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The United States nuclear industry is under significant public pressure to close the nuclear fuel cycle. Closing the fuel cycle will require that nuclear waste, fission products and transuranics, except plutonium (actinides), be disposed of by permanent burial. The actinides can be buried with the fission products, but this will increase the time that nuclear waste must be isolated from the environment. This paper will describe criteria for design of special reactors to fission actinides as well as new reactors that do not produce actinides.

This paper concluded that the single most important consideration when designing a reactor to burn actinides, or not produce, actinides is to maintain as hard a neutron spectrum (mostly high energy neutrons) as possible to fission the actinides and not produce new actinides. To maintain a hard neutron spectrum, the use of U-238 and the use of non-fuel nuclides (oxygen, carbon, molybdenum, sodium, steel) should be minimized. An actinide burning reactor should minimize the use of driver fissile nuclides (U-235 and Pu-239) so that most of the fissions occur in the actinides.

Hard Spectrum Reactor Design Criterion for Actinide Utilization

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Battelle is responsible for the initial typing, the graphics and the final copying of this thesis. I have found working at Battelle to be so enjoyable and rewarding that I would recommend a career at Battelle to any OSU graduate.

This project could not have been completed without the love, support and patience of my wife, Megeen. Meg's support was the driving force behind that finally got me to complete this project after I had been out of school for nearly six years. I will forever be grateful for Meg's help in the final editing of this thesis. With my love I dedicate this thesis to Meg and I hope that Meg will accept the completion of this project as a token of my love for her.

My final comment is to those students that have yet to complete their theses. When you finally finish your thesis, sleeping in on the weekends is much more enjoyable after the guilt of not having finished has been removed.

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Hard Spectrum Reactor Design Criterion for Actinide Utilization

1.0 Introduction

The nuclear industry in the United States is under serious attack from the public because the industry has failed to close the nuclear fuel cycle. This attack appears to be justified in the face of the present stockpile of spent fuel and the predicted enormous buildup of spent fuel in the future.[1,2,3] The spent fuel can be broken into four major categories: uranium, plutonium, fission products and actinides (in this paper actinides will be considered to include neptunium, americium, curium, berkelium and californium). The uranium in the spent fuel has been depleted from 2-4% U-235 at initial loading to 0.5%-1.0% U-235 when discharged. U-235 is valuable because it is the only fissionable nuclide found in nature and the recovered U-235 could be used again. The plutonium in the spent fuel is also valuable because it contains significant portions of Pu-239 and Pu-241 that are fissionable and can be used as light water reactor (LWR) fuel. The cesium and strontium recovered from the spent fuel could have substantial value to the public as a radiation source for

food irradiation or as a power source used in remote locations. The noble metals (palladium, rhodium, and ruthenium) recovered from the spent fuel have some value. However, these metals are slightly radioactive that could limit their use in industry. The actinides are formed by the capture of neutrons in U-238 (the main component of uranium) or capture of neutrons in plutonium that does not result in fission. The fission products and the actinides need to be disposed of so that the nuclear industry can close the fuel cycle and protect the public from any potential hazard from these materials.

At the present time the nuclear industry is not interested in doing anything with spent fuel; either because the price of uranium is so low that there is no economically compelling reason to reprocess the spent fuel and recycle the uranium and plutonium, or the nuclear industry does not believe that the political system will allow the industry to reprocess the spent fuel even when they demonstrate that spent fuel can be reprocessed safely. Public concern, the political climate and the price of uranium will eventually change, however, making it possible

to once again reprocess spent fuel in this country. When reprocessing is in full swing, the problem of disposing of fission products and actinides will need to be dealt with.

Fission product waste management is well in hand for two reasons. First, the technology to isolate fission products from the biosphere appears to be in hand [4-7] even if the political process has slowed the implementation of this technology. Second, most of the fission products, which are radioactive and therefore a hazard, have relatively short half lives. The hazard of a radioactive material is based on the decay type, the decay energy, the decay rate and the quantity of material. Most of the fission products decay by gamma and beta emissions. The average energy of these particles is on the order of 1 MeV.[8] 92.1% of the fission products have half lives less than 30 years[5]. These fission products have very high decay rates and are very hazardous, but a half life of 30 years or less means that in 600 years the concentrations of these fission products will have dropped by at least one million. Fission products with half lives greater than 100,000 years (7.8% of the fission products)[5] have very low decay rates and are therefore a relatively low hazard. Fortunately, there are virtually no fission products (.1%)

in the 30 year to 100,000 year half life range that would be moderately hazardous and persistent. Geological isolation[4-7] becomes an attractive solution for fission product waste management.

The actinide waste management problem is less certain than fission product disposal. While the technology proposed for fission product isolation from the biosphere will work equally well for actinides, the half lives of the actinides are primarily in the 10 to 100,000 year range. Actinides generally do not decay to a stable nuclide like most fission products but rather to other radioactive nuclides. The long half lives and long decay chains would require isolation for millions of years. These extended isolation times can be argued as reasonable[4,5,6] if the isolation site is carefully selected (salt domes, granite, volcanic tuff or other sites that are stable and have little or no groundwater movement) and the isotope migration mechanisms are included as part of the isolation. The actinide waste disposal problem has led to a number of studies indicating that removal of actinides from the fission product waste will reduce the long term waste hazard.[9-13]

Estimates of actinide material buildup[3] indicate that in 1980 there were 3 tonnes of actinide material. Those same estimates indicate that by 2000 there will be 57.9 tonnes of actinide material accumulating at the rate of 2.2 tonnes per year. The estimated accumulation rates suggest three possible solutions to the actinide buildup problem. The first is to simply bury the actinides as they are produced, presently the course the nuclear industry has chosen. The second choice is to design reactors that do not produce actinides; however, reactors of this type could not be dominant until after the year 2000 (assuming a 15 year construction time). This would also require changing the basic design of the reactors used in the United States and probably would not be feasible with the public's present perception of the nuclear industry. The third solution would be to build a few actinide burning reactors (ABR's) to dispose of the actinide material as it was recovered from spent fuel. This possibility led to several studies attempting to fission actinides in a few dedicated ABR's as an alternative to burial for actinide disposal.[14-21]

These previous studies attempted to fission actinides in light water reactors (LWR), liquid metal fast breeder reactors (LMFBR), and fusion devices. These actinide burner reactor studies produced unimpressive results for two basic reasons. The first problem is that LWRs and LMFBRs, as currently envisioned, have neutron energy distributions (called neutron spectra) that have average neutron energies far less than the fission threshold of most actinides. This has led to reactor designs that produced more actinides than were fissioned, or at best broke even. The second problem with recycling actinides in LWRs or LMFBRs is that each reactor can handle only small quantities of actinides; therefore, many more reactors would be needed to dispose of the actinide material. The fusion reactor would have neither of these problems, but it is not presently feasible. Beaman[21] suggested a solid metal sphere of actinides leading to spectacular burnout rates. However, the fuel melted in the metal sphere, which is clearly not satisfactory. Beaman then suggested an actinide burner reactor based on a proposed commercial LMFBR design using oxide fuel that burned more actinides than it produced. This study will demonstrate improved actinide burnup by optimizing the reactor design to fission actinides.

The objective in designing an ABR is to design a reactor that fissions those actinides that will not fission with low energy neutrons. All of the actinides, uranium, and plutonium have approximately the same neutron fission cross section of 1 to 2 barns above 300 keV while below this neutron energy only a few actinides (Am-242m, Cm-243, Cm-245, Cf-245, and Cf-251) have appreciable fission cross sections. This high neutron energy fission threshold requires a demanding reactor design.

A reactor designed to produce power by fissioning actinides in a hard neutron spectrum (a reactor with most of the neutrons above the fission threshold of most actinides) can be achieved by using a few simple ideas. The first goal in the design of an ABR is to reduce the number of non-actinide nuclides in the core, similar to the metal actinide sphere discussed by Beaman[21]. The non-actinide nuclides reduce the energy of a neutron with each interaction, thus reducing the average neutron energy in the core and reducing the number of neutrons that can cause fission in all of the actinides. The second goal in designing an ABR is to reduce the amount of uranium in the core. Uranium, particularly U-238, has a high inelastic scattering cross section that significantly reduces the

average neutron energy with each interaction. The third goal in designing an ABR is to reduce the amount of plutonium or U-235 in the core. Plutonium (specifically Pu-238, Pu-239, and Pu-241) or U-235 is a double-edged sword in an ABR because plutonium or U-235 fissions using the low energy neutrons, thereby increasing the average neutron energy. However, plutonium or U-235 also fissions with high energy neutrons, thus competing with the actinides for high energy neutrons and reducing the number of actinide fissions for a specific power level. These goals need to be combined in such a way as to produce a reactor that has a high average neutron energy while fissioning actinides in a configuration that can be cooled (unlike Beaman's[21] metal sphere design).

These design goals suggest that an ABR should be designed much like Experimental Breeder Reactor-II (EBR-II). EBR-II in all likelihood has the hardest neutron spectrum (or the highest average neutron energy) of any power reactor presently operating in the world. EBR-II is a proven design that has worked for the last 29 years providing the bulk of the data now available about LMFBRs. The EBR-II design, along with a design based on the Super

Phenix (LMFBR) and a Gas Cooled Fast Reactor (GCFR), will be examined in this paper as possible reactor designs. Several possible fast reactor fuel types will be examined as well.

This paper will present the important ideas that need to be considered when an ABR is designed. Chapter 2 will describe possible designs and fuel types of an ABR. Chapter 3 will describe the actinide cross sections pointing out particularly the need for high energy neutrons to fission the actinides. Chapter 4 will describe the effects of infinite lattice reactor design on the neutron spectrum and average neutron energy. Chapter 5 will show how different infinite lattice reactor designs change the calculated amount of actinide material removed. Chapter 5 will also present the calculated results for each of the actinide fuels burned in a small region of the EBR-II reactor. Chapter 6 will present the conclusions of this paper. Appendices A, B, C, and D have been included for the reader who would like more description of the NCINR8 (pronounced incinerate) code and cross section library used. Appendix E presents the infinite neutron energy

distributions (spectra) for the liquid metal fast breeder reactor and the gas cooled fast reactor configurations. It is hoped that after reading this paper, it will be clear that a special ABR can be designed to fission actinides.

2.0 Preliminary Actinide Burner Reactor Designs

This chapter will describe the rod configurations and fuel types that were examined in this paper. The rod configurations have been based on the Experimental Breeder Reactor II (EBR-II), the Super Phenix reactor (a French commercial liquid metal fast breeder reactor) (LMFBR), and a gas cooled fast reactor (GCFR). The LMFBR and the EBR-II reactors are currently operating while the GCFR has been considered as a possible fast reactor. The fuels considered are a pure actinide metal, an actinide metal with 50 weight percent molybdenum, an actinide oxide, and an actinide carbide. Since the ABR design is so demanding, a standard EBR-II core will be used to drive a 10 cm diameter actinide region because some of the actinide fuel configurations will not be critical.

The isotopic ratios of the actinide fuels examined here are determined by the fractions of plutonium and uranium removed during reprocessing and the time between reactor discharge and reprocessing. The actinide fuel studied in this paper was assumed to come from a typical LWR and allowed to cool for two years before reprocessing. The uranium, plutonium and actinides isotopic for

discharged LWR fuel have been reported in reference 22. During reprocessing it was assumed that 99.9% of the plutonium was removed and either 99.9% of the uranium was removed (referred to as ACT3X9) or 99.9999% of the uranium was removed (referred to as ACT6X9). Removing 99.9999% of the uranium may not be feasible, but the ACT6X9 fuel represents the upper practical limit for an actinide fuel. The isotopic number densities of pure metallic ACT3X9 and ACT6X9 are shown in Table 1.

This study used four fuel types with either ACT3X9 or ACT6X9 that have been considered for fast reactors at various times. The first fuel examined in this study is an all metal fuel similar to the fuel used originally in EBR-II. The second fuel considered is also a metal fuel that is 50% molybdenum and 50% actinides by volume. The last two fuels considered for the ABR are an actinide oxide fuel (2 oxygen atoms per actinide atom) and an actinide carbide fuel (1 carbon atom per actinide atom). The use of diluents, molybdenum, oxygen, or carbon in the fuel reduces the amount of actinides that can be inserted into a pin of a fixed size and increases the power density (kw/kg) in the fuel. The use of a diluent such as oxygen or carbon will produce a fuel with different and possibly more desirable

material characteristics, while the use of a molybdenum diluent could substantially increase the actinide power density. A comparison of the relative actinide metal density for each of the four fuel types is shown in Table 2. Also, Table 2 has the number densities of the diluents used in each of the four fuel types (metal has no diluent).

This study will use a standard EBR-II core with all of the rod configurations and fuel types, since some of fuels cannot be made critical. The EBR-II core has been modeled as a metal fuel that is 80 cm in diameter and 40 cm tall with 45.85 atom percent U-235, 42.49 atom percent U-238, and 11.66 atom percent molybdenum.

Three different fuel configurations are used in this study. The fuel configurations represent an EBR-II configuration, commercial LMFBR configuration (Super Phenix was used as a model) and a hypothetical GCFR configuration. These fuel configurations are described in Table 3. The pin size and configuration is assumed not to change with fuel type. The reactor for all conditions will produce an average power of 600 kw/L of thermal energy. The initial

fuel load requirements for an ABR the size of EBR-II (80 cm diameter by 40 cm tall) are shown in Table 4 for all of reactor types and fuel types with both ACT6X9 and ACT3X9 actinide materials.

TABLE 1. Composition of the Pure Metal Actinide Fuel in Atoms/Barn-cm

Isotope	ACT6X9	ACT3X9
U-234	7.5336E-09	3.4029E-06
U-235	4.8216E-07	2.1775E-04
U-236	2.7130E-07	1.2256E-04
U-238	5.6822E-05	2.5670E-02
Np-237	3.0430E-02	1.3742E-02
Pu-238	9.5847E-06	4.3284E-06
Pu-239	2.8770E-04	1.2993E-04
Pu-240	1.1644E-04	5.2544E-05
Pu-241	4.7963E-05	2.1650E-05
Pu-242	2.0528E-05	9.2696E-06
Am-241	6.9852E-03	3.1576E-03
Am-242m	3.7752E-05	1.7953E-05
Am-243	4.2981E-03	1.7411E-03
Cm-242	4.2621E-06	1.9729E-06
Cm-243	3.1836E-06	1.4383E-06
Cm-244	1.0570E-03	4.7702E-04
Cm-245	7.8954E-05	3.5662E-04
Cm-246	9.4360E-06	4.2621E-06
Cm-247	1.2007E-07	5.4195E-08
Cm-248	7.7944E-09	3.5231E-09
Cf-250	9.2847E-11	4.1937E-11
Cf-252	1.5345E-12	6.1375E-13

TABLE 2. Heavy Metal Fraction and Fuel Diluent Number Densities for ACT3X9 Fuel

	Metal Density Pure Metal Density	Diluent Number Density
Metal	1.0000	0.0000E+00
Carbide	0.6598	2.3717E-02
Oxide	0.5301	4.4570E-02
50 wt % Molybdenum	0.3714	4.0907E-02

TABLE 3. Fuel Pin Description of Various Actinide Burning Reactors (Dimensions are in millimeters)

	EBR-II	LMFBR	GCFR
Fuel Pitch	5.66	8.64	11.4
Pin OR	2.21	3.75	4.11
Pellet OR	1.83	2.75	3.69
Clad Thickness	0.23	0.55	0.3
Gas Gap	0.15	0.45	0.12

TABLE 4. Initial Actinide Fuel Load for Various Actinide Burning Reactor Designs and Fuel Types

Reactor Configuration	Fuel Type	Tonnes of ACT6X9 Loaded Into the Reactor	Tonnes of ACT3X9 Loaded Into the Reactor
EBR-II	Metal	1.308	1.376
LMFBR		1.271	1.337
GCFR		1.315	1.383
EBR-II	50 wt %	0.486	0.511
LMFBR	Molybdenum	0.472	0.496
GCFR		0.488	0.513
EBR-II	Oxide	0.693	0.729
LMFBR		0.674	0.709
GCFR		0.697	0.733
EBR-II	Carbide	0.863	0.908
LMFBR		0.839	0.882
GCFR		0.868	0.912

3.0 Actinide Cross Sections

The design of a successful Actinide Burner Reactor (ABR) requires a thorough understanding of how each material will affect it. Insight to the affect of materials in an ABR can be gained by the examination of each materials neutron interaction cross section.

The infinitely dilute cross section (a single atom in a uniform flux) collapsed into two large energy groups will provide a convenient way to study the cross sections for the materials, which might be used in an ABR. The cross section collapsing was done to preserve the relative reaction rates if the individual nuclides were in a fission neutron spectrum. The cross sections were collapsed into a "fast" group ranging in neutron energy from 10 MeV to .498 MeV and a "slow" group ranging from 498 keV to 41 keV. Neutron energies below 41 keV were not considered since an actinide burning reactor should have very few neutrons below 41 keV. The "fast" group in an ABR represents the neutrons produced during the fission process, while the "slow" group of neutrons are produced from fission neutrons scattering off of material in the reactor. The results of the cross section collapsing are shown in Tables 5 through

9 for total cross section, scattering cross section, fission cross section, absorption cross section, and the fission-to-capture ratio (not really a cross section), respectively.

Table 5 shows the total cross sections for possible ABR materials. The total cross section for the actinides is nearly constant in the "fast" group (7 to 8 barns), while the "slow" group ranges over 8.5 to 12.3 barns. The nearly constant fast cross section indicates that all of the actinides have an equal probability to interact with a fast neutron. The non-actinide materials have somewhat smaller total cross sections, however, these materials are almost entirely neutron scatters.

Table 6 shows the scattering cross section for possible ABR materials in the "fast" and "slow" groups. This table shows for the "fast" group, Uranium has 3 of the highest 4 scattering materials; and that U-238 has the highest probability of scattering of all the materials considered. A single scattering event in the "fast" group removes a significant amount of the neutron energy since the scattering interactions are mostly inelastic events, which means that some of the neutron energy is given up to

the nucleus. This clearly demonstrates why removing as much uranium as possible is important in an ABR design. The scattering cross sections of oxygen and molybdenum, while small, still represent the single largest source of scattering events in the actinide fuel. Using these isotopes, oxygen outnumbers the actinides by a factor of two and the molybdenum outnumbers the actinides by a factor of about three. The scattering events in oxygen and molybdenum make these materials undesirable for an ABR. Oxygen has a very large (14 barn) inelastic scattering cross section at 442 keV that significantly degrades the neutron spectrum in an ABR. The scattering events in the "slow" group are not important in terms of the ABR since the neutrons in this group usually do not fission many of the actinides anyway.

Table 7 shows the fission cross section for the actinides in the "fast" and "slow groups. In Table 7 it becomes clear why a hard spectrum reactor is required to fission actinides. The actinides all have "fast" fission cross sections that are 1 to 2 barns and are comparable to the "fast" fission cross sections of Pu-239 and Pu-241; while the "slow" fission cross sections for many of the actinides are insignificant when compared to the fission

cross sections of Pu-239 and Pu-241. The relatively small "slow" fission cross section of the actinides will result in most of the fissions occurring in plutonium with the actinides capturing neutrons and moving up the actinide chain, exactly the opposite of destroying actinides.

Table 8 shows the capture cross sections for possible ABR materials in the "fast" and "slow" groups. The capture cross sections are small in both the "fast" and "slow" groups. The non-actinide materials that might occur in an ABR all have very small absorption cross sections making them insignificant to the ABR except for the use as structural material and as a source of neutron scattering events. The capture cross section of the actinide is only important when compared to the fission cross section.

Table 9 has the fission-to-capture cross section ratio for the actinides. The fission-to-capture ratio is the number of fissions that occur for every capture event. The objective of an ABR is to fission the actinides so as not to form new actinides. The comparison of the fission-to-capture ratio in Table 9 clearly shows how effective the "fast" group is compared to the "slow" group with most of the actinides having a fission-to-capture ratio in the

"fast" group 10 times larger than in the "slow" group. The importance of the fission-to-capture ratio of the actinide can be visualized by assuming that a person could (magically) place 1000 grams of Am-241 in a pure "fast" spectrum and 1000 grams of Am-241 in a pure "slow" spectrum. By the time all of the Am-241 had either captured a neutron or fissioned, 922 g of the Am-241 in the "fast" spectrum would have fissioned while only 57 g of the Am-241 in the "slow" spectrum would have fissioned.

TABLE 5. Total Microscopic Cross Section for Possible Actinide Burning Reactor Materials

Nuclide	Microscopic Cross Sections "Fast"	"Slow"
U-234	8.0	10.7
U-235	7.3	9.7
U-236	7.6	10.5
U-238	7.4	10.2
Np-237	7.3	9.5
Pu-238	7.1	9.6
Pu-239	7.4	7.7
Pu-240	7.2	9.7
Pu-241	7.6	11.0
Pu-242	7.2	9.4
Am-241	7.3	10.4
Am-242m	7.0	11.6
Am-243	7.5	10.7
Cm-242	7.1	10.4
Cm-243	7.1	12.3
Cm-244	7.1	10.4
Cm-245	7.5	11.8
Cm-246	7.2	10.8
Cm-247	7.4	10.4
Cm-248	7.2	11.3
Bk-249	7.4	8.6
Cf-249	7.4	10.1
Cf-250	7.6	10.0
Cf-251	7.4	10.1
Cf-252	7.5	8.8
Mo	3.8	3.5
O	2.2	4.1
C	6.1	7.1
Na	3.0	3.8
Cr	3.4	4.4
Fe	3.2	4.2
Ni	3.1	6.4
Mn	3.5	5.7

TABLE 6. Scattering Microscopic Cross Sections for Possible Actinide Burning Reactor Materials

Nuclide	Microscopic Cross Sections "Fast"	"Slow"
U-234	6.6	10.3
U-235	6.0	8.1
U-236	6.8	10.2
U-238	7.0	10.0
Np-237	5.7	8.7
Pu-238	4.9	8.6
Pu-239	5.6	7.9
Pu-240	5.7	9.4
Pu-241	5.9	9.1
Pu-242	5.8	9.3
Am-241	5.8	9.4
Am-242m	5.8	8.8
Am-243	6.1	10.4
Cm-242	5.2	9.8
Cm-243	4.9	10.3
Cm-244	5.1	9.8
Cm-245	5.5	9.6
Cm-246	5.6	10.7
Cm-247	5.3	8.3
Cm-248	5.6	11.1
Bk-249	6.4	8.5
Cf-249	5.6	8.5
Cf-250	5.5	8.3
Cf-251	5.6	8.1
Cf-252	5.3	8.4
Mo	5.4	8.5
O	3.8	3.6
C	2.2	4.1
Na	3.0	3.8
Cr	3.4	4.4
Fe	3.2	4.2
Ni	3.1	6.4
Mn	3.5	5.2

TABLE 7. Fission Microscopic Cross Sections for Possible Actinide Burning Reactor Materials

Nuclide	Microscopic Cross Sections "Fast"	"Slow"
U-234	1.29	0.17
U-235	1.22	1.33
U-236	0.65	0.00
U-238	0.34	0.00
Np-237	1.51	0.13
Pu-238	2.23	0.92
Pu-239	1.82	1.53
Pu-240	1.49	0.15
Pu-241	1.63	1.74
Pu-242	1.37	0.09
Am-241	1.39	0.06
Am-242m	2.12	2.65
Am-243	1.34	0.03
Cm-242	1.83	0.15
Cm-243	2.15	1.95
Cm-244	1.83	0.15
Cm-245	1.96	2.07
Cm-246	1.56	0.03
Cm-247	2.12	1.96
Cm-248	2.08	1.92
Bk-249	0.99	0.01
Cf-249	1.68	1.87
Cf-250	1.04	1.51
Cf-251	1.68	1.87
Cf-252	2.15	0.25
Mo	0.0	0.0
O	0.0	0.0
C	0.0	0.0
Na	0.0	0.0
Cr	0.0	0.0
Fe	0.0	0.0
Ni	0.0	0.0
Mn	0.0	0.0

TABLE 8. Capture Microscopic Cross Sections for Possible Actinide Burning Reactor Materials

Nuclide	Microscopic Cross Sections "Fast"	"Slow"
U-234	0.141	0.296
U-235	0.061	0.275
U-236	0.141	0.296
U-238	0.057	0.135
Np-237	0.066	0.670
Pu-238	0.024	0.132
Pu-239	0.011	0.182
Pu-240	0.057	0.193
Pu-241	0.089	0.213
Pu-242	0.037	0.133
Am-241	0.117	0.940
Am-242m	0.074	0.197
Am-243	0.029	0.304
Cm-242	0.062	0.414
Cm-243	0.005	0.071
Cm-244	0.062	0.414
Cm-245	0.018	0.162
Cm-246	0.028	0.105
Cm-247	0.018	0.161
Cm-248	0.036	0.088
Bk-249	0.055	0.142
Cf-249	0.060	0.142
Cf-250	0.060	0.142
Cf-251	0.060	0.142
Cf-252	0.018	0.147
Mo	0.012	0.059
O	0.009	0.0
C	0.001	0.0
Na	0.002	0.001
Cr	0.006	0.007
Fe	0.008	0.015
Ni	0.059	0.013
Mn	0.004	0.013

TABLE 9. Fission-to-Capture Cross Section Ratio for Possible Actinide Burning Reactor Materials

Nuclide	Fission-to-Capture Ratio "Fast"	"Slow"
U-234	9.1	0.6
U-235	20.0	4.8
U-236	4.6	0.0
U-238	6.0	0.0
Np-237	22.8	0.2
Pu-238	92.7	7.0
Pu-239	159.5	8.4
Pu-240	26.2	0.8
Pu-241	18.4	8.2
Pu-242	37.1	0.7
Am-241	11.9	0.1
Am-242m	28.5	13.4
Am-243	45.7	0.1
Cm-242	29.8	0.4
Cm-243	421.9	27.4
Cm-244	29.8	0.4
Cm-245	108.3	12.7
Cm-246	56.5	0.3
Cm-247	117.7	11.9
Cm-248	43.7	0.7
Bk-249	16.6	0.1
Cf-249	28.3	13.1
Cf-250	34.3	10.6
Cf-251	28.3	13.1
Cf-252	121.2	1.7
Mo	0.0	0.0
O	0.0	0.0
C	0.0	0.0
Na	0.0	0.0
Cr	0.0	0.0
Fe	0.0	0.0
Ni	0.0	0.0
Mn	0.0	0.0

The sensitivity of the fission-to-capture ratio to the neutron energy can hardly be felt by comparing the "fast" and "slow" values. In Figures B1, B2, and B3 (in appendix B) the fission-to-capture ratio as a function of energy for Am-241 and Am-243 is typical of the actinides that will not fission in a thermal flux (Np-237, Cm-242, Cm-244, Cm-246, Bk-249, Cf-250, Cf-252), while the fission-to-capture ratio for Am-242m is typical of the actinides that do fission in a thermal flux (Cm-243, Cm-245, Cm-247, Cf-249, and Cf-251). The sharp drop in the fission-to-capture ratio of Am-241 and Am-243 clearly demonstrates the need to keep the spectrum in an ABR "fast" in order to effectively burn actinides.

The cross section used in this study were provided by R. E. Schenter. The cross section set was derived from a preliminary version of ENDF/B-V cross section data and collapsed to a 29 group set of cross sections in a Bondarenko format[24]. The cross sections set used in this study have been described more generally in appendices B and D and in references 26 and 27.

The NCINR8 computer code used in this study is described in appendix A and in reference 26. Bench mark data for the NCINR8 computer code are presented in appendix D and in reference 26. The bench mark data indicates that NCINR8 calculates reasonable critical masses. The NCINR8 code calculated a bare critical mass of 52.2 kilograms for U-235, while the reported value is 49.0 kilograms[25]. The code calculated a bare metal critical mass of 12.3 kilograms for Pu-239, while the reported value is 10.0 kilograms[25]. The reasonable critical mass indicates that the spectra calculated in NCINR8 and the cross sections used for this study will be adequate for comparative studies.

4.0 Actinide Burner Reactor Spectra

The success of an ABR will depend on the reactor having a sufficiently hard (high energy) neutron spectra to cause fissions in all of the actinides. The neutron energy sensitivity of the actinide fission cross section shown in chapter 2, requires that careful attention be paid to the effect of all the aspects in an ABR design on the neutron spectrum. Any fuel used will have to have a neutron multiplication factor of greater than 1 or be driven by a more reactive fuel. The reactivity of an actinide fuel is strongly dependent of the fission-to-capture ratio that was shown in Chapter 2 to vary dramatically with the neutron spectra. Presented in this chapter will be a qualitative and quantitative description of how various ABR designs affect the neutron spectra, along with a description of the effect of spectra on reactivity. Figures 1 through 4 visually present the effect of reactor fuel type on neutron spectra. These figures represent ACT6X9 (LWR fuel with 99.9% of the Pu and 99.9999% of the U removed) in all combinations of fuel type (pure metal, 50 weight percent Molybdenum metal, oxide, carbide) and in an infinite array lattice EBR-II configuration, which was described earlier. The infinite neutron spectra for the LMFBR and GCFR rod

configurations are very similar to the EBR-II spectra and are shown in appendix E. Figures 1 through 4 and the figures in appendix E have been scaled so that the energy multiplied by flux is integrated over log energy such that the area is 1. Energy multiplied by flux has been used on the abscissa. Therefore, the area over an energy interval corresponds to the total flux in that energy interval.

The neutron spectra for ACT6X9 metal in an EBR-II configuration is shown in figure 1. The ACT6X9 metal fueled reactors have the hardest spectrum of all the fuel types because there are no extraneous materials to slow down the neutrons. The spectra for the actinide metal (while the hardest of all spectra) still has a peak neutron flux at a neutron energy of 300 keV to 400 keV. The fission-to-capture ratios of Am-241 and Am-243 for neutrons with an energy of 300 keV to 400 keV is 0.1 to 0.2 (shown in chapter 3), which means that a significant number of the neutrons end up causing fission in the plutonium.

The neutron spectra for ACT6X9 molybdenum (50-50 mix by volume) in an EBR-II configuration is shown in figure 2. The molybdenum fuel is visibly softer and shifted to lower energies, than the all metal fuel. In the molybdenum fuel

the peak neutron energy flux occurs at energies between 200 to 400 keV. While the peak flux in the molybdenum fuel has about the same peak neutron energy as the all actinide metal fuel, substantially more of the neutrons in the molybdenum fuel are at lower energies; which will result in fewer fissions in Am-241 or Am-243 in the molybdenum fueled configurations.

The neutron spectra for ACT6X9 oxide fuel in an EBR-II configuration is shown in figure 3. The oxide fuel has a large dip in the 400 to 500 keV range due to a large oxygen inelastic scattering resonance (14 barns) at 442 keV. This scattering resonance results in the peak neutron flux energy occurring at 200 to 300 keV. More importantly the inelastic scattering in oxygen moves neutrons from energies that could cause actinide fissions to energies that most likely will not cause actinide fissions. The oxide fuel obviously is not acceptable for an Actinide Burning Reactor.

The neutron spectra for ACT6X9 carbide fuel in an EBR-II configuration is shown in figure 4. The carbide fuel definitely has a harder (higher neutron energy) spectra than the oxide or molybdenum fuels but appears to have a

softer spectrum than the all metal fuel does. The peak neutron flux occurs at an energy of 300 to 400 kev, which is the same energy as the all metal fuel cases. However, a visual comparison indicates that an all actinide metal fuel would be better for a ABR than any other fuel type.

Figures 1 through 4 give a feel for the effect of fuel material on the neutron spectra, however, a quantitative description of the neutron spectra would be more useful in examining the various spectra. Using a single number, the average neutron energy was chosen to ease comparison of the neutron spectra. The average neutron energy was used since it provided a measure of the neutron energy for a specific reactor and fuel combination. Table 10 has the average neutron energies for both ACT6X9 and ACT3X9 fuels in various reactor configurations.

FIGURE 1. ACT6X9 Metal Spectrum in an Infinite EBR-II Configuration

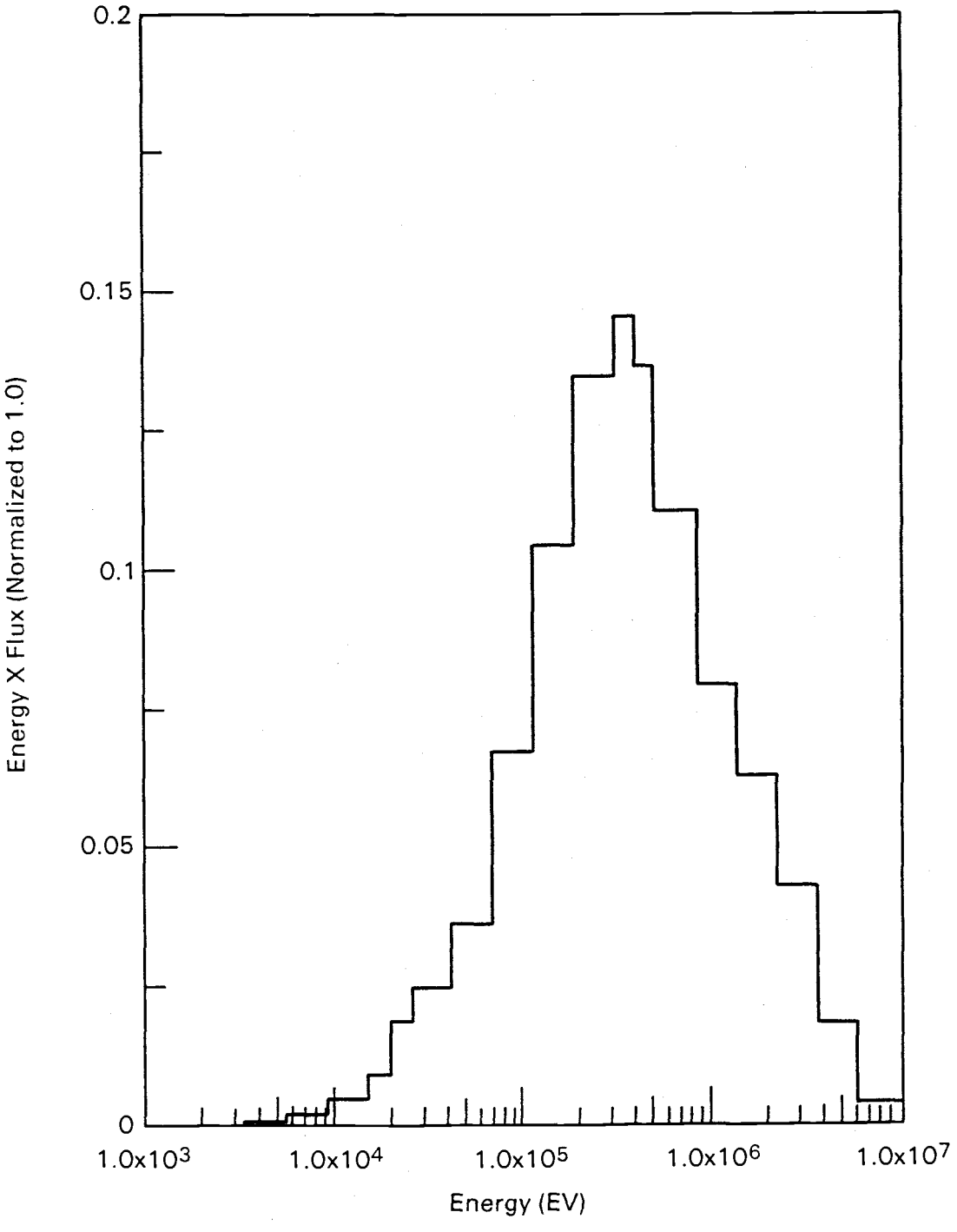


FIGURE 2. ACT6X9 50 wt % Molybdenum Spectrum in an Infinite EBR-II Configuration

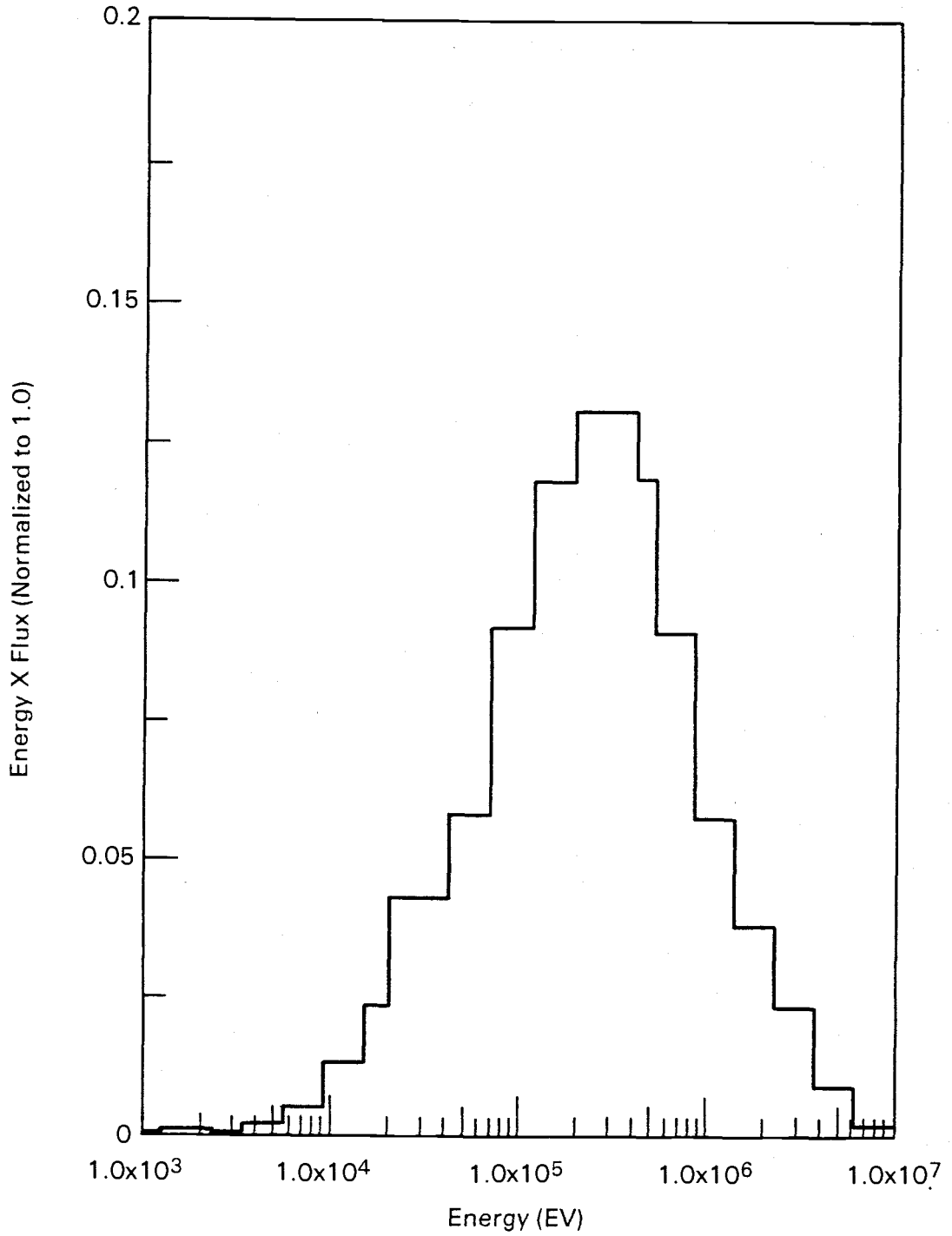


FIGURE 3. ACT6X9 Oxide Spectrum in an Infinite EBR-II Configuration

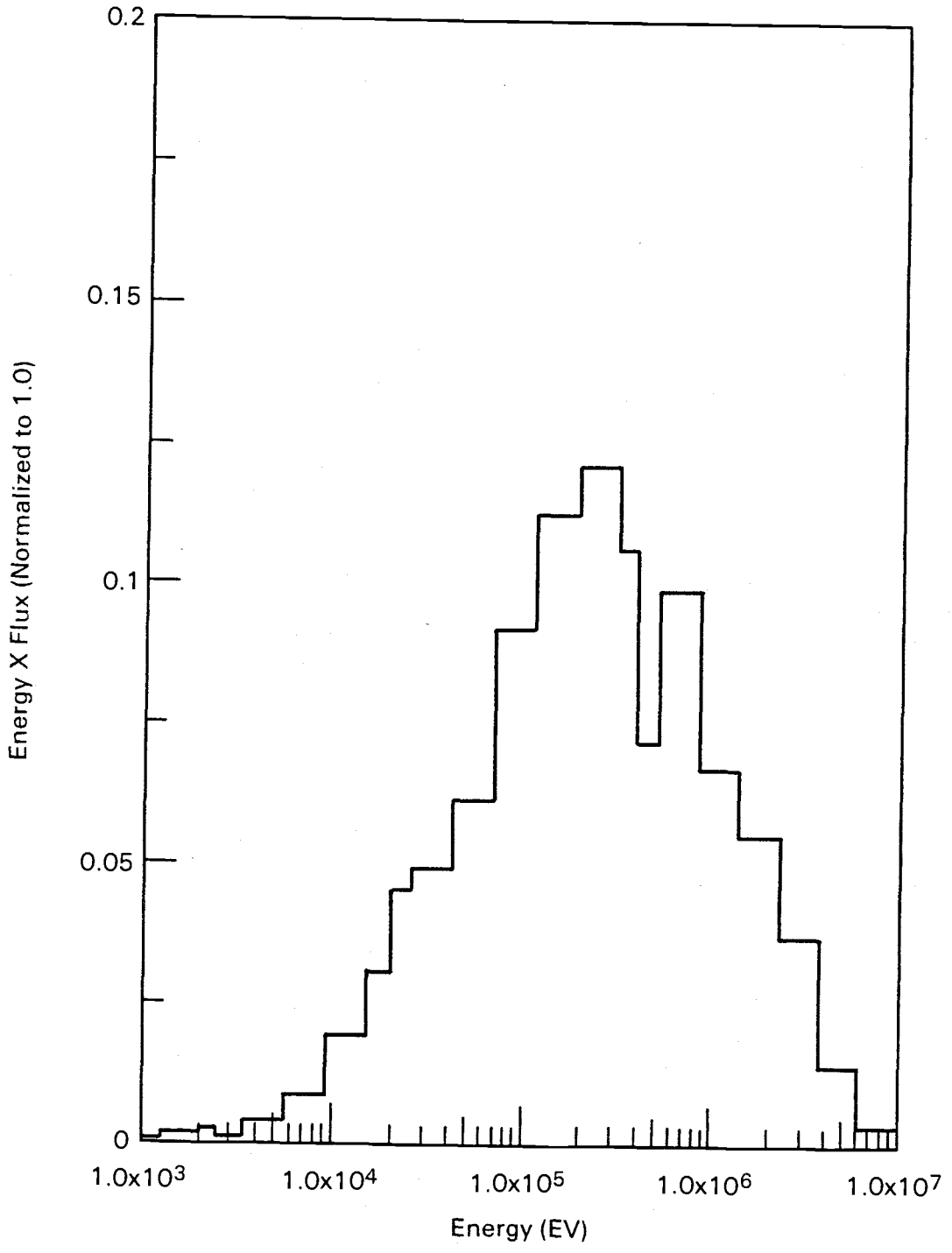


FIGURE 4. ACT6X9 Carbide Spectrum in an Infinite EBR-II Configuration

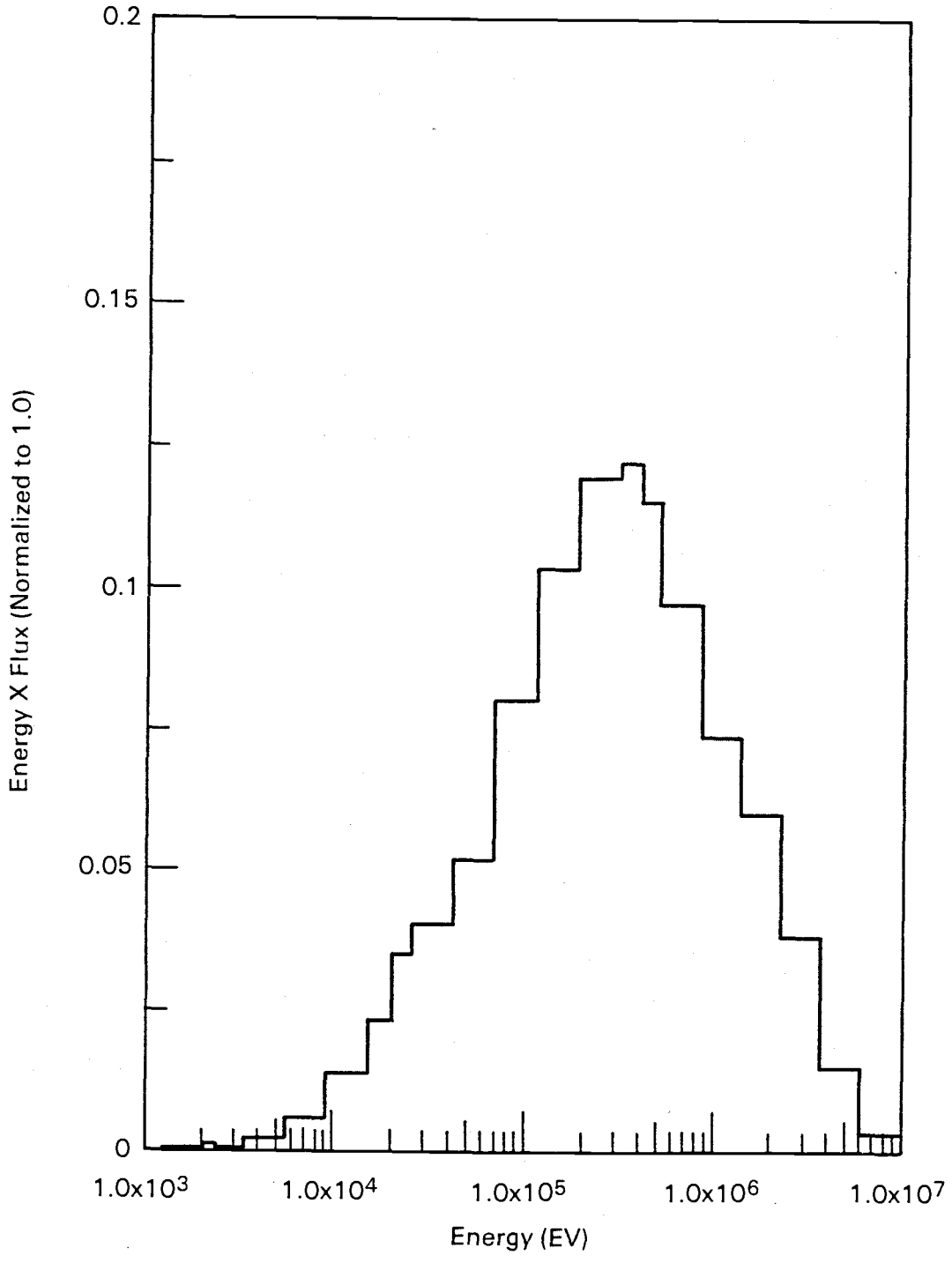


TABLE 10. Average Neutron Energy for Possible Actinide Burning Reactor Configurations

Configuration	Fuel Type	ACT6X9 Average Neutron Energy (KeV)	ACT3X9 Average Neutron Energy (KeV)
Infinite EBR-II	Metal	348	242
Infinite LMFBR		351	242
Infinite GCFR		376	267
Infinite EBR-II	50 Wt %	219	160
Infinite LMFBR	Molybdenum	219	159
Infinite GCFR		246	183
Infinite EBR-II	Oxide	222	138
Infinite GCFR		224	138
Infinite LMFBR		243	153
Infinite EBR-II	Carbide	263	180
Infinite LMFBR		265	180
Infinite GCFR		286	199

The average neutron energies shown in Table 10 show the average neutron energy of each neutron spectra for each fuel and reactor type. The relative ranking of each fuel type is readily apparent by examining table 10. By considering only one reactor's configuration, i.e. infinite EBR-II, the hardest to softest spectrum (highest average energy to lowest) are as follows: ACT6X9 metal, ACT6X9 Carbide ACT3X9 metal, ACT6X9 oxide, ACT6X9 molybdenum, ACT3X9 carbide, ACT3X9 molybdenum, and ACT3X9 oxide. The

ACT6X9 oxide and ACT6X9 molybdenum have nearly the same average neutron energy and their order would change if the infinite GCFR lattice was used to rank the fuel types.

Using the average neutron energy makes it possible to identify the effect of each fuel and reactor design change on the spectra. The use of a sodium coolant instead of helium would reduce the average neutron energy by 25 Kev. The use of a carbide fuel instead of metal would reduce the average neutron energy by 90 Kev, while using an oxide or molybdenum fuel instead of a metal fuel would reduce the average neutron energy by 120 Kev. The use of the ACT3X9 fuel instead of an ACT6X9 fuel would reduce the average neutron energy by about 80 Kev. These changes in the reactor fuel and design are nearly independent from each other and are additive. The change of a gas cooled ACT6X9 metal fueled design to an EBR-II ACT3X9 oxide fuel reactor should reduce the average neutron energy by 225 Kev while the actual calculated reduction in Table 10 is 238 Kev. When the average neutron energy change is laid out in this fashion it is easy to see how each material in the reactor would affect the neutron spectra in the Actinide Burning Reactor.

For any reactor to function the neutron multiplication factor (keff) must be equal to or greater than 1.0 (i.e. the reactor must be critical). In the case of an ABR, if a particular reactor and fuel type are not critical, a driver fuel region a region high in fissile material (plutonium or U-235) can be added until the system is critical or fissile material can be added homogeneously to the fuel. However, this would reduce the effectiveness of the ABR since more of the fissions would occur in the fissile material and not in the actinides as the amount of fissile material was increased. The design objective of ABR is to maximize the amount of actinide fissions for a given power level while maintaining a critical reactor.

One measure that can be used to determine if a reactor is critical is the multiplication factor for an infinite reactor lattice (k-infinity). For any reactor to work k-infinity must be greater than 1.0 and for an ABR the size of EBR-II the k-infinity needs to be about 1.4 due to the high neutron leakage of a small reactor core. For those fuel and reactor types that do not have a sufficiently large k-infinity, additional fissile material would be needed to bring the reactor critical, which would reduce the effectiveness of the ABR.

The k-infinity as a function of exposure for ten fuel and reactor types is shown in Tables 11 and 12. Tables 11 and 12 have a fixed 600 kw/L power level over a 3 year exposure. Table 11 has the k-infinities for ACT6X9 fuel in a EBR-II configuration with metal fuel, 50 wt % Molybdenum fuel, carbide fuel, and oxide fuel. Table 11 has ACT6X9 fuel in a GCFR metal fueled configuration. Table 12 has ACT3X9 fuel in the same fuel and reactor configurations as in Table 11. The four fuel types in the EBR-II configuration represent how k-infinity will change for all of the fuel and reactor configurations. The GCFR metal configuration was included since it is the hardest spectrum reactor of all those presented. It is interesting to note that only the ACT6X9 metal fueled reactors have a k-infinity sufficiently large to make the ABR reactor described here critical.

The k-infinity as a function of exposure shown in Table 11 for the ACT6X9 fuel has two interesting features. The first being that only the metal fuel configurations have a k-infinity sufficiently large to make the ABR described here a critical reactor. Second, all of the fuel types have an increasing k-infinity as the exposure increases (opposite of a typical LWR). The increase in k-

infinity is due to the production of more reactive fissile isotopes (Pu-238, Pu-239, Am-242m, Cm-243, ect) from the less reactive fissionable isotopes (i.e. U-238, Np-237, Am-241, Cm-242 etc.). The two largest contributors to the increase in k-infinity are Np-237 being converted to Pu-238 and Am-241 being converted to Am-242m. These cores are very effective at breeding fissile fuel.

The k-infinity as a function of exposure, shown in Table 12 for the ACT3X9, fuel has the same breeding effect as the ACT6X9 fuel. The breeding that occurs in the ACT3X9 fuels is from the conversion of U-238 to Pu-239. While significant breeding occurs in the ACT3X9 fuels, only the metal fuels approach a sufficiently high k-infinity to fuel the ABR presented here, however, none of the ACT3X9 fuels are sufficiently reactive to make an ABR critical at the beginning of exposure. The 50 wt % molybdenum ACT3X9 fuel breeds for the first 720 days, then destruction of fissile material to maintain 600 kw/L overcomes the production of fissile isotopes after 720 days. The ACT3X9 fuels are clearly unacceptable as a ABR fuel unless driver fuel is provided.

TABLE 11. K-infinity of ACT6X9 Fuel as a Function of Time at 600 Kw/l

Exposure (Days)	EBR-II Metal	EBR-II 50 Wt % Molybdenum	EBR-II Oxide	EBR-II Carbide	GCFR Metal
0	1.412	0.803	1.023	1.171	1.486
180	1.431	0.901	1.090	1.214	1.502
360	1.450	0.970	1.144	1.251	1.518
540	1.466	1.017	1.186	1.283	1.532
720	1.481	1.047	1.220	1.309	1.544
900	1.494	1.065	1.246	1.330	1.555
1080	1.504	1.072	1.266	1.346	1.564

TABLE 12. K-infinity of ACT3X9 Fuel as a Function of Time at 600 Kw/l

Exposure (Days)	EBR-II Metal	EBR-II 50 Wt % Molybdenum	EBR-II Oxide	EBR-II Carbide	GCFR Metal
0	0.750	0.474	0.622	0.772	1.022
180	1.047	0.713	0.855	0.931	1.109
360	1.120	0.816	0.981	1.034	1.176
540	1.177	0.866	1.058	1.104	1.228
720	1.221	0.886	1.106	1.154	1.268
900	1.256	0.886	1.134	1.190	1.301
1080	1.283	0.870	1.147	1.214	1.326

5.0 Actinide Burner Reactor Results

In designing an Actinide Burning Reactor (ABR) it is important to determine the actual number of actinides to be fissioned. This chapter considers only Am-241 and Am-243, since americium is considered the most difficult to manage of the actinides in terms of quantity, activity, and hazard. Am-241 and Am-243 are representative of the actinides that do not fission with "slow" neutrons. In this chapter, results of an infinite lattice burn will be presented for all of the fuel and reactor types. In Chapter 3 it was shown that only metal ACT6X9 will work in an ABR. Results will be presented for a small region of actinide fuel driven by pseudo EBR-II fuel.

For the purposes of this paper the results will be presented as the amount of Am-241 and Am-243 fissioned in a three year exposure. All of the reactor and fuel configurations will be compared against the ACT6X9 metal fuel in an EBR-II reactor lattice. The results for the driven assemblies assume a constant reactor power of 120 Mw. While the results for the infinite lattice runs assume a constant average power of 600 kw/L.

The flux required to maintain an infinite lattice of 600 kw/L is a function of the fission cross section. The beginning of exposure, flux and average neutron energy are shown in Table 13 for each of the fuel and reactor configurations. In Table 13 when the average neutron energy drops, the flux increases, the 50 wt % molybdenum being the only exception. (The 50 wt % molybdenum fuel has significantly less actinide material per unit volume than the other fuel types and therefore has to have a larger flux to maintain 600 kw/L). The high fluxes and low average neutron energy of the oxide and molybdenum fuels has substantial neutron absorption in Am-241 and Am-243.

Neutron absorption of Am-241 or Am-243 as a function of the initial Am-241 and Am-243 in the fuel and reactor types is shown in Table 14. By comparing Tables 13 and 14 it is apparent that those designs with highest fluxes (and lowest average neutron energies) had the largest fraction of Am-241 and Am-242 absorptions. However, the only absorptions that count are those that result in a fission, since Am-241 and Am-243 capturing a neutron produces another actinide.

TABLE 13. Flux and Average Neutron Energy for Infinite Medium Actinide Burnup

Actinide Configuration	ACT6X9 Flux (neutrons cm*cm*se)	ACT6X9 Average Neutron Energy (kev)	ACT3X9 Flux (neutrons cm*cm*sec)	ACT3X9 Average Neutron Energy (kev)
EBR-II metal	2.15E+15	350	5.21E+15	242
LMFBR	2.20E+15	351	5.33E+15	242
GCFR	2.04E+15	376	4.95E+15	267
EBR-II 50 Wt %	8.24E+15	219	1.92E+16	160
LMFBR Molybdenum	8.45E+15	219	1.97E+16	159
GCFR	7.73E+15	246	1.84E+16	183
EBR-II oxide	4.84E+15	222	1.22E+16	128
LMFBR	4.95E+15	224	1.24E+16	138
GCFR	4.63E+15	243	1.18E+16	153
EBR-II carbide	3.59E+15	263	8.10E+15	180
LMFBR	3.67E+15	265	8.27E+15	180
GCFR	3.42E+15	286	7.79E+15	199

TABLE 14. Fraction of Initial Am-241 and Am-243 Number Density Consumed by Neutron Absorption after 3 Years at 600 Kw/l

Actinide Configuration		ACT6X9 Am-241	ACT6X9 Am-243	ACT3X9 Am-241	ACT3X9 Am-243
EBR-II	metal	22.9%	13.6%	42.5%	25.8%
LMFBR		23.5%	14.0%	43.3%	26.5%
GCFR		21.9%	12.9%	40.8%	24.4%
EBR-II	50 Wt %	60.9%	39.7%	81.7%	60.5%
LMFBR	Molybdenum	62.0%	40.6%	82.6%	61.7%
GCFR		58.4%	37.0%	80.2%	57.6%
EBR-II	oxide	46.7%	30.6%	73.0%	54.3%
LMFBR		47.6%	31.4%	73.9%	55.3%
GCFR		44.8%	28.9%	71.4%	51.8%
EBR-II	carbide	36.8%	23.2%	58.4%	39.5%
LMFBR		35.2%	22.0%	59.3%	40.3%
GCFR		35.2%	22.0%	56.6%	37.5%

The fraction of Am-241 or Am-243 neutron absorptions that fission is strongly related to the energy of the neutrons. The Am-241 and Am-243 neutron spectra average fission-to-absorption ratio for each fuel and reactor configuration is shown in Table 15. Examination of Tables 13 and 15 will show that the higher the average neutron energy, the larger the fission-to-absorption ratio. The only exception to this correlation is in the case of oxide fuel. The discrepancy in oxide fuel is due to the large oxygen inelastic scattering resonance at 442 kev. This

resonance moves neutrons (which are nearly too soft to fission Am-241 and Am-243) to a much lower neutron energy, greatly affecting average neutron energy. However, since the neutrons entering the resonance are already at the fission threshold of Am-241 or Am-243, the large drop in neutron energy does not correspond in a large change in fission cross section. For the purposes of this study it was assumed that the fission-to-absorption ratio does not change with exposure. By combining the fission-to-absorption ratios with the fraction of neutron absorptions, and the relative initial americium concentration; a consistent measure of the effectiveness of an ABR can be achieved.

TABLE 15. Fission-to-Absorption Ratio of Am-241 and Am-243 in Various Infinite Actinide Configurations

Actinide Configuration		ACT6X9 Am-241	ACT6X9 Am-243	ACT3X9 Am-241	ACT3X9 Am-243
EBR-II	metal	33.8%	53.6%	23.2%	37.4%
LMFBR		34.0%	53.6%	23.3%	37.5%
GCFR		35.8%	56.6%	25.1%	40.7%
EBR-II	50 Wt %	21.0%	33.2%	15.0%	23.1%
LMFBR	Molybdenum	21.0%	33.9%	15.1%	23.2%
GCFR		22.9%	37.4%	16.6%	26.2%
EBR-II	oxide	24.5%	37.6%	15.7%	22.7%
LMFBR		24.8%	37.9%	15.9%	22.9%
GCFR		26.2%	40.5%	17.0%	25.1%
EBR-II	carbide	28.1%	43.5%	19.3%	29.2%
LMFBR		29.9%	46.5%	19.4%	29.4%
GCFR		29.9%	46.5%	20.9%	32.1%

The relative amount of Am-241 or Am-243 fissioned in an infinite lattice for each of the fuel and reactor configurations is a measure of the performance of the ABR designs. The relative amount of Am-241 and Am-243 fissioned, represented as a fraction of the americium in EBR-II ACT6X9 metal fueled reactor, is shown in Table 16. Comparison of the fissions based on a fixed amount of material is shown in Table 16 reveals that the quickest way to convert actinides to fission products is in an ACT6X9

metal fueled reactor (the GCFR being the quickest). Recall from Chapter 4 that the ACT6X9 metal fuel configurations are the only fuels that can be made critical.

To overcome the low reactivity of some of the actinide fuels, a comparison of the relative actinide destruction power was made for driven fuels. The driver reactor was modeled after the EBR-II reactor. The reactor was 40 cm tall and 80 cm in diameter. The outer 35 cm of the reactor had a fuel consisting of 45.85% U-235, 42.49% U-235, and 11.66% molybdenum. Molybdenum was used instead of the EBR-II rare metal mix due to inadequacies in the NCINR8 cross section library. The reactor had a 40 cm tall by 10 cm diameter section where burnup of each of the actinide fuels in EBR-II rods could be calculated. The reactor was run at an average power of 120 Mw or an average power density of 600 kw/L. The calculated fluxes and average neutron energies in the center of the reactor were similar for all of the actinide fuels.

TABLE 16. Fraction of Initial Am-241 and Am-243 Number Density Consumed by Fission in 3 Years at 600 Kw/l Relative to the Initial ACT6X9 EBR-II Metal Number Density in Infinite Lattice Calculations

Actinide Configuration		ACT6X9 Am-241	ACT6X9 Am-243	ACT3X9 Am-241	ACT3X9 Am-243
EBR-II	metal	7.7%	7.3%	4.5%	4.3%
LMFBR		7.8%	7.3%	4.4%	4.3%
GCFR		7.9%	7.4%	4.7%	4.5%
EBR-II	50 Wt %	4.8%	4.9%	2.1%	2.4%
LMFBR	Molybdenum	4.7%	5.0%	2.0%	2.4%
GCFR		5.0%	5.2%	2.3%	2.5%
EBR-II	oxide	6.1%	6.1%	2.8%	2.9%
LMFBR		6.1%	6.1%	2.8%	2.9%
GCFR		6.3%	6.2%	2.9%	3.1%
EBR-II	carbide	6.8%	6.7%	3.3%	3.4%
LMFBR		6.8%	6.7%	3.3%	3.4%
GCFR		7.0%	6.8%	3.5%	3.6%

For a driven ABR at constant power the calculated power density of the actinide region is dependent on the non-U-238 actinide number density. U-238 has a fission barrier of 180 kev and virtually no fission cross section below 800 kev. The beginning of exposure calculated power density, flux and average neutron energy for each of the actinide fuels are shown in Table 17. In Table 17 it is apparent that the ACT3X9 and ACT6X9 have power densities that are proportional to the non-U-238 actinide atom density. The calculated average neutron energies shown in

Table 17 for each of the actinide fuels are significantly higher than the calculated average neutron energies for the infinite lattices due to the large quantity of U-235 in the driver region converting slow neutrons to fission neutrons. The calculated average neutron flux energy shows only a small deviation between the actinide fuels as would be expected for a small actinide region driven by a much larger reactor. The infinite lattice case demonstrated that flux was the single most important factor in determining how much Am-241 or Am-243 was destroyed.

In a driven reactor the amount of Am-241 or Am-243 absorption is independent of the actinide fuel type. The calculated actinide absorption shown in Table 18 has variations comparable to the variations in flux shown in Table 17. This correlation with flux is consistent with the results for the infinite lattice calculations. However, as with the infinite lattice calculations, Am-241 or Am-243 absorptions are not the most useful way to compare different cases.

TABLE 17. Flux and Average Neutron Energy for Various Actinide Configurations in an EBR-II Reactor Operated at an Average Power Density of 600 Kw/l

Actinide Configuration	Power Density (Kw/l)	Flux (neutrons cm*cm*sec)	Average Neutron Energy (kev)
ACT6X9			
EBR-II metal	1120	3.08E+15	485
EBR-II moly	369	2.90E+15	446
EBR-II oxide	538	2.96E+15	425
EBR-II carbide	694	2.98E+15	451
ACT3X9			
EBR-II metal	551	2.94E+15	444
EBR-II moly	188	2.86E+15	427
EBR-II oxide	257	2.90E+15	399
EBR-II carbide	368	2.90E+15	425

TABLE 18. Fraction of Initial Am-241 and Am-243 Number Density Consumed by Neutron Absorption after being Driven for 3 Years in an EBR-II Reactor

Actinide Configuration	ACT6X9 Am-241	ACT6X9 Am-243	ACT3X9 Am-241	ACT3X9 Am-243
EBR-II metal	33.5%	21.8%	32.3%	20.4%
EBR-II moly	31.8%	20.0%	31.5%	19.6%
EBR-II oxide	32.8%	21.0%	31.8%	19.6%
EBR-II carbide	33.0%	21.3%	32.3%	20.5%

The fission-to-absorption ratios of Am-241 and Am-243 in a driven reactor shown in table 19 are nearly independent of the actinide fuel type as expected, due to the small deviations in the average neutron flux energy.

The driven fission-to-absorption ratios are significantly larger than the ratios for the infinite lattice cases as expected by the harder spectrum. The high fission-to-absorption ratios should lead to substantial Am-241 and Am-243 fissioning rates for the driven actinide fuels.

TABLE 19. Fission-to-Absorption Ratio of Am-241 and Am-243 in a Driver EBR-II reactor

Actinide Configuration	ACT6X9 Am-241	ACT6X9 Am-243	ACT3X9 Am-241	ACT3X9 Am-243
EBR-II metal	45.5%	67.7%	42.3%	64.1%
EBR-II moly	42.3%	64.3%	41.0%	62.6%
EBR-II oxide	41.5%	63.2%	39.8%	60.9%
EBR-II carbide	43.3%	64.9%	41.3%	62.4%

The relative amount of Am-241 or Am-243 fissioned in a driven lattice for each of the fuel types is a fair measure of ABR fuel performance. The relative amounts of Am-241 and Am-243 fissioned compared to the amount of each nuclide in an ACT6X9 metal fueled lattice are shown in Table 20. The results shown in Table 20 are similar to the infinite lattice results in Table 14, and clearly indicate that the ACT6X9 metal fuel is most effective for converting actinides to fission products. The driven lattices in all cases were more effective at fissioning Am-241 and Am-243

than the infinite lattices. This result is expected due to the significantly higher average neutron energy in the driven lattice cases. The results in Table 14 and Table 20 indicate that; to fission the most actinides a reactor should use ACT6X9 metal fuel.

TABLE 20. Fraction of Initial Am-241 and Am-243 Number Density Consumed by Fission in 3 Years in a Driver EBR-II Relative to the Initial ACT6X9 EBR-II Metal Number Density in the Driven Lattice

Actinide Configuration		ACT6X9 Am-241	ACT6X9 Am-243	ACT3X9 Am-241	ACT3X9 Am-243
EBR-II	metal	15.2%	14.8%	6.2%	5.9%
EBR-II	moly	5.0%	4.8%	2.3%	2.1%
EBR-II	oxide	7.2%	7.0%	3.0%	2.8%
EBR-II	carbide	9.4%	9.1%	4.0%	3.8%

The results presented in this chapter offer a few simple rules that can be followed to design an ABR. The first rule is to remove as much U-238 as possible (i.e. use ACT6X9 fuel). The second rule is to remove the non-fissionable nuclides from the fuel (i.e. use a metal fuel). The final rule is to remove as much non-fuel material from the core as possible (i.e. use a GCFR). The ultimate

execution of these three rules is a solid metallic actinide reactor (i.e. the design first suggested by Beaman[8]). Note, a close approximation to this can be made with an ACT6X9 metal fueled EBR-II.

6.0 Conclusions

For the United States nuclear industry to revive, the industry will need to solve two problems. The first and most difficult problem will be to lower reactor lead times, which will also lower reactor costs. The second problem that must be solved is to close the back end of the nuclear fuel cycle and dispose of actinide waste nuclides. The closing of the nuclear fuel cycle with reprocessing will occur when political and economic conditions improve so that reprocessing can become a reality. When reprocessing begins, significant quantities of actinide waste will be generated that can be handled by three different methods.

The first method of disposing of actinide wastes is to bury the actinides with the fission products alone. Actinide burial seems to be the method that the nuclear industry has opted for since no other disposal method is currently being studied. Burial will probably be satisfactory for the actinides. The actinide burial site will have to be secure for a longer time than for a fission-product-only burial site.

The second method of disposing of actinide wastes is to fission them in a specifically designed ABR. The ABR in question was shown to be possible with this research. The EBR-II sized reactor described in this paper would burn about 50 kilograms of actinide fuel in a year. However, to burn all of the ACT6X9 material produced in the year 2000, nearly eight thermal gigawatts of ABR would need to be on line, assuming that the ABR could operate at 80% capacity. If ACT3X9 fuel and driver material were used instead of ACT6X9 fuel, three to ten times the power would need to be produced with ABR's to burn all of the actinides. Since the ACT6X9 fuel would be difficult to produce and the use of ACT3X9 fuel would require several times as many ABR's, a new design of reactor that did not produce actinide material would seem to be a better solution.

The third solution to the actinide disposal problem is to design reactors that do not produce actinides (or do not produce significant amounts of actinide waste). This "newly" designed reactor would have to follow the same design philosophy as the design as an ABR reactor. This philosophy is the same used in the early breeder reactors, except that the early breeder reactors included a large

fraction of U-238 to produce Pu-239. These early breeder reactors, due to the high concentrations of fissile materials, had a very high average neutron energy and produced virtually no actinides.

The design philosophy used to build an ABR can lead to a reactor that will reduce the quantities of actinide waste or lead to a reactor that does not produce actinides. Primarily, the objective of an actinide burning reactor is to maintain a hard neutron spectrum, while fissioning the maximum amount of actinides. To achieve this goal, the use of uranium should be minimized, specifically U-238 and U-234, since these nuclides have very high inelastic scattering cross sections that significantly reduce the average neutron energy. Secondly, the use of non-fuel materials (oxygen, molybdenum, carbon, and sodium) in the core must be held to a minimum, since these materials only scatter neutrons and soften the neutron spectrum. Thirdly, the fissile driver material (U-235 or plutonium) should only be used to achieve criticality in the core to maximize the number of actinide fissions at a specific power. Finally, the reactor has to be designed in a fashion that

can be cooled; unlike the solid actinide core suggested by Beaman[21]. The optimum of this design philosophy would be a Gas Cooled Fast Reactor fueled with a metallic actinide fuel with all of the uranium and plutonium removed.

References

1. Spent Fuel and Waste Inventories and Projections, ORO-778, UC-70, August 1980.

2. Blomeke, J. O., C. W. Kee and J. P. Nichols, Projections of Radioactive Wastes to be Generated by the U.S. Power Industry, ORNL-TM-3965, February 1974.

3. Spent Fuel and Radioactive Waste Inventories and Projections as of December 31, 1980, DOE/NE-0017, September 1981.

4. Cohen, B. L., "Environmental Hazards in Radioactive Waste Disposal", Physics Today, pp. 9-13, January 1976.

5. Cohen, B. L., "High Level Radioactive Waste from Light Water Reactors", Reviews of Modern Physics, 49, No. 1, pp. 1-20, 1977.

6. Cohen, B. L., "The Disposal of Radioactive Wastes from Fission Reactors", Scientific American, 236, No. 6, pp. 21-31, June 1977.

7. Schneider, K. J. and A. M. Platt, High Level Radioactive Waste Management Alternatives, BNWL-1900, 1974.

8. Walker, F. W., et. al., Chart of the Nuclides, General Electric Company, Twelfth Edition, 1977.

9. Blomeke, J. O. and A. G. Groff, "An Overall Assessment of Actinide Partitioning and Transmutation", Trans. Am. Nucl. Soc., 34, pp. 412-413 (1980).

10. Finney, B. C. and D. W. Tedder, "Actinide Partitioning Process for Fuel Cycle Wastes", Trans. Am. Nucl. Soc., 34, pp 412-413 (1980).

11. Tedder, D. W. and J. O. Blomeke, Actinide Partitioning and Transmutation Program, Progress report, July 1 - September 30, 1977, ORNL/TM-6174 (February 1978).

12. Clairborne, H. C., Effect of Actinide Removal on the Long-Term Hazard of High-Level Waste, ORNL-TM-4724, (January 1975).

13. Partitioning of Long Lived Nuclides from Radioactive Waste - FY 1975 Annual Report, BNWL-1926, UC-70, April 1976.

14. Beaman, S. L. and E. A. Aitken, "Feasibility Studies of Actinide Recycle in LMFBRs as a Waste Management Alternative", Trans. Am. Nucl. Soc., 23, 547, 1976.

15. Clairborne, H. C., Neutron Induced Transmutation of High Level Wastes", ORNL-TM-3964 (December 1972).

16. "High Level Radioactive Waste Management Alternatives", Prepared by Battelle Northwest Laboratories, WASH-1297 (1974).

17. Jenquin, U. P. and B. R. Leonard, Jr., "Actinide Transmutation in Fusion Reactor Blankets", Trans. Am. Nucl. Soc., 23, 540 (1976).

18. Pigford, T. H. and J. S. Choi, "Actinide Transmutation in Fission Reactors", Trans. Am. Nucl. Soc., 27, 450 (1972).

19. McKay, Alwyn, "Destroying Actinides in Nuclear Reactors", Nucl. Eng. Int. (January 1978).

20. Cooperative Nuclear Data and Methods Development, GEFR-14074-6, UC-79E. April 1977.

21. Beaman, S. L., Actinide Recycle in LMFBRs as a Waste Management Alternative, GEFR-00482, UC-79E, (August 1978).

22. Mann, F. M. and R. E. Schenter, "Actinide Cross Sections Calculations and Evaluations", Trans. Am. Nucl. Soc., 23, 546 (1976).

23. Mann, F. M. and R. E. Schenter, "Calculated Neutron Capture Cross Sections to the Americium Ground and Isomeric States", Nuc. Sci. Eng., 63, pp. 242, (1977)

24. Mann, F. M. and R. E. Schenter, HEDL Evaluation of Actinide Cross-Sections for ENDF/B-V, HEDL-TME-77-54, UN-79d, (August 1977).

25. Clayton, E. D., Anomalies of Criticality, BNWL-SA-4868, Rev. 5, June 1979.

26. Robinson, A. H., "Studies of Actinide Waste Burnup and Use in Fast Spectra", RLO/2227/T32-4, (March 1978)

27. Robinson, A. H., "Studies of Actinide Waste Burnup and Use in Fast Spectra", RLO/2227/T32-6, (July 1978)

APPENDICES

APPENDIX A

Description of the NCINR8 Program

Program and description written by Graeme Shirley

NCINR8 uses multigroup shielded cross section data in the Bondarenko format to perform spectrum, one-dimensional diffusion, and isotopic depletion calculations during burnup and subsequent decay for a fast reactor. Developed for the purpose of analyzing fast reactor designs for destruction of actinide wastes, the code determines inventories on both a mass and hazard basis for as many as two dozen or more actinides during the burn, then follows this inventory during a 1000-year decay.

The sequence of operations in a complete NCINR8 run begins with the reading of the cross section library. Because input and output operations are faster with unformatted data, and because the cross section information may be read several times, all subroutine inputs are unformatted NCINR8 checks for unformatted input, and if that is not available, unformatted files are created. (These may be saved for future calculations involving the

same isotope set, since several CPU seconds may be used in creating the unformatted files.) Regional number densities are determined for each isotope, then zero-leakage fine group fluxes are calculated in each region by sweeping through the groups and iterating on the elastic down-scattering source. Fine group fluxes and shielded cross sections are combined to determine regional broad group cross sections. A one-dimensional diffusion calculation is performed to determine spatial flux distribution for each coarse group. Convergence of the one-dimensional calculation is expedited by the use of Chebyshev extrapolation. A neutron balance is performed to determine leakage from each broad group and region. These leakage terms are applied to a new spectrum calculation to modify collapsed cross sections, and a second one-dimensional diffusion calculation is performed. The sequence of spectrum and one-dimensional calculations is repeated a specified number of times.

For each region, the spectrum output is a set of broad group cross section and the one-dimensional output is a set of volume-weighted fluxes. These cross sections and fluxes are combined in a zero-dimensional burnup calculation which uses a fourth-order Runge-Kutta numerical technique. After

each specified burn increment, spectrum and/or one-dimensional calculations are updated, and the burn continues. At the completion of the burn, a summary, on a total-core basis, is produced, and a thousand year decay of the core actinide inventory is performed, also using a fourth-order Runge-Kutta.

An objective in writing NCINR8 has been to include options to select any of a variety of computation paths while maintaining a simple input format. Thus, some variable act both as input variables and as control variables. (As an example, NSINT(1), the number of spatial intervals in region 1, is an input variable to one-dimensional calculation. However, a zero value acts to bypass one-dimensional calculations and specifies zero-leakage burns.)

Every run begins with creation of unformatted isotope data tapes. If the number of regions (NR) is zero, the run is complete. For NR greater than zero, the variable dimension pointers are calculated and the remaining input variables are read and checked for error conditions. After number densities are calculated, the number of initial spectrum-one-dimensional iterations (NSTART) will, if zero,

cause a decay only case to be run. If NSTART is greater than zero, a spectrum calculation is performed if the number of space intervals in region 1 (NSINT(1)) is zero, the path branches to a zero-leakage (K-infinite) burn option, otherwise the sequence of NSTART spectrum-one-dimensional iterations is performed. After the iterative process (NSINT(1)>0) or the first spectrum (NSINT(1) = 0), a value of zero for the number of burn sequences to be performed (NBURN) causes a branch to a decay option. If NBURN is greater than zero, the core inventory (or density if a k-infinity burn) is determined. If power (or power density for a K-infinity burn) is greater than zero, the burnup (including spectrum and/or one-dimensional updates) is completed. After the burn (Power>0) or inventory calculation (Power = 0). A core inventory summary prior to and after the burn sequence is presented. After this summary, or after the initial spectrum (and if performed, one-dimensional) calculation(s) (NBURN = 0), a decay of the core inventory is performed if the decay control variable (IDECAY) is non-zero. (If NSTART is zero, the decay is performed automatically following the decay option, a new case begins. This case must the same isotope set.)

Because NCINR8 both uses variable dimensioning and uses the same storage locations for different variables in different subroutines, data storage requirements are complicated. The largest storage allocation is, typically, the shielding factor array. Shielding factors are functions of isotope, fine group, number of shielding cross sections (sigma-o's), type of cross section and temperature; since most available cross sections are in one temperature, the temperature dependence has been eliminated in the library and code. Even removing this dependence, however, the shielding factors could still be written as four-dimensional array. For 35 isotopes, 29 groups, 6 sigma-o's and 4 cross sections, 24360 storage locations would be allocated for shielding factors alone. It is apparent that this array size may be reduced considerably by the following:

1) Eliminate shielding factors for the total cross section. The total cross sections are used only for initial estimates of the shielding cross sections.

2) Eliminate dimensioning for fission cross section shielding factors for isotopes with no fission cross section or unshielded cross sections.

3) Allow the option of including some isotopes as infinitely dilute, thus requiring no shielding factors. This is explained in the section on the use of the isotope data library.

4) Only store shielding factors for those groups for which they are provided. (For example, if an isotope has shielding factors for fine groups 2 through 15, the data need only be stored for these 14 shielded groups, rather that the total number of groups.)

The shielding factor array is stored linearly rather than as a four- (five-) dimensional array. This allows counting the number of shielding factors when making the unformatted data tapes and allocating the minimum storage by using this count (MTOPF) in setting variable dimension pointers. Other large arrays are those associated with

inelastic matrices. These would naturally be dimensioned to the number of down-scattering groups for each fine group fine (or broad) group and for each isotope (or region). Here the effect on total is storage is reduced by:

1) Determining the last group for which there is non-zero inelastic down-scattering for any isotope (N1NMGM), then dimensioning only to this level, and

2) Using the same storage locations for inelastic scattering as for shielding factors where possible.

APPENDIX B

Discussion of the Cross Section Data

The cross sections used in NCINR8 were obtained from R.E. Schenter. The data were prepared by him in either 40 or 29 groups from a preliminary version of ENDF/B-V. Since NCINR8 uses only 29 group data, the 40-group data, which was the only data available for some isotopes, was collapsed to 29 groups. Since the upper part of the group structure is identical, only the lower energy groups were combined to obtain the 29-group set. In many cases 29-groups data was provided by R.E. Schenter and no collapsing was needed.

When collapsing was required, the data were combined by simply averaging the values for two or more groups. This was only done at low energies where the flux is extremely low. In the ABR's being studied here, the flux in groups below number 25 is down by a million from the peak value.

Inelastic scatter transfer matrices were provided in the data sets. No collapsing of the data was required. Elastic down-scatter data was also provided. The contribution to slowing down by the elastic scattering is iterated on in NCINR8 in the same manner as FCC-IV (see reference 1).

A more complete discussion of the data provided by R.E. Schenter is given in references 2 and 3.

Plots of the fission-to-capture cross section ratio of Am-241, Am-242m, and Am-243 are given in Figures B1, B2, and B3. These plots show the rather strong fission threshold effect in these isotopes. Because we are trying to eliminate the actinides by fission, rather than capture, it is necessary to have a spectrum peak above about 300 kev.

In Appendix C the results of calculations to verify the cross sections and the NCINR8 code are presented.

FIGURE B1. Am-241 Fission-to-Capture Ratio as a Function of Energy

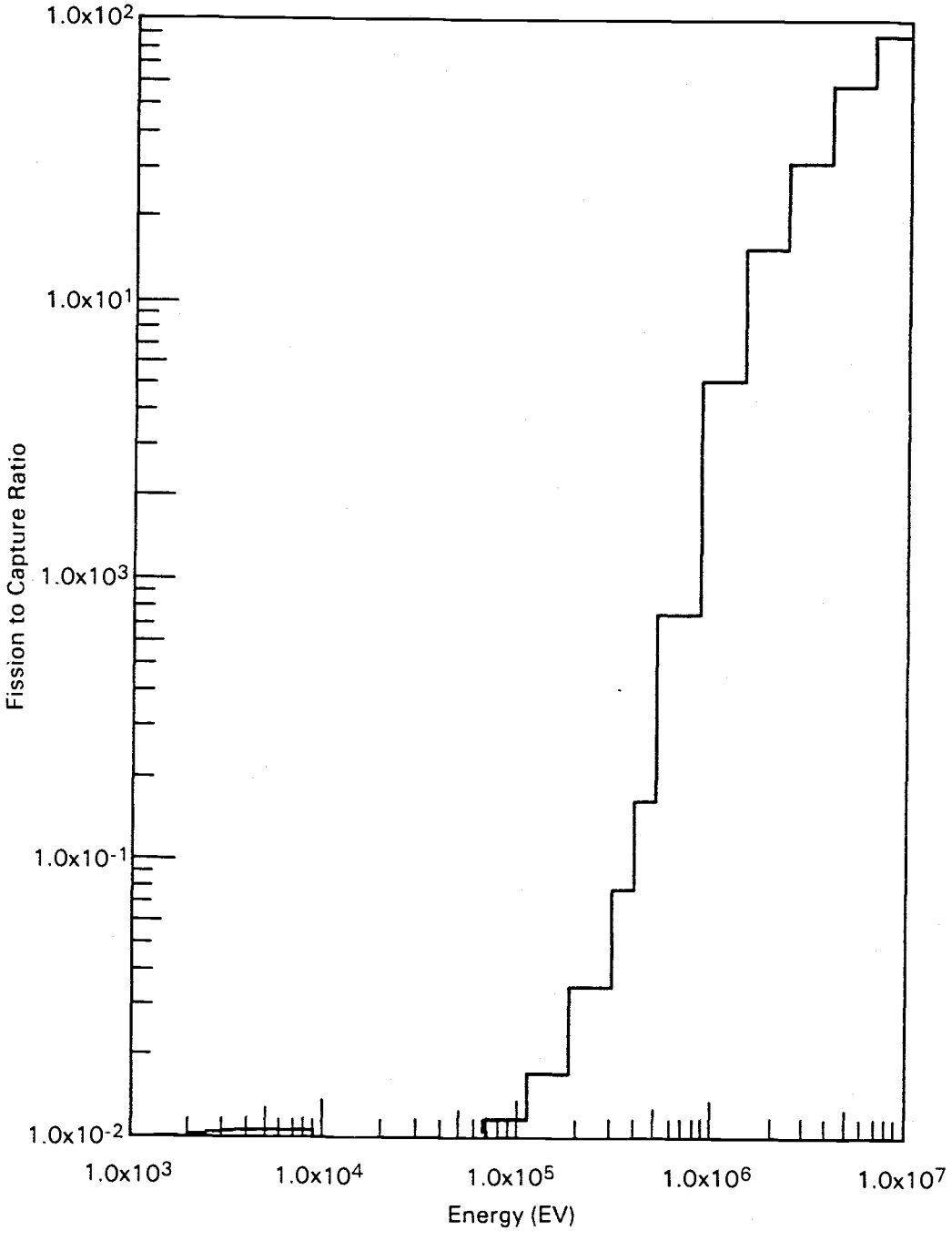


FIGURE B2. Am-242m Fission-to-Capture Ratio as a Function of Energy

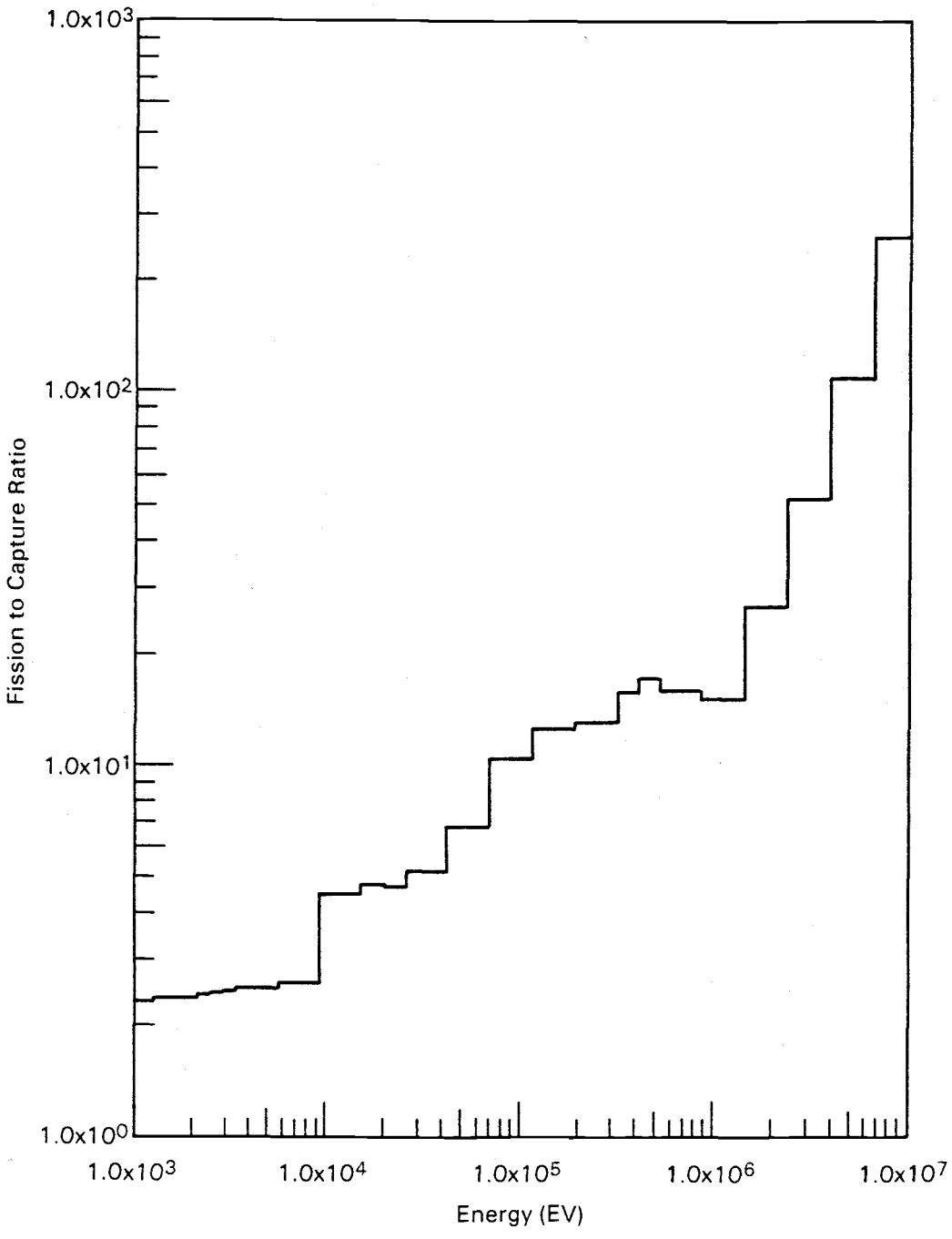
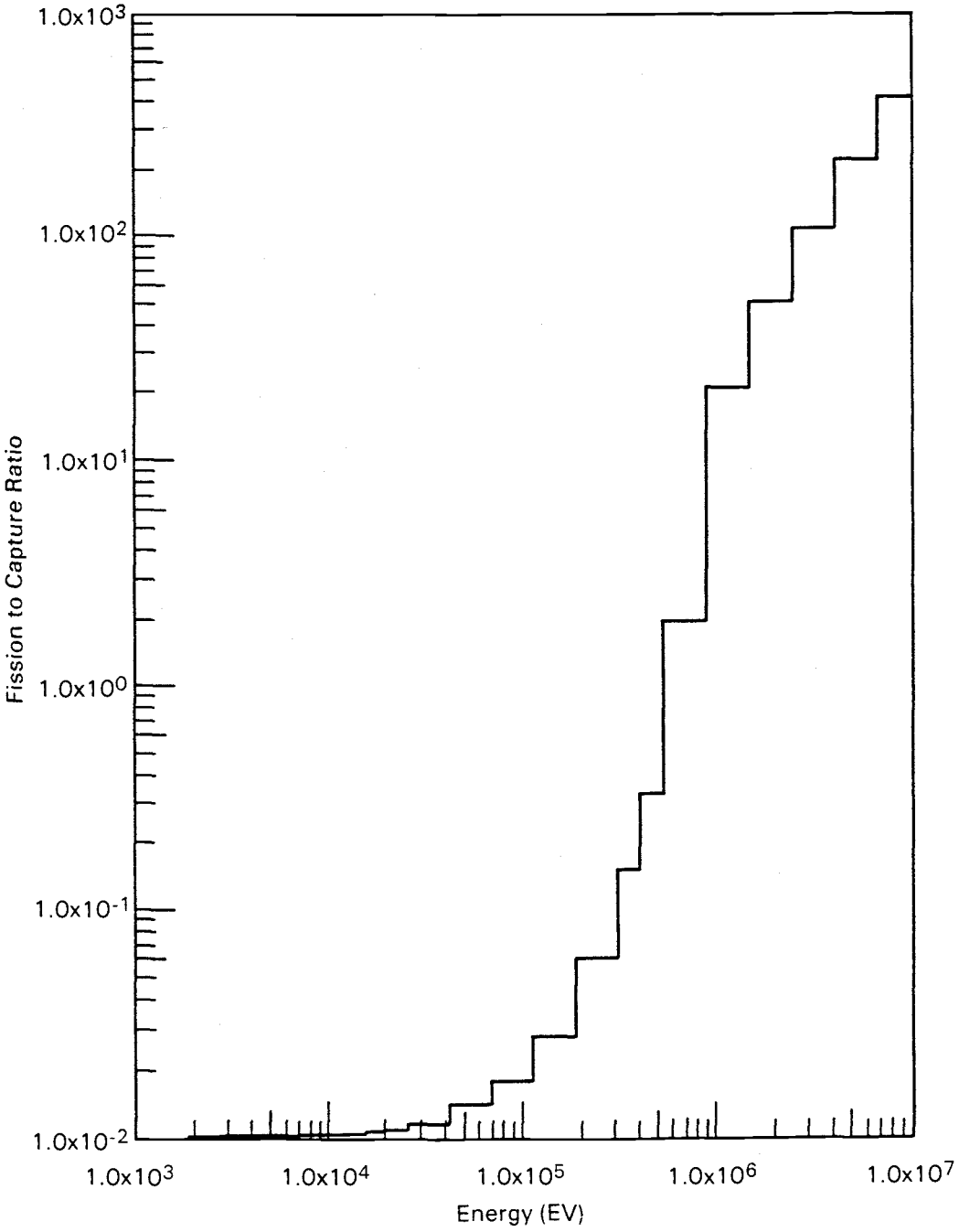


FIGURE B3. Am-243 Fission-to-Capture Ratio as a Function of Energy



APPENDIX C

Verification of the Accuracy of the NCINR8 Code and the Cross Sections

In order to check the accuracy of the computer code and cross sections a series of bare spherical criticals was calculated with NCINR8. The results of these calculations are compared to those of E.D. Clayton in Table C1. For U-235, Np-237, and Pu-239 the agreement is quite good considering that we are using diffusion theory for small metal spheres and out temperature is 300 degrees F. The results for Am-241, Cm-244, and the actinide mixes are probably more accurate than Clayton's since we used a preliminary version of ENDF/B-V for the cross sections. The Am cross section was changed significantly in go in from version IV to version V of ENDF/B. The newer cross sections yield a larger critical mass for Am-241 and Cm-244.

Table C-1

Critical Mass and Radius for Various Actinides

Isotope	O.S.U. Results[1] Radius (cm)	Critical Mass (kg)	BNWL-SA-4868[2] Critical Mass (kg)
U-235	8.712	52.24	49.
Np-237	9.792	76.14	88.
Pu-239	5.318	12.26	10.0
Am-241	14.83	185.1	58.
Cm-244	8.165	30.78	27.7
ACT6X9 [3]	10.77	90.04	
ACT3X9 [4]	25.26	1220.0	

1. Calculated by the NCINR8 diffusion theory code in spherical geometry at 300 degrees F.

2. From Table III of Anomalies of Criticality by E.D. Clayton BNWL-SA-4868 Rev. 5, June 1979

3. See Table 1 for a list of the isotopes present in ACT6X9.

4. See Table 1 for a list of the isotopes present in ACT3X9.

APPENDIX D

Library of Isotopes, Burnup Chains and Alpha Decay Chains in the NCINR8 Code.

TABLE D1. Isotopes and Half-Lifes of Nuclides in the NCINR8 Library

Fuel Materials

	Half-Life (years)		Half-Life (years)
U-234	2.442E+5	Cm-242	4.467E-1
U-235	7.136E+8	Cm-243	2.804E+1
U-236	2.341E+7	Cm-244	1.791E+1
U-238	4.476E+9	Cm-245	7.739E+3
		Cm-246	4.768E+3
		Cm-247	1.451E+7
Np-237	2.142E+6	Cm-248	3.500E+5
Pu-238	8.792E+1		
Pu-239	2.439E+4	Bk-249	8.519E-1
Pu-240	6.542E+3		
Pu-241	1.501E+1	Cf-249	3.522E+2
Pu-242	3.870E+5	Cf-250	1.311E+1
		Cf-251	9.008E+2
Am-241	4.335E+2	Cf-252	2.632E+0
Am-242m	1.521E+2		
Am-243	7.376E+3	Pseudo fission products	

Coolant Materials

Na-23
He-4

Clad Materials

Cr-52
Mn-55
Fe
Ni

Diluent Materials

C-12
O-16
Mo

FIGURE D1. NCINR8 Burnup Chains

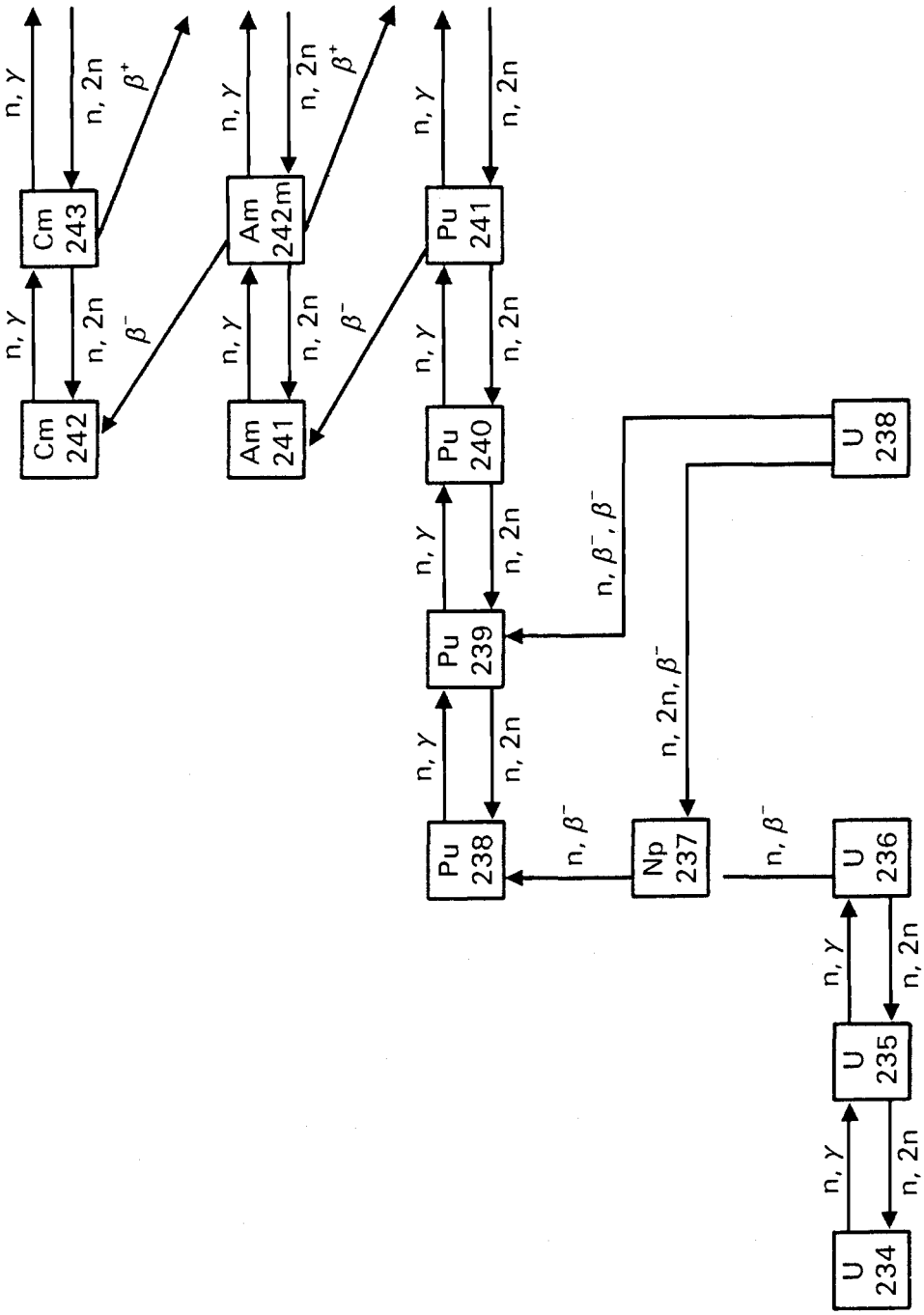


FIGURE D1. NCINR8 Burnup Chains
(continued)

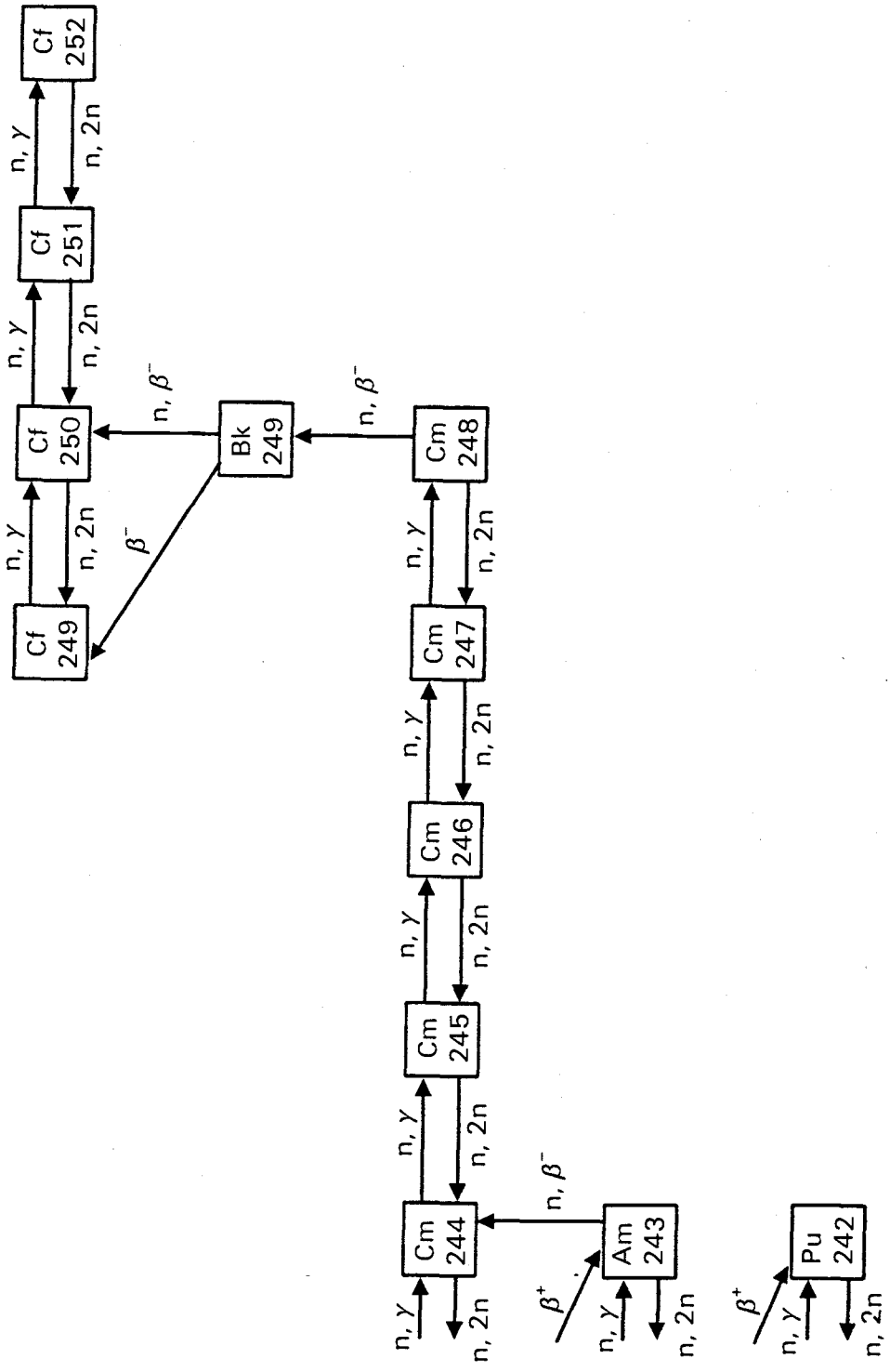


TABLE D2. NCINR8 Alpha Decay Chains

Parent Nuclide	First Decay	Second Decay	Third Decay	Fourth Decay
Cf-252	Cm-248			
Cf-251	Cm-247	Am-243	Pu-239	U-235
Cf-250	Cm-246	Pu-242	U-238	
Cf-249	Cm-245	Pu-241	Np-237	
Bk-249	Cm-245	Pu-241	Np-237	
Cm-247	Am-243	Pu-239	U-235	
Cm-246	Pu-242	U-238		
Cm-245	Pu-241	Np-237		
Cm-244	Pu-240	U-236		
Cm-243	Pu-239	U-235		
Cm-242	Pu-238	U-234		
Am-243	Pu-239	U-235		
Am-242m	Pu-238	U-234		
Am-241	Np-237			
Pu-242	U-238			
Pu-241	Np-237			
Pu-240	U-236			
Pu-239	U-235			
Pu-238	U-234			

APPENDIX E

Infinite Neutron Spectra for LMFBR and GCFR Rod Configurations

Appendix E contains the infinite neutron spectra for the LMFBR and GCFR rod configurations using various ACT6X9 fuel materials. Figures E1 through E4 have the infinite LMFBR configurations for ACT6X9 metal, ACT6X9 molybdenum, ACT6X9 oxide, and ACT6X9 carbide, respectively. Figures E5 through E8 have the infinite GCFR configurations for ACT6X9 metal, ACT6X9 molybdenum, ACT6X9 oxide, and ACT6X9 carbide, respectively.

FIGURE E1. ACT6X9 Metal Spectrum in an Infinite LMFBR Configuration

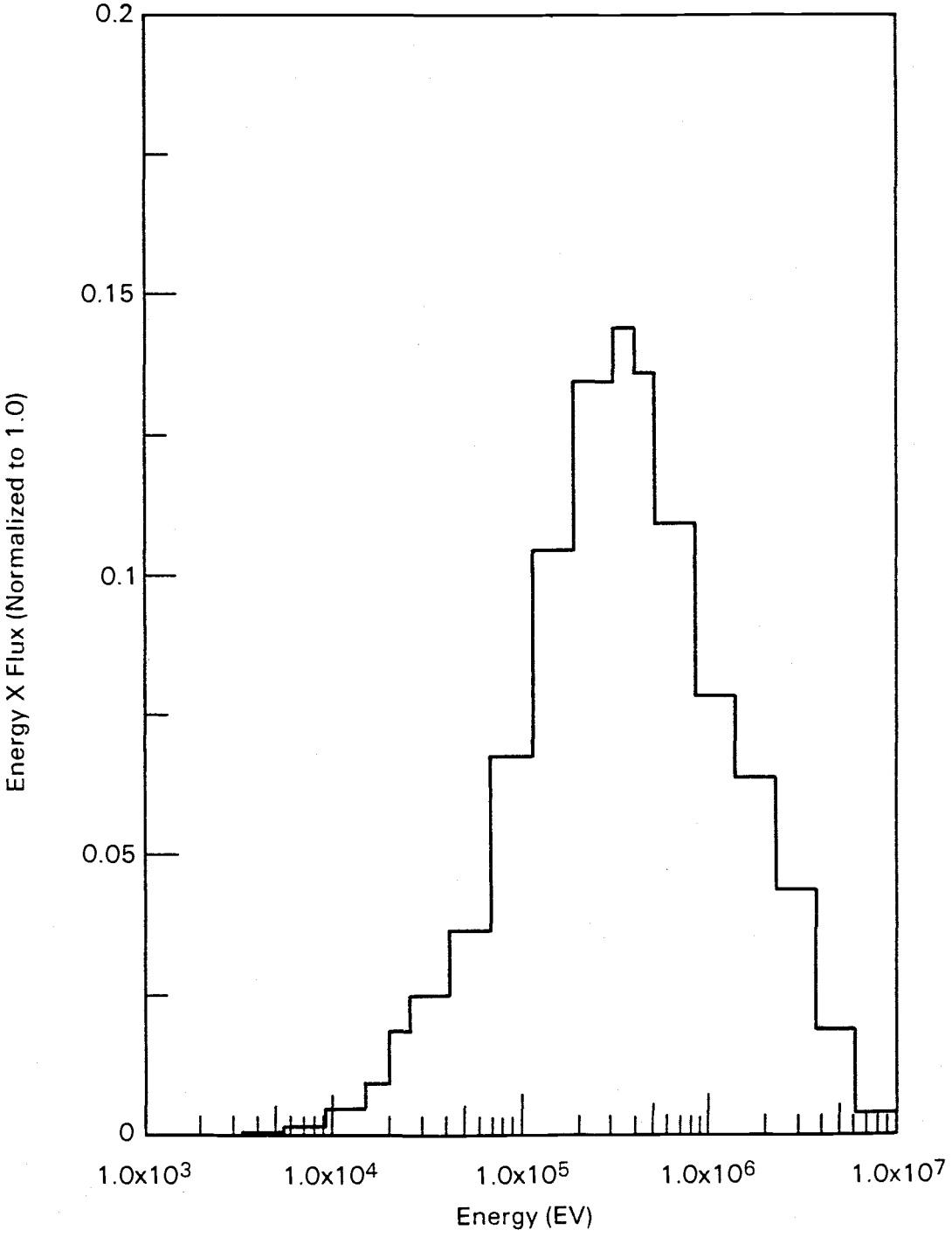


FIGURE E2. ACT6X9 50 wt % Molybdenum Spectrum in an Infinite LMFBR Configuration

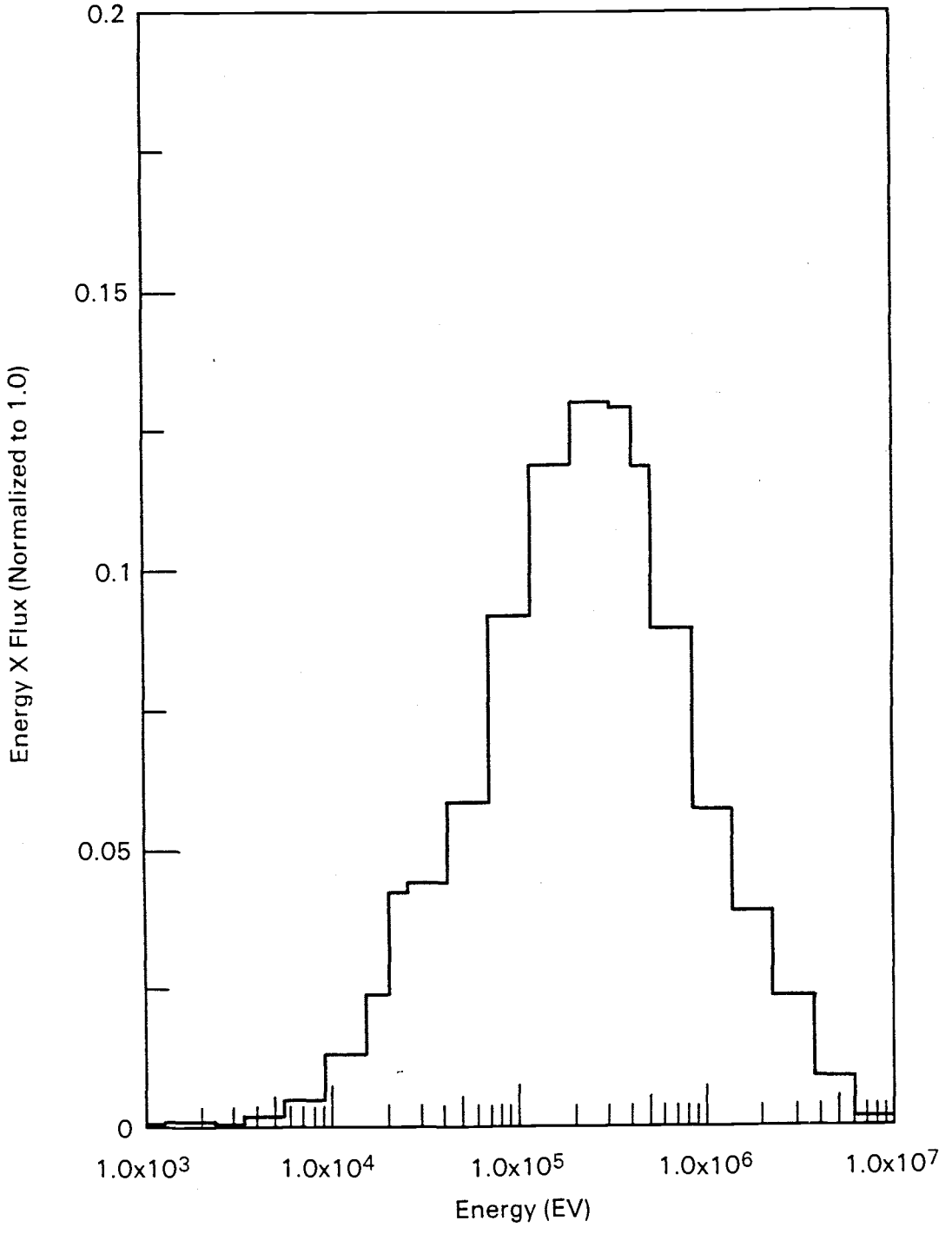


FIGURE E3. ACT6X9 Oxide Spectrum in an Infinite LMFBR Configuration

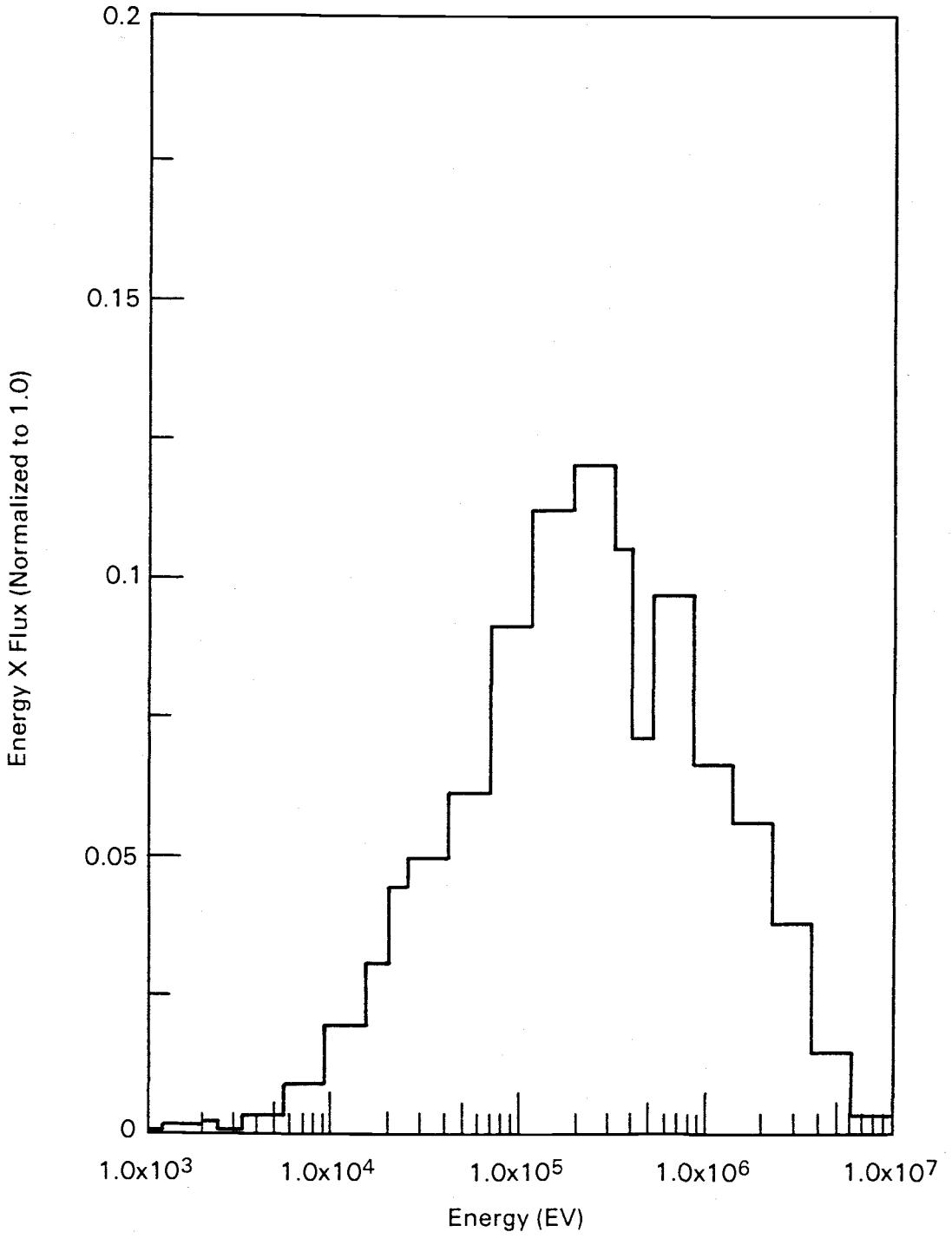


FIGURE E4. ACT6X9 Carbide Spectrum in an Infinite LMFBR Configuration

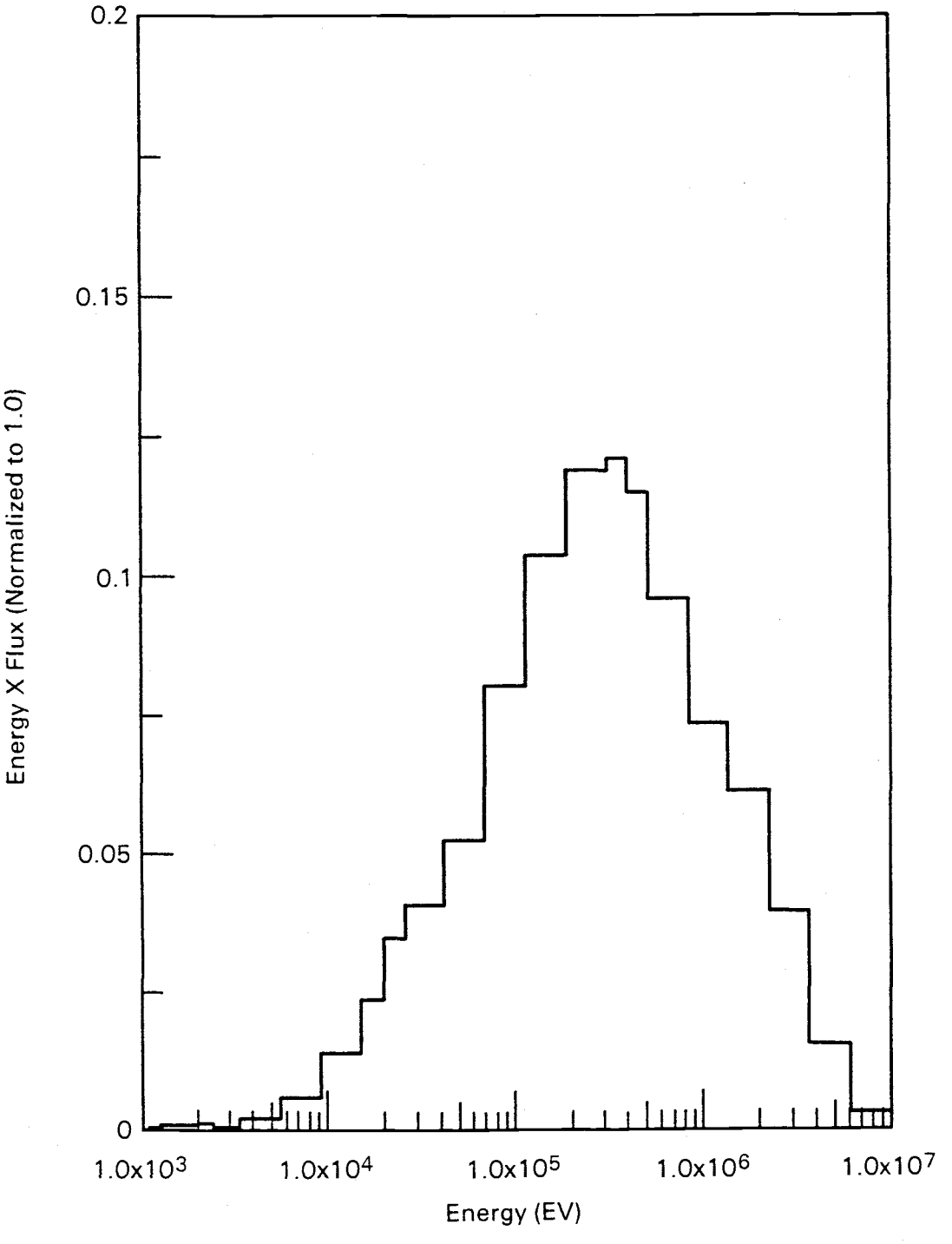


FIGURE E5. ACT6X9 Metal Spectrum in an Infinite GCFR Configuration

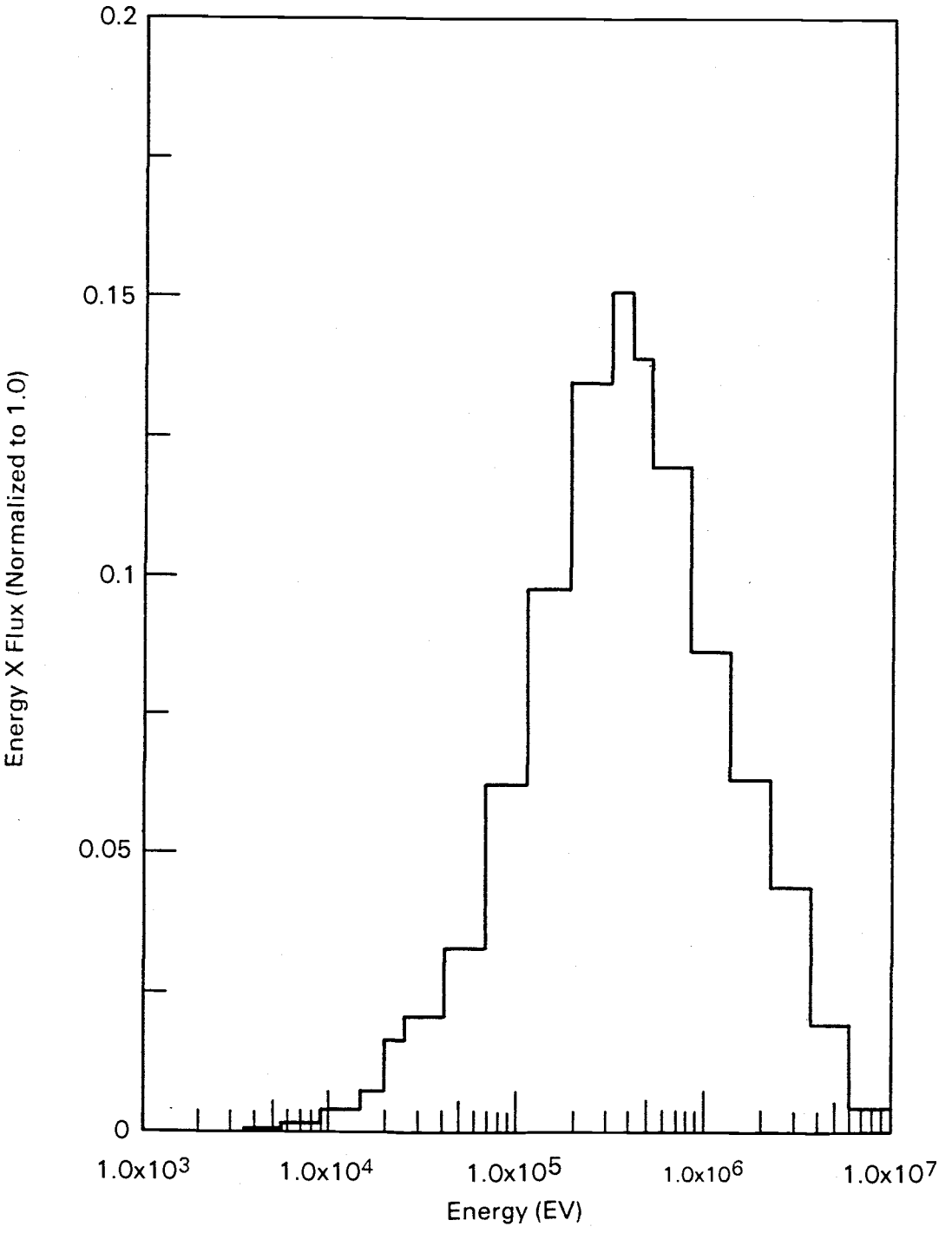


FIGURE E6. ACT6X9 50 wt % Molybdenum Spectrum in an Infinite GCFR Configuration

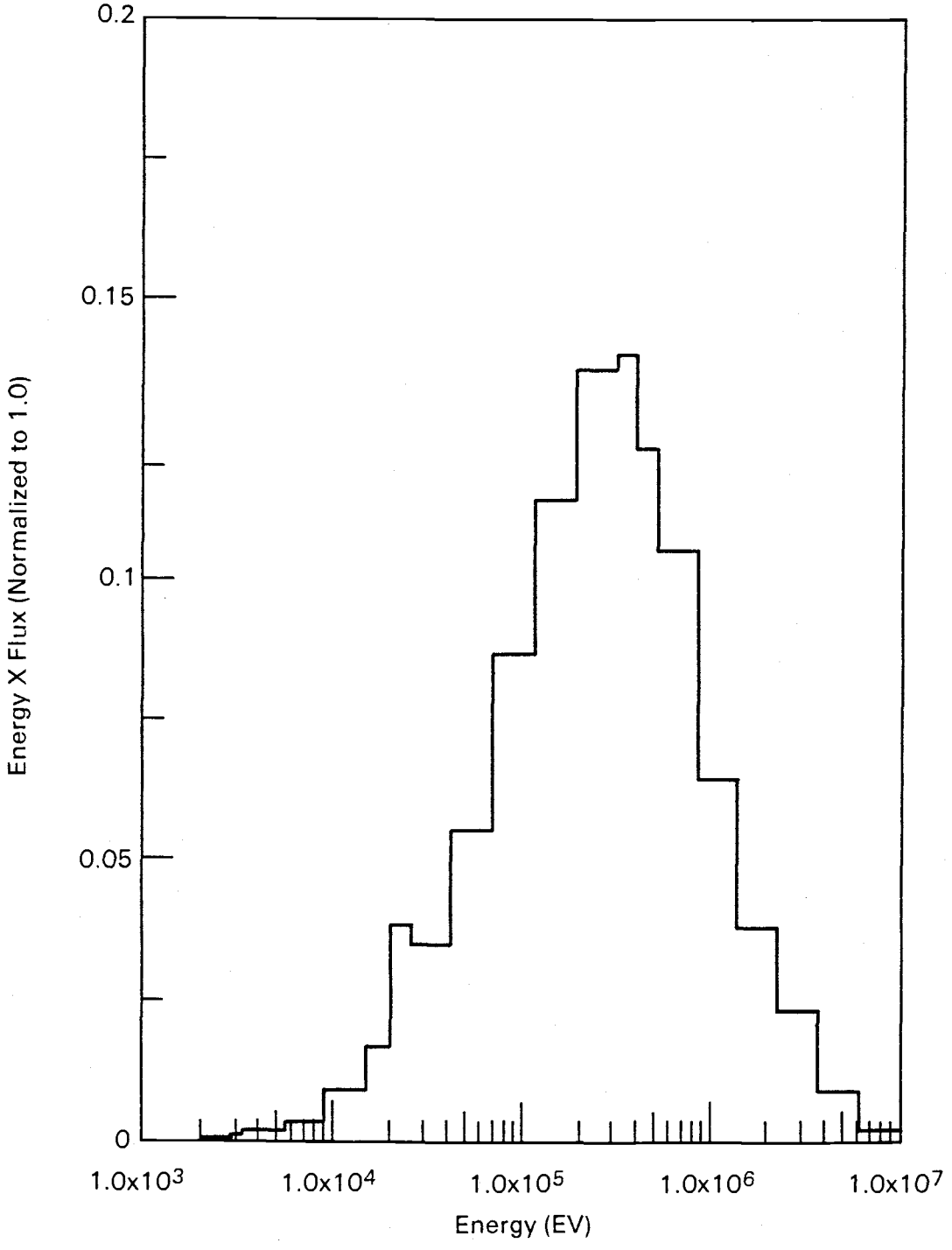


FIGURE E7. ACT6X9 Oxide Spectrum in an Infinite GCFR Configuration

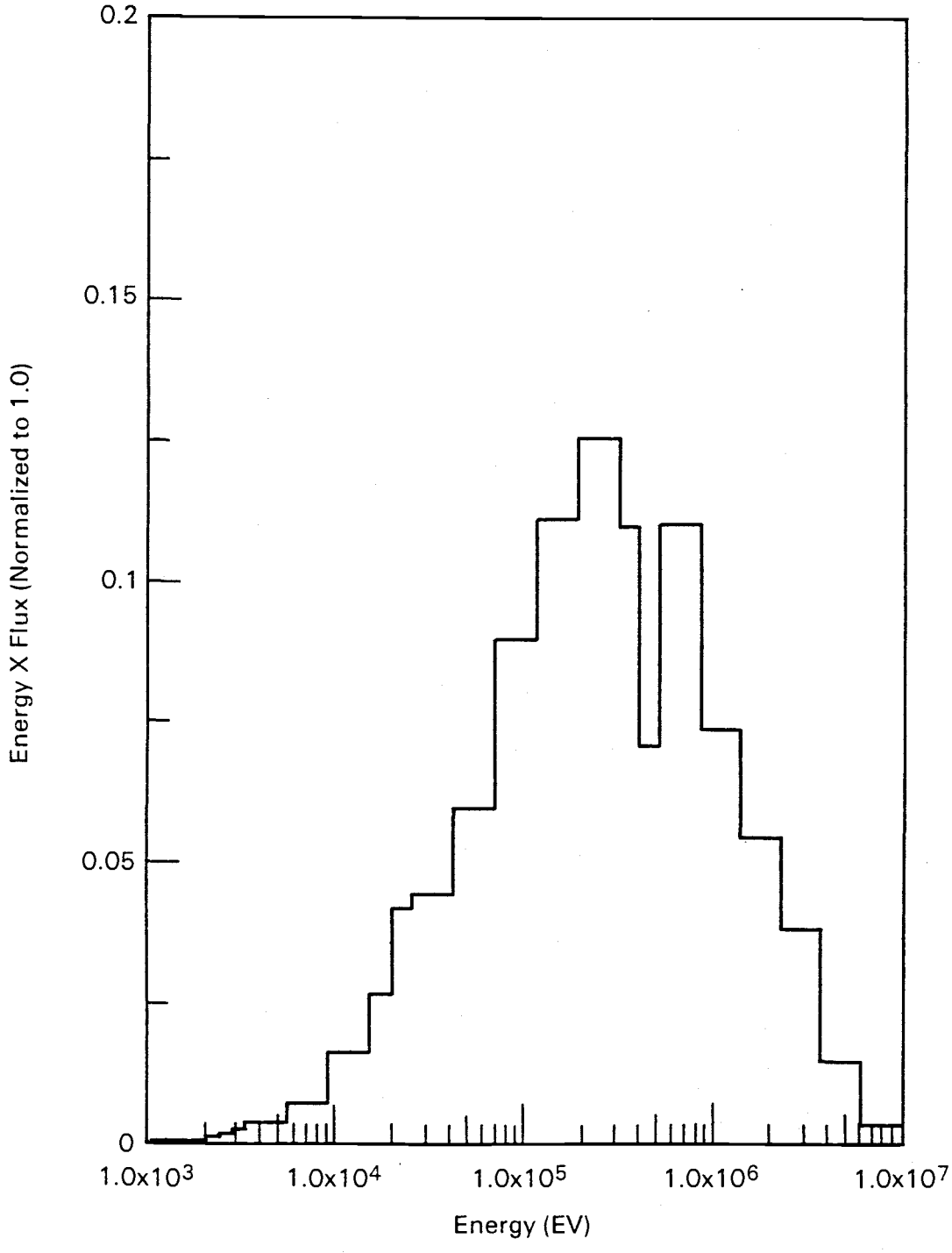


FIGURE E8. ACT6X9 Carbide Spectrum in an Infinite GCFR Configuration