\beta^-}$ ^{166}Ho (26.80 h) $\xrightarrow{\beta^-}$ ^{166}Er

^{171}Er (7.52 h) $\xrightarrow{\beta^-}$ ^{171}Tm (1.92 y) $\xrightarrow{\beta^-}$ ^{171}Yb

^{177}Lu (160.10 d) $\xrightarrow[78.5\%]{\beta^-}$ ^{177}Hf

^{177}Lu (160.10 d) $\xrightarrow[21.5\%]{IT}$ ^{177}Lu (6.71 d) $\xrightarrow{\beta^-}$ ^{177}Hf

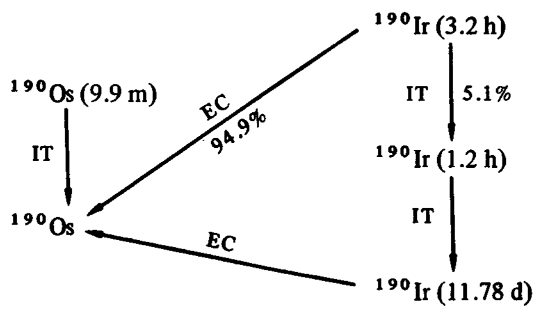
^{184}Re (169 d) $\xrightarrow[25.3\%]{EC}$ ^{184}W

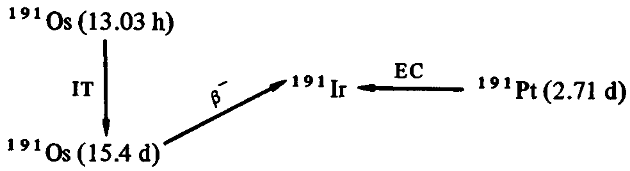
^{184}Re (169 d) $\xrightarrow[74.7\%]{IT}$ ^{184}Re (38.0 d) $\xrightarrow{EC}$ ^{184}W

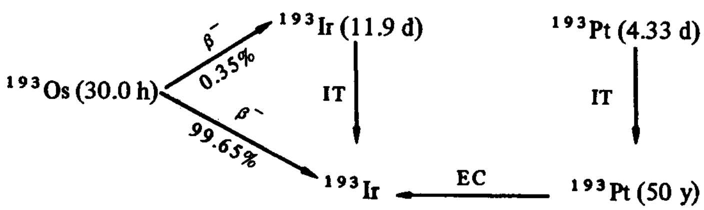
^{186}W $\xleftarrow[6.8\%]{EC}$ ^{186}Re (90.64 h) $\xrightarrow[93.2\%]{\beta^-}$ ^{186}Os (2.0E15 y) $\xrightarrow{\alpha}$ ^{182}W

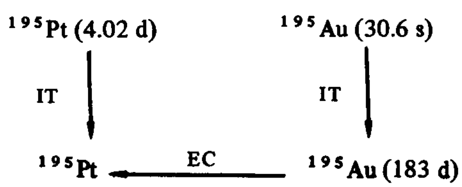
^{187}W (23.83 h) $\xrightarrow{\beta^-}$ ^{187}Re (4.7E10 y) $\xrightarrow{\beta^-}$ ^{187}Os

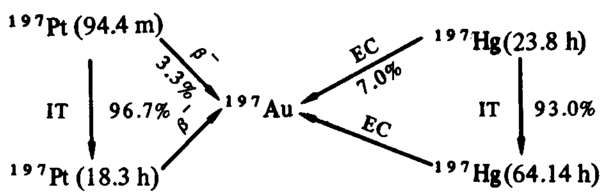
^{188}W (69.4 d) $\xrightarrow{\beta^-}$ ^{188}Re (16.98 h) $\xrightarrow{\beta^-}$ ^{188}Os


^{190}Ir (3.2 h)
^{190}Os (9.9 m)
IT
5.1%
EC
94.9%
IT
^{190}Ir (1.2 h)
^{190}Os
IT
EC
^{190}Ir (11.78 d)

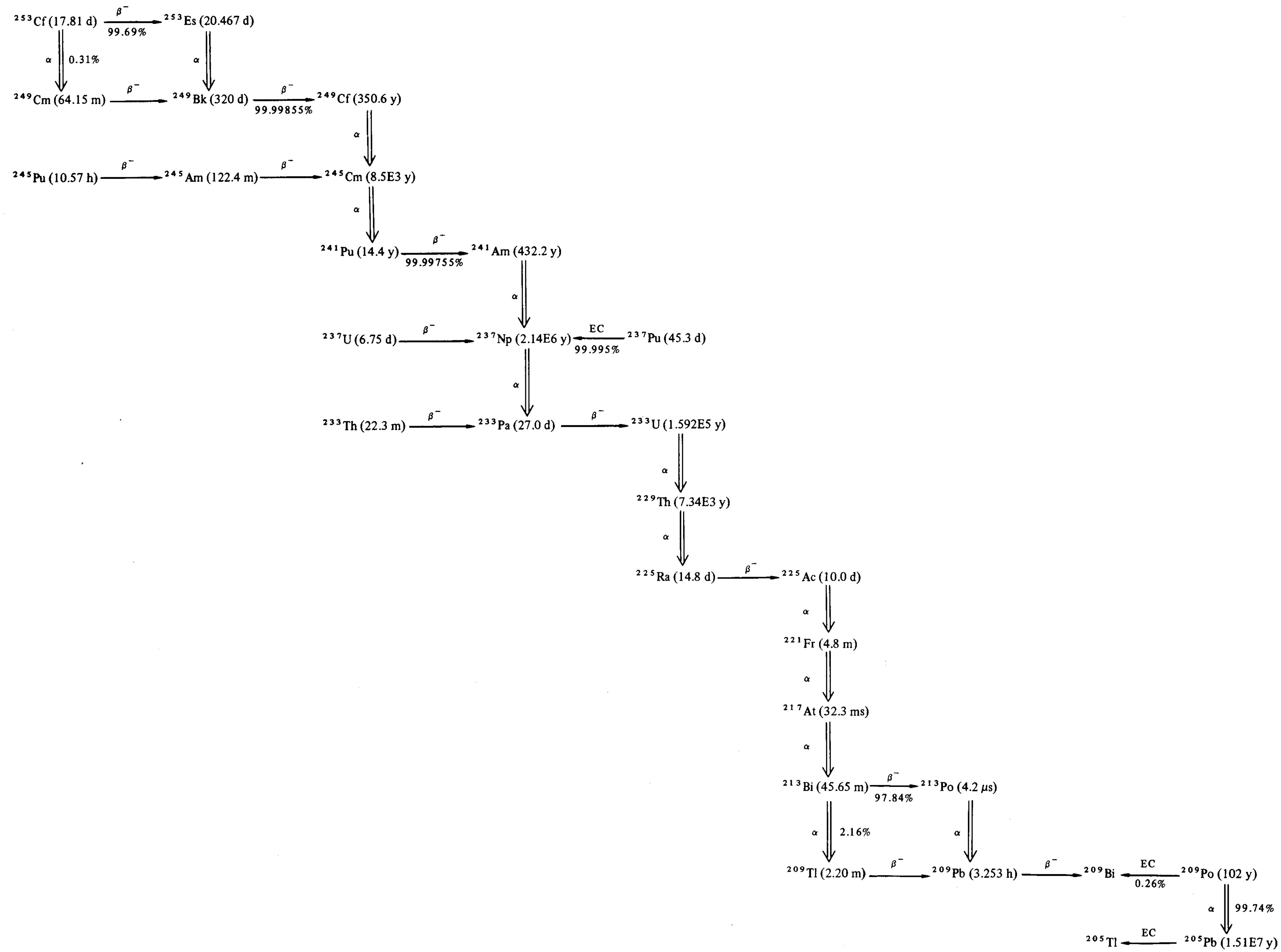

^{191}Os (13.03 h)
IT
β^-
^{191}Ir
EC
^{191}Pt (2.71 d)
^{191}Os (15.4 d)


^{193}Ir (11.9 d)
^{193}Pt (4.33 d)
β^-
0.35%
^{193}Os (30.0 h)
IT
IT
β^-
99.65%
^{193}Ir
EC
^{193}Pt (50 y)

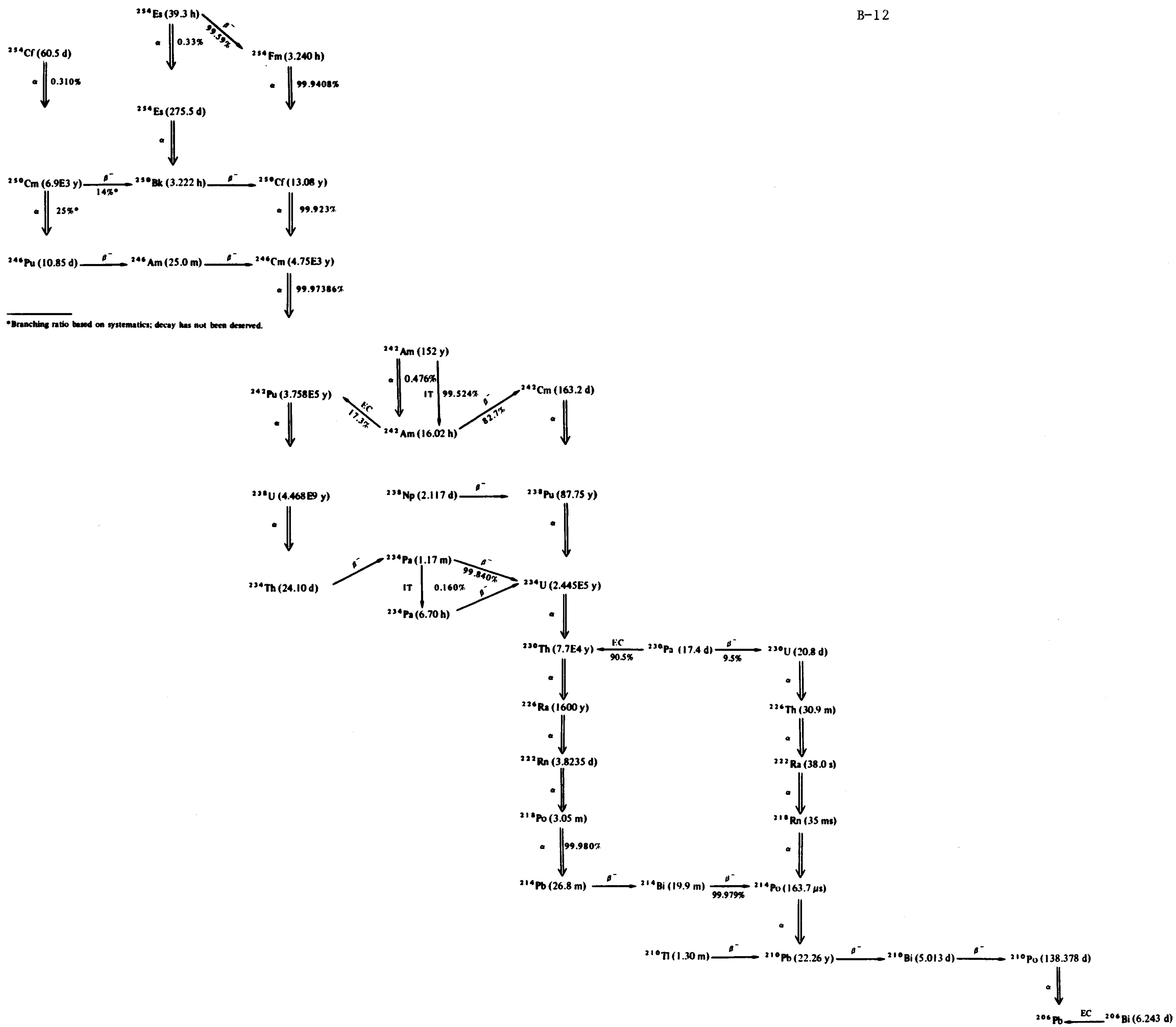

^{195}Pt (4.02 d)
^{195}Au (30.6 s)
IT
IT
^{195}Pt
EC
^{195}Au (183 d)


^{197}Pt (94.4 m)
β^-
3.3%
EC
7.0%
^{197}Hg (23.8 h)
IT
96.7%
β^-
^{197}Au
EC
IT
93.0%
^{197}Pt (18.3 h)
^{197}Hg (64.14 h)

Neptunium Series

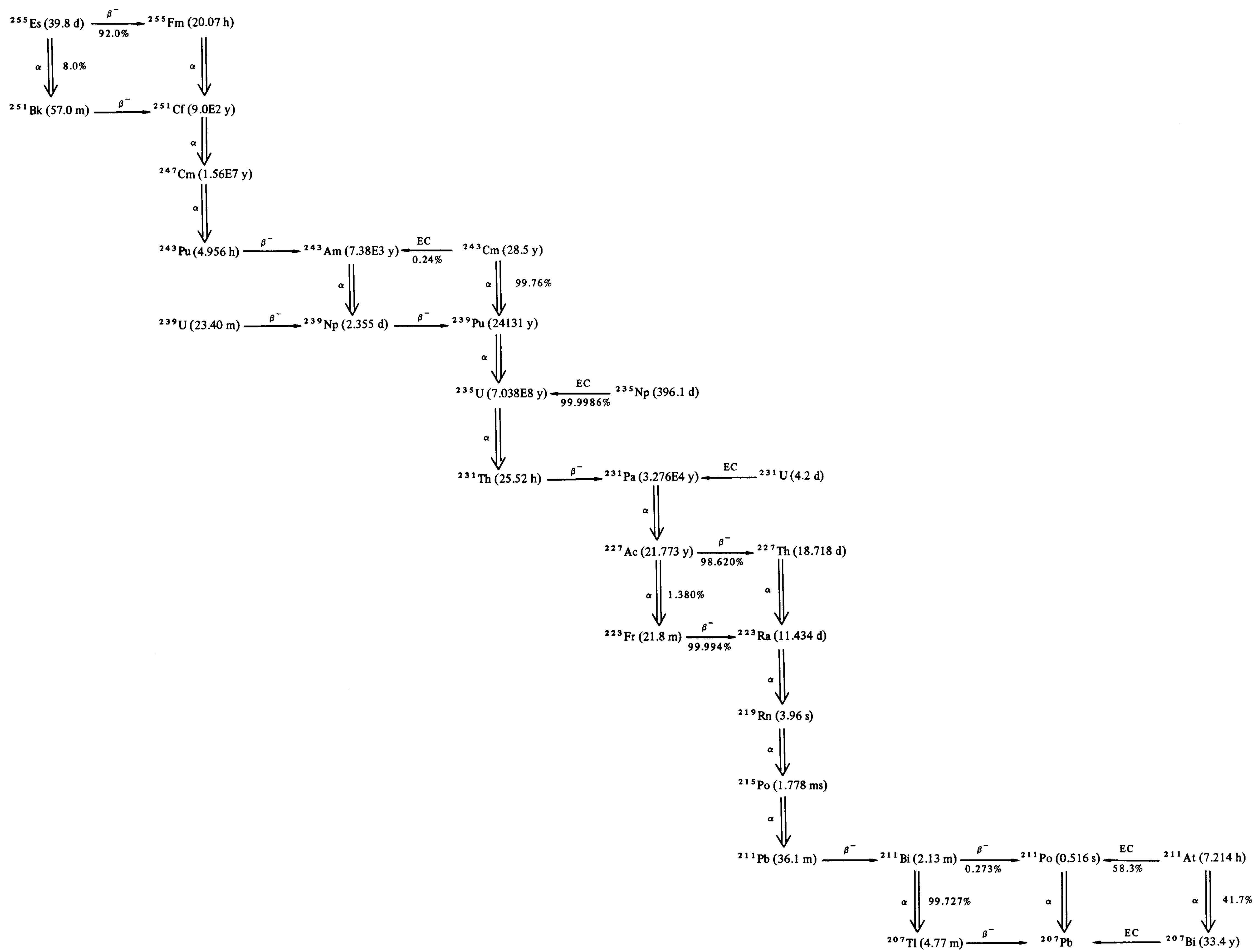


Uranium Series

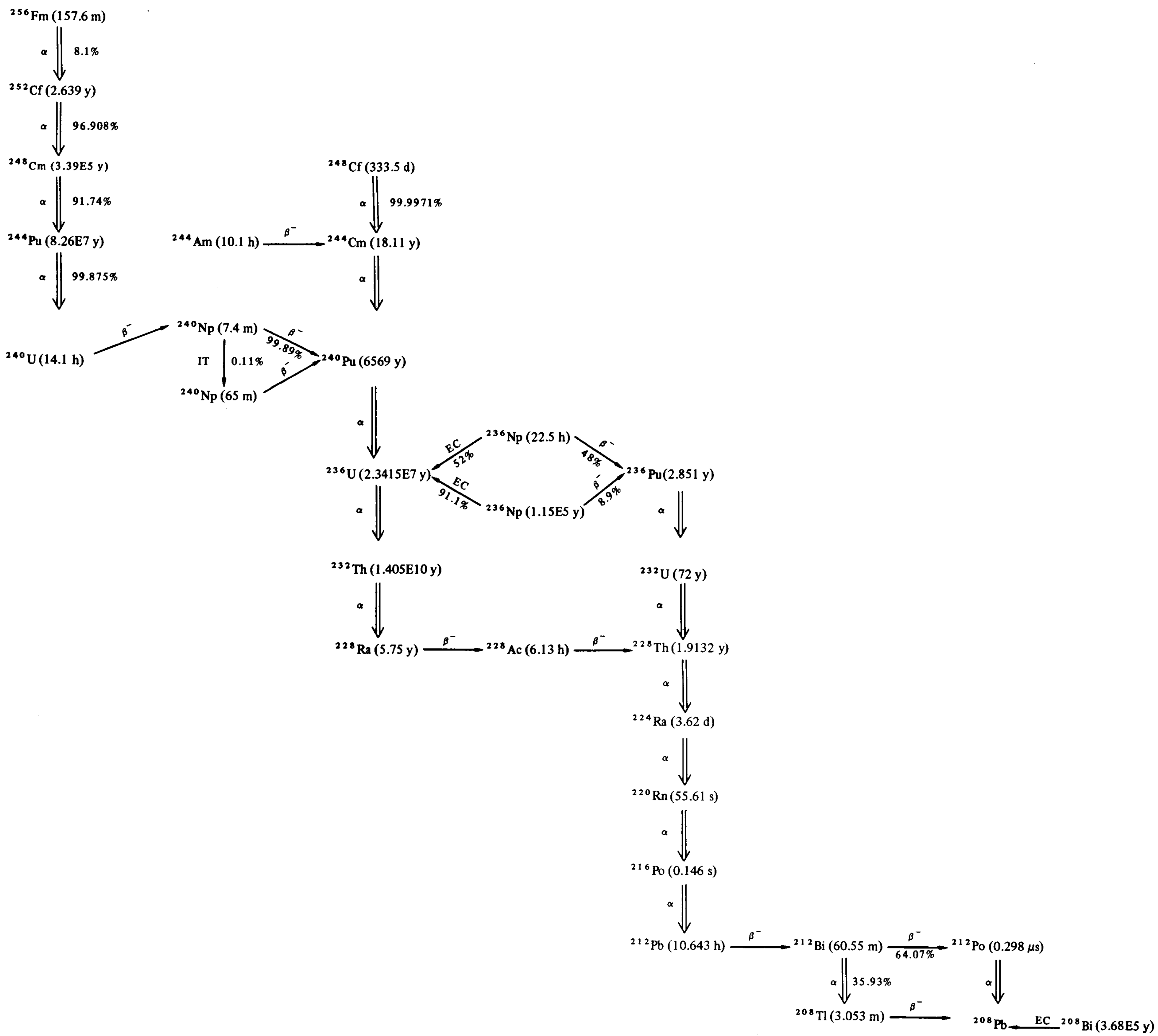


*Branching ratio based on systematics; decay has not been deserved.

Actinium Series



Thorium Series



B.1 DEFINITIONS

The symbols appearing in the previous decay chains and chain definitions are:

s	-	second
m	-	minute
h	-	hour
d	-	day
y	-	year
α	-	alpha-particle emission
β^+, β^-	-	beta-plus, beta-minus as positive or negative β-particle emission.
EC	-	orbital electron capture
IT	-	isometric transition (decay from an excited metastable state to a lower state).

B.2. REFERENCE FOR APPENDIX B

1. D. C. Kocher, Radioactive Decay Data Tables, DOE/TIC-11026, January 1981.

APPENDIX C. QUALITY ASSURANCE PROGRAM

The purpose of this section is to describe the Quality Assurance (QA) Plan for the Nuclear Wastes Characteristics (NWC) Program. The basic requirements of the plan have been developed in accordance with those reported in 10 CFR Pt. 60[1] and applicable parts of Appendix B of 10 CFR Pt. 50.[2] Many elements of the sections in these two documents are not applicable to the NWC Program because they were designed for geologic repositories, nuclear power plants, and fuel reprocessing plants, while the NWC Plan deals with data and information collection and dissemination. However, the 18 elements given in 10 CFR Pt. 50 have been used as guidelines in the following discussion.

C.1 ORGANIZATION

The NWC Program is sponsored by the Office of Civilian Radioactive Waste Management of DOE and is guided by the technical support team supplied by Weston. ORNL uses a matrix form of management which encompasses a dual pathway system. Funding control and administrative responsibilities follow the program line, while the execution of work and technical responsibilities follow the division line. Two divisions, Chemical Technology and Computing and Telecommunications, are responsible for the technical effort. QA requirements and/or guidance documents and in-house QA coordinators are as follows:

DOE Headquarters

"Quality Assurance Program Requirements for Nuclear Power Plants," ASME NQA-2-1983.

"Quality Assurance Program Requirements," Nuclear Energy Programs, DOE, Document RDT F2-2, August 1973.

DOE-Oak Ridge Operations

"Quality Assurance," DOE-ORO, Document OR IMD 02XX.

Oak Ridge National Laboratory

"Quality Assurance Program," Martin Marietta Energy Systems, Inc., Policy Procedures Manual GP-5, pp. 1-6, April 1, 1985.

"Quality Assurance Program," ORNL Standard Practice Procedure Supplement, D-2-16S.

ORNL QA Program Director: P. B. Hoke

Chemical Technology Division QA Coordinator: T. K. Bayles

Nuclear Waste Programs QA Coordinator: T. F. Scanlan

C.2 QA PROGRAM PLAN

This plan is based on program experience to date and will summarize the procedures used to collect and generate information for the report. The following three aspects will be discussed: (1) physical characteristics of spent fuel, (2) the decay characteristics as predicted by ORIGEN2, and (3) the construction of the data base. Simplified flow diagrams of the methods used for data collection and/or information generated are given in Figs. C.1–C.3.

C.2.1 Physical Characteristics of Spent Fuel

A detailed literature survey was performed utilizing the DOE/RECON[3] interactive on-line information retrieval system for references covering: fuel elements and assemblies, Westinghouse, Combustion Engineering, and General Electric standard reactors, and several specific reactors. Approximately 600 records were identified from the Energy Database and the Power Reactor Docket Information Database and examined. To augment the literature survey and to eliminate gaps and inconsistencies in the available data, a questionnaire covering selected physical characteristics of fuel assemblies and rods was prepared and sent to the various vendors. Their input was compared with literature citations and any discrepancies were resolved through telephone calls. Finally, a copy of the draft report was sent to each of the vendors for their review and comments. The final document contains all pertinent revisions requested by the vendors. Figure C.1 depicts a general flowsheet of the process.

C.2.2 Radioactive Decay Predictions by ORIGEN2

The principle guiding the development of any computer code is that the phenomena of interest can be described analytically to a sufficient level of accuracy. A predictive code is typically a compilation of individual mechanisms (e.g., neutron flux, radioactive decay, photons per decay, neutron cross sections) which are combined into an overall analytical scheme. Usually the mechanisms and the methods by which they are combined are based on a variety of a priori assumptions. The QA process must be concerned with each of the following aspects: the modeling of the individual mechanisms, the manner in which the mechanisms are combined, the underlying assumptions, and the possible lack of inclusion of important mechanisms. With respect to ORIGEN2 as well as many other computer codes, the techniques of validation and verification can be applied.

The output from any code is the prediction of one or more quantities of interest - for the ORIGEN2 code these may include mass concentration, radioactivity, thermal power, and neutronic and photonic emissions. In any case, there are specific output information and output quantities of interest that provide the reason for performing the calculation. The validation process relates then to the confidence that can be placed on the accuracy of values predicted for these specific output quantities. The specific output quantities of importance are seen to serve as "indicators" for determining whether or not a code provides satisfactory agreement with experimental results (i.e., indicators that can be used to measure a code level of verification). The identification or choice of indicators is a crucial step in the validation process because that forces a selection of the most important output variables and provides a basis for making quantitative specifications of validation goals. By comparing the selected output quantities (indicators) with experimental data in a quantitative manner, one can evaluate the extent to which validation goals are achieved.

Unfortunately, very few adequate benchmarks exist for verification purposes for ORIGEN2, particularly in the case of modern light-water reactors. Virtually no measurements have been made of either photon spectra or neutron emission rates, and verification will be extremely difficult because of the dependence of measurements on self-shielding, geometry, and detector efficiency. The benchmark status with respect to the composition and thermal power is somewhat better since measurements have been made and documented. One of the major problems is the lack of detailed information on the power history of the benchmark spent fuel assemblies. However, a few comparisons have been made between ORIGEN2 and reasonably characterized materials. The results from these tests are summarized in papers by Croff[4] and Schmittroth.[5] Also, the Materials Characterization Center (MCC), which has the responsibility to provide spent fuel approved testing materials for use in investigation of nuclear waste forms, has performed selected radionuclide analyses and compared their results with ORIGEN2.[6]

Schmittroth found that the agreement between ORIGEN2 decay heat calculations and calorimetric measurements of PWR spent fuel assemblies was within 5%. However, as he stressed, his conclusions are appropriate only for intermediate decay times (<50 y). Preliminary conclusions from the MCC study showed a good correlation (±5%) between selected radionuclide concentrations (^{235}U, ^{239}Pu, and ^{137}Cs) and ORIGEN2 predictions, but the values calculated for ^{99}Tc and ^{237}Np were between 7 and 10% higher than those determined experimentally. Croff summarized the earlier work (in substantial agreement with the previous citations) and identified specific areas that need to be undertaken to improve and extend the usefulness of ORIGEN2. The calculational pathway used by ORIGEN2 is shown in Fig. C.2.

C.2.3 Data Base Construction

One of the governing factors in constructing a data base from computer-generated information is to eliminate the possibility that data will be lost or modified during the formulation of the system. As is shown in Fig. C.3, all data have been handled electronically, and data manipulation and the preparation of the data sets were performed on-line using two main frame computers (IBM-3033 and PDP-10) and a personal computer (IBM-PC). In addition, random checks were made by experts after each of the transfers to ascertain that there was no loss of information. Next, data from each of the burnup cases for each of the three output files were compared with information taken from the original ORIGEN2 output. The diskettes containing the data base were then sent to the technical monitor, sponsor, and lead sites for review and critique. Their input was used to assist in preparing a revised edition of the data base.

C.3 REFERENCES FOR APPENDIX C

1. Code of Federal Regulations, Disposal of High-Level Wastes in Geologic Repositories, 10 CFR Part 60, pp. 563-617 (January 1985).
2. Code of Federal Regulations, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, 10 CFR Part 50, App. B, pp. 475-78 (January 1985).
3. U. S. Department of Energy, DOE/RECON User's Manual, DOE/TIC-4586-R1, May 1981.
4. A. G. Croff, "ORIGEN2: A Versatile Computer Code for Calculating the Nuclide Composition and Characteristics of Nuclear Materials," Nucl. Technol. 62, 335-52 (September 1983).
5. F. Schmittroth, "ORIGEN2 Calculations of PWR Spent Fuel Decay Heat Compared with Calorimeter Measurements," Proceedings Fuel Reprocessing and Waste Management, pp. 2-69 to 2-74, August 1984.
6. J. O. Barner, Characteristics of LWR Spent Fuel MCC - Approved Testing Material - ATM-101, PNL-5109, June 1984.

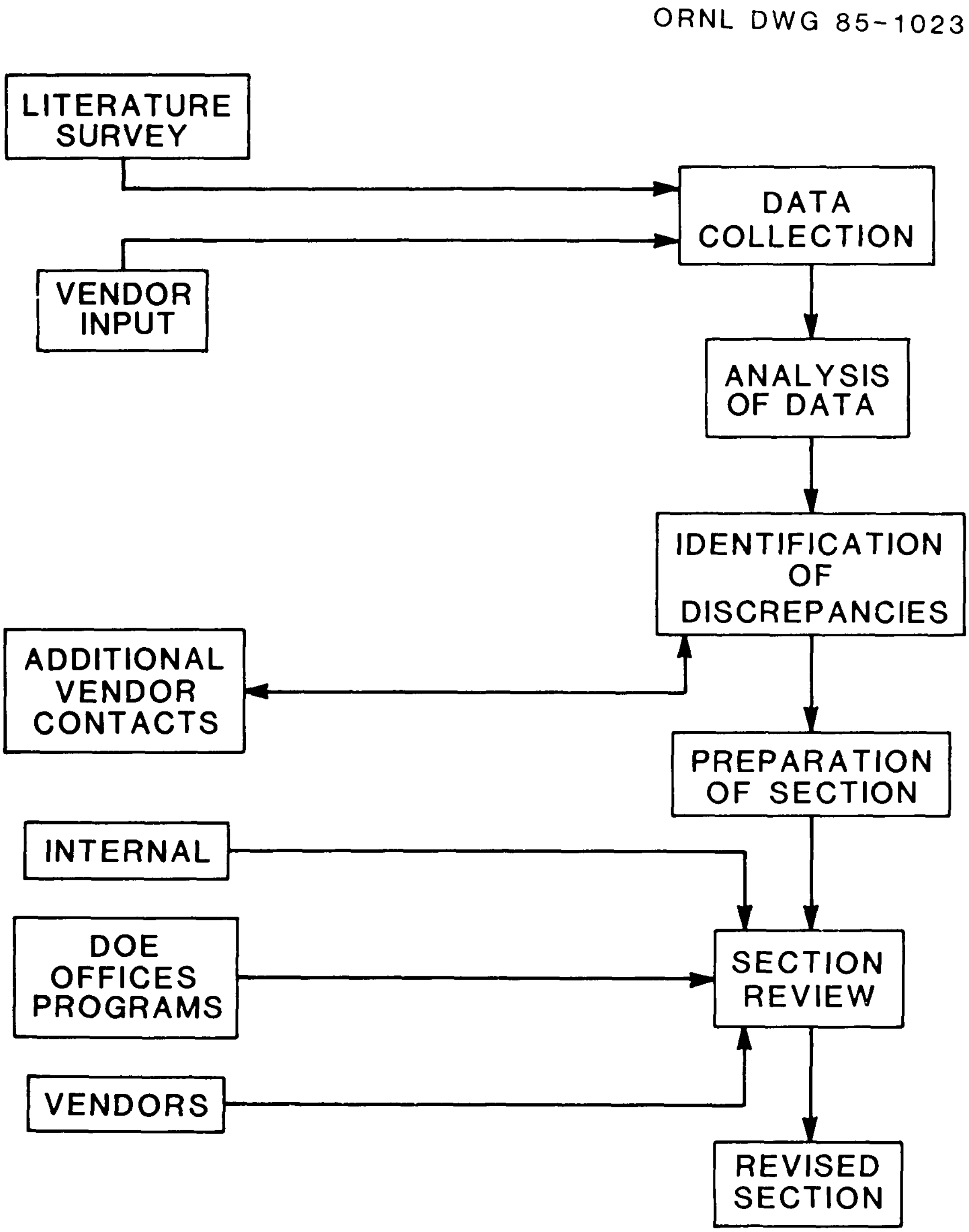


Fig. C.1. Data collection pathway for spent fuel characteristics.

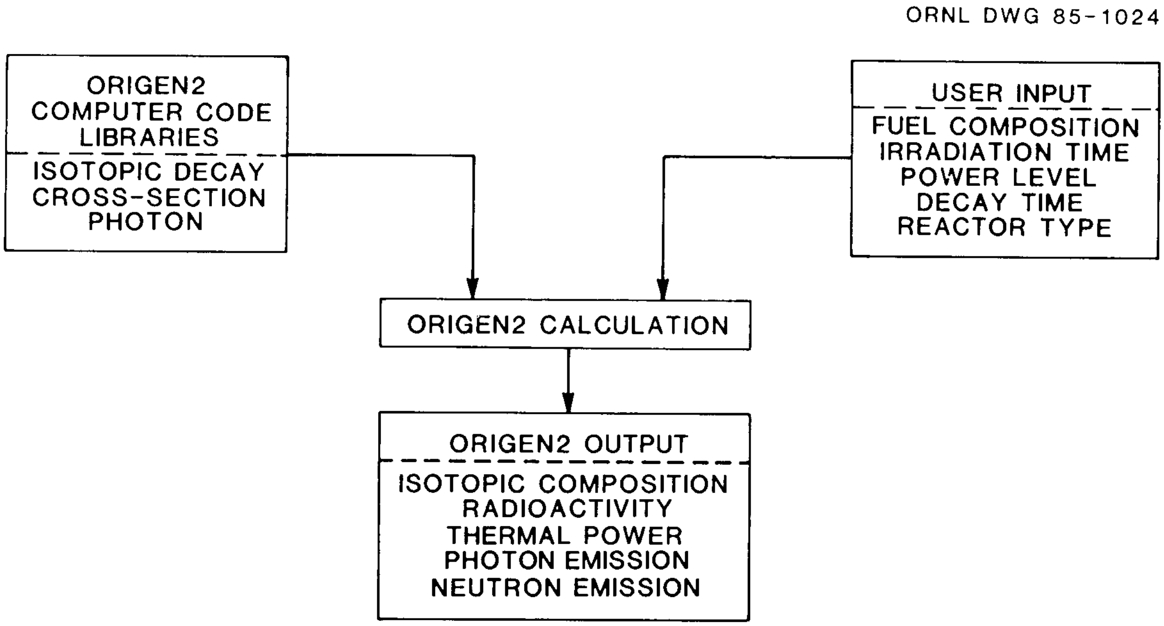


Fig. C.2. Calculational pathway for ORIGEN2-generated data.

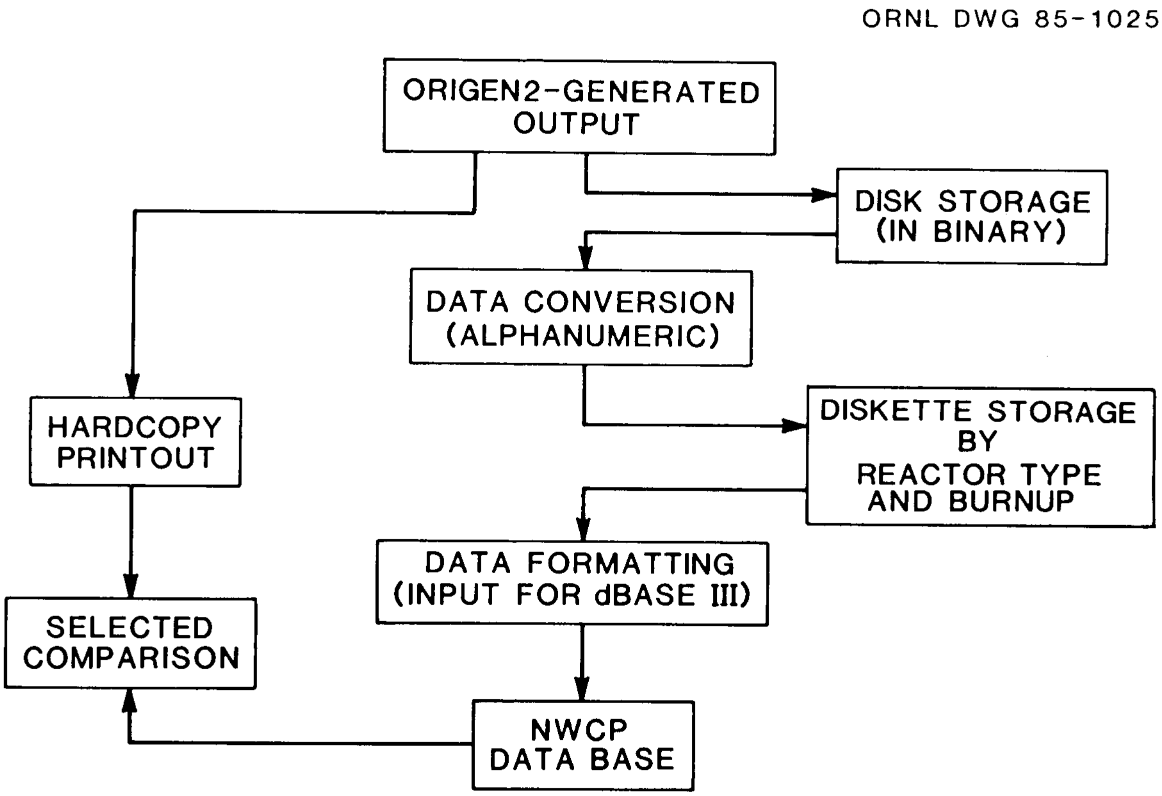


Fig. C.3. Construction pathway for the nuclear waste characteristics data base.

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