Conf-911149--2

TRANSITION FROM HEU TO LEU FUEL IN ROMANIA'S 14-MW TRIGA REACTOR*

ANL/CP--74817

DE92 004138

M. M. Bretscher and J. L. Snelgrove
Argonne National Laboratory
Argonne, Illinois 60439 USA

DEC 13 1991

To be presented at the

XIV International Meeting on
Reduced Enrichment for
Research and Test Reactors

4-7 November 1991
Jakarta, Indonesia

---

*Work supported by the U.S. Department of Energy, Office of International Affairs and Energy Emergencies under Contract No. W-31-109-ENG-38.

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

TRANSITION FROM HEU TO LEU FUEL IN ROMANIA'S 14-MW TRIGA REACTOR

M. M. Bretscher and J. L. Snelgrove
Argonne National Laboratory
Argonne, Illinois 60439 USA

ABSTRACT

The 14-MW TRIGA steady state reactor (SSR) located in Pitesti, Romania, first went critical in the fall of 1979. Initially, the core configuration for full power operation used 29 fuel clusters each containing a 5 x 5 square array of HEU U (10 wt%) - ZrH - Er (2.8 wt%) fuel-moderator rods (1.295 cm o.d.) clad in Incoloy. With a total inventory of 35 HEU fuel clusters, burnup considerations required a gradual expansion of the core from 29 to 32 and finally to 35 clusters before the reactor was shut down because of insufficient excess reactivity. At this time each of the original 29 fuel clusters had an average ^{235}U burnup in the range from 50 to 62%.

Because of the U.S. policy regarding the export of highly enriched uranium, fresh HEU TRIGA replacement fuel is not available. After a number of safety-related measurements, the SSR is expected to resume full power operation in the near future using a mixed core containing five LEU TRIGA clusters of the same geometry as the original fuel but with fuel-moderator rods containing 45 wt% U (19.7% ^{235}U enrichment) and 1.1 wt% Er. Rods for 14 additional LEU fuel clusters will be fabricated by General Atomics.

In support of the SSR mixed core operation numerous neutronic calculations have been performed. This paper presents some of the results of those calculations.

INTRODUCTION

The Steady State Reactor (SSR) located in Pitesti, Romania, is a 14-MW TRIGA research reactor and is operated by the Institute for Nuclear Research (INR). It is equipped with two beam tubes, one radial and one tangential, and an underwater thermal column used for research purposes. Radioisotopes are also produced for both medical and industrial applications. However, the reactor is mostly used for long-term irradiation testing of power reactor fuel components including CANDU-type UO_2 fuel pellets. This is an important part of a development program for the fabrication of CANDU fuel.

Each of the SSR fuel clusters consists of a square 5 x 5 array of fuel rods each with an active diameter of 1.295 cm and an active height of 55.88 cm. The cylindrical uranium-zirconium hydride-erbium fuel/moderator pellets are tightly enclosed within Incoloy 800 tubes having a 0.0406 cm wall thickness. A square aluminum shroud, 8.89 cm on a side, encloses the 25 fuel pins and spacers. Each of the 8 control assemblies consists of a square annular arrangement of sintered natural B_4C compacts and an aluminum follower rod. Unclad beryllium reflector blocks, some containing a central irradiation hole, make up the radial reflector.

The SSR first went critical in November 1979.[1] Full power (14-MW) operation began in 1980 with 29 HEU fuel clusters. As burnup proceeded, the core size was first increased to 32 clusters and later to 35 clusters which used the entire fuel inventory. After about 13,600 MWD's of operation,[2] the SSR was shut down in 1990 because of insufficient excess reactivity and because of the unavailability of fresh HEU TRIGA replacement fuel. Figure 1 shows the present configuration of the SSR core.[3]

After 13,200 MWD's of reactor operation, the fuel pins were gamma-scanned to measure their axially-averaged total MWD exposures.[4] Some shuffling of pins within fuel clusters was done to maximize the available reactivity. The best estimate[5] for the total MWD's exposure for each of the 35 fuel clusters used in the Fig. 1 core configuration is given in Table 1. This table also includes the cluster-averaged ^{235}U burnups which were calculated from the MWD exposure data by methods to be discussed later.

The SSR is expected to resume operation in the near future beginning with five fresh LEU clusters in a 35-cluster HEU/LEU mixed core configuration. Within the next year 14 additional LEU fuel elements will become available. Table 2 summarizes the fuel specifications for both the original HEU and the new LEU clusters.

This paper is mostly concerned with an evaluation of the expected lifetime of the HEU/LEU mixed-fuel SSR cores. Calculations of important safety parameters for mixed core operations are incomplete at this time. Using special instrumented pins, numerous fuel temperature measurements will be made as a function of reactor power at important locations before full-power operation begins. These measurements are needed to assure that no fuel pin temperature exceeds 750 ^{o}C at full-power steady state operation.

ATOM DENSITIES, CROSS SECTIONS, AND MODELING METHODS

Before calculating fuel lifetimes, it is worthwhile to compare analytical results for excess reactivities with corresponding measured values. This provides an indication of the magnitude of biases which may be present in the calculations. For the case of the SSR, measured excess reactivities have been determined for the initial critical configuration and for the standard 29-cluster core, both with fresh HEU fuel.[1] Also, an estimate of the excess reactivity for the current core configuration (Fig. 1) has been made by the SSR staff.[2]

However, before such calculations and comparisons can be made it is necessary to determine representative atom densities in the fresh uranium-zirconium hydride-erbium fuel rods. This requires some knowledge of the fuel fabrication procedures.[6] The process begins by mixing pure powders of uranium, erbium, and zirconium in the weight fractions specified in Table 2. This mixture is arc-melted in a graphite crucible during which time a small amount of carbon (typically about 0.4 wt%) is taken up from the crucible into the molten alloy. After casting into molds, the fuel pellets are machined to the proper dimensions. Thereafter they are hydrated under carefully controlled conditions of hydrogen pressure, temperature, and time. Some final machining is necessary to assure close contact between the pellets and the Incoloy tubing before sealing the fuel tubes under a helium atmosphere.

While the fuel is in the molten state the available carbon unites with zirconium to form ZrC. During the hydration process the balance of the zirconium forms the zirconium hydride phase, $ZrH_{1.6}$. In addition, erbium hydride forms, although whether it forms $ErH_{1.6}$ or ErH_2 is uncertain. For the purpose of calculations we assume the state to be ErH_2. Analyses of the end product show that UC_2 and UH_3 are not present. Knowing the weight fractions (WF) of uranium, erbium, and zirconium in the initial powder (Table 2) and assuming the alloy contains

0.4 wt% C, the weight fractions of U, ZRC, $ZrH_{1.6}$, and ErH_2 in the final product can be calculated. The density (ρ_m) of the fuel/moderator material may be expressed in terms of these weight fractions by using the principle of volume conservation. Thus,

$$\rho_m = \left[\frac{WF(U)}{\rho_U} + \frac{WF(ZrC)}{\rho_{ZrC}} + \frac{WF(ZrH_{1.6})}{\rho_{ZrH_{1.6}}} + \frac{WF(ErH_2)}{\rho_{ErH_2}}\right]^{-1}$$

where $\rho_{ZrC} = 6.73$, $\rho_{ZrH_{1.6}} = 5.64$, and $\rho_{ErH_2} = 8.36$ g/cc.

The density of uranium metal varies somewhat with enrichment but can be expressed in terms of the density of natural uranium (19.05 g/cc). The above equation for the density neglects the void fraction because it is negligibly small. It also neglects small and unpredictable erbium volatility losses which may take place while the fuel is in the molten state. With the weight fractions and density determined, it is a simple matter to calculate the required atom densities in the fuel pin. Table 3 shows these results for the unirradiated HEU and LEU fuels.

The broad, multigroup, burnup-dependent cross sections used in this study were generated by the EPRI-CELL code.[7] The EPRI-CELL fine group libraries were derived from the ENDF/B data files (some cross sections from Version V but most from Version IV) using the processing codes NJOY[8] for the thermal library and MC^2-2[9] for the fast library. Cross section sets were generated at fuel temperatures corresponding to 296K, 500K, and 800K in both 3 and 8 groups. Table 4 shows the group energy boundaries for each set. Because of the strong absorption resonance in ^{167}Er at low energy (0.46 eV) it was felt that the additional thermal groups in the 8-group set would be needed for more accurate calculations.

For the calculations reported here the SSR was modeled in XYZ geometry. Details concerning the dimensions of the fuel elements, the beryllium reflector pieces, the B_4C control rods, and the aluminum followers are given in Ref. 10. M. Ciocanescu[3] provided the information needed to model the experimental loops AL6 and C-1 shown in Fig. 1. In the XY plane each fuel cluster is represented by a square homogenized fuel region surrounded by a thin square annulus of aluminum and water representing the shroud region. The height of the active fuel column was divided into five axial segments of equal length. For diffusion and burnup calculations the DIF3D[11] and REBUS-3[12] codes were used. Relative to a detailed continuous-energy Monte Carlo calculation using the VIM code[13] for a 29-cluster model of the SSR, the best DIF3D calculations are obtained when the shroud region is explicitly modeled and when the 8-group cross section set is used, as Table 5 shows. For these reasons all the remaining neutronic calculations given in this report are based on the use of 8-group cross sections and explicit representations of the shroud and fuel regions for each cluster.

CALCULATED VERSUS MEASURED EXCESS REACTIVITIES FOR SSR FRESH FUEL ASSEMBLIES

As a test of the adequacy of these cross sections, modeling procedures, and computational methods, calculated and measured excess reactivities have been compared for two SSR assemblies each with all fresh HEU fuel clusters. The first assembly is the configuration obtained during the initial approach-to-critical experiments.[1] Figure 2 shows the SSR fuel cluster loading sequence. With all experimental regions and unoccupied grid positions filled with water, the SSR went critical upon the addition of the 17th fuel cluster. Based on an estimated value of β_{eff} of 0.007 and the elevations of the calibrated control rods, the measured excess reactivity for this 17-cluster core was \$1.43.[6] After these measurements were completed the core was expanded to the standard 29-cluster configuration, which is also shown in Fig. 2. This larger core serves as our second test case for which the excess reactivity was found to be $7.78.[1] Table 6 compares our calculated values with these measured results. A direct comparison between calculation and measurement independent of β_{eff} will be made when we obtain information regarding the elevation of the control rods at criticality for each of these cores. Table 6 shows that the calculations are in reasonable agreement with the measurements.

EXCESS REACTIVITY FOR THE PRESENT SSR CORE CONFIGURATION WITH 35 BURNED HEU FUEL CLUSTERS

This core configuration is illustrated in Fig. 1. Before calculations can be made, however, it is necessary to obtain burnup-dependent atom densities for the homogenized fuel region for each of the fuel clusters beginning with the MWD exposure data given in Table 1. This was accomplished by first performing a non-equilibrium REBUS-3 burnup calculation beginning with the 35-cluster SSR core configuration (Fig. 1 but with all fresh HEU fuel). The calculation covered a total burnup span of 1000 full power (14-MW) days. Burnup-dependent atom densities were printed out at numerous intermediate burn steps. These data of axially-averaged cluster atom densities versus MWD's exposure were fit by the least squares process to polynomials so that the fitting coefficients could be used to interpolate burnup-dependent atom densities corresponding to the MWD's exposure for each of the 35 fuel clusters listed in Table 1. Some typical fits and interpolated results are shown in Figs. 3-7. The same procedure was used to obtain the average ^{235}U percent burnup for each of the 35 clusters. These results are given in Table 1 and displayed in Fig. 8. Similar procedures were followed to obtain exposure-dependent concentrations of ^{3}H, ^{3}He, and ^{6}Li in the beryllium reflector (see Fig. 8). These poisons result from the ^{9}Be(n,α)^{6}He fast neutron reaction, the ^{6}Li(n,α)^{3}H thermal neutron reaction, and the beta-decay of tritium into ^{3}He. After about 1000 FPD's of reactor operation these poisons have a reactivity effect of about 0.25% δk/k. Following a two-year shutdown period this reactivity increases to about 0.37% δk/k due to the buildup of ^{3}He, which has a very large thermal neutron absorption cross section.

The atom densities found from the above procedure are axially-averaged values. For three-dimensional calculations it is necessary to superimpose on these average values an appropriate axial burnup distribution. Figure 9, supplied by M. Ciocanescu,[5] shows such an axial distribution of the measured ^{137}Cs fission product activity. This distribution was numerically integrated over each of the five equal axial fuel segments to obtain segment-to-average burnup ratios. These ratios, combined with the data in Table 1, were used to construct axially-dependent exposure data for each of the 35 fuel clusters. Corresponding axially-dependent atom densities were then obtained using the methods already described subject to the constraint that the total number of atoms for each burnup-dependent isotope in each fuel cluster be preserved. In this manner burnup and axially-dependent atom densities were obtained for each of the fuel clusters for ^{2}H, ^{149}Sm, ^{166}Er, ^{167}Er, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu and the lumped fission products.

These axially-dependent atom densities were used in a three-dimensional diffusion calculation for the 35-cluster core shown in Fig. 1 with all the control rods fully withdrawn. The eigenvalues obtained without and and with equilibrium xenon (at 14 MW) were 1.0326 and 1.0067, respectively. At the time the SSR was shut down the estimated excess reactivity at full power with equilibrium xenon was about $1.[2] As mentioned earlier, the cross sections used for these calculations were generated for an average fuel temperature of 227 °C. From Ref. 5, the SSR average core fuel temperature is 170 °C at 10 MW. Thus, our calculated results agree quite well with Romania's best estimate for the present state of the SSR.

FUEL LIFETIME ESTIMATES FOR PROPOSED SSR HEU/LEU MIXED CORES

Having established the validity of the cross sections and modeling methods used for SSR neutronic calculations, estimated lifetimes of proposed mixed cores will now be evaluated. With the expectation that the Institute for Nuclear Research (INR) will soon receive five LEU fuel clusters, M. Ciocanescu[3] proposed the core configuration shown in Fig. 10 for their use. Following the procedures and methods discussed earlier, three-dimensional non-equilibrium REBUS-3 burnup calculations were carried out for this core configuration. The results of these calculations are illustrated in Fig. 11. They were done with the 8-group cross sections for an average fuel temperature of 227 °C at 14 MW. The sudden initial drop in reactivity in Fig. 11 is the result of the rapid buildup of equilibrium xenon. Figure 11 shows that if no changes other than burnup occur during the total burn cycle, the five LEU elements are expected to extend the SSR core lifetime by about 250 full-power (14 MW) days. Nearly the same result is obtained if control rods 1 and 2 are fully withdrawn while the others are banked together and adjusted to near critical throughout the burn cycle. Although the calculation has not been performed, some improvement in lifetime would be expected if the total burn cycle were divided into a few shorter cycles with the fuel shuffled between cycles so that elements of minimum burnup would be located in positions of maximum worth.

Within a year the Institute for Nuclear Research is expected to receive a shipment of 14 additional LEU fuel elements from General Atomics for use in the SSR. When these elements arrive, M. Ciocanescu proposes to reconfigure the core from the 35-cluster arrangement shown in Fig. 10 to the more standard 29-cluster configuration described in Fig. 12. Beginning with atom densities corresponding to 200 FPD's of operation of the Fig. 10 core, the results of three-dimensional calculations for the lifetime of the 29-cluster core are summarized in Fig. 13. Clearly evident from this figure is the dramatic improvement in the core lifetime which results if after each subcycle of 126 FPD's the reactor is shutdown for 7 days and the fuel shuffled in a way so that, at least approximately, the least burned clusters are located in positions of maximum worth and visa versa. Figure 13 shows that by such a procedure the core lifetime is extended from about 310 FPD's without fuel shuffling to about 560 FPD's with shuffling. How the fuel elements were shuffled from cycle to cycle is indicated in Table 7. The final axially-averaged burnup status for each of the HEU and LEU fuel clusters after 546 FPD's of operation of the last core (Fig. 12) is summarized in Table 8. Peak axial ^{235}U burnups are estimated to be as high as 94% for the HEU and 47% for the LEU fuel.

CONCLUSION

The 14-MW TRIGA research reactor in Pitesti, Romania, has been shutdown because the fuel is too highly burned to continue operations. However, 125 unirradiated LEU fuel pins for five clusters have been shipped to the reactor site to allow the SSR to resume activities for which it was intended. Based on neutronic methods tested against known SSR critical configurations, the five LEU clusters located in high worth positions in a 35-element core should allow the SSR to operate up to 250 full-power days. This will allow the time needed by General Atomics to fabricate fuel pins for 14 additional LEU clusters. Returning to the standard 29-cluster configuration but with a mixed HEU/LEU core with 19 LEU bundles, our calculations show that by periodically reshuffling the fuel the SSR could operate for an additional 560 full power days. At this time new fuel will be needed. However, some additional operation time may be possible by expanding the core from 29 to 35 fuel elements, although no calculations of this nature have been undertaken.

Before the SSR will resume operations at full power with mixed HEU/LEU cores, important safety-related measurements will be made at low power. These measurements are designed to assure that at 14 MW operation no fuel pin will have a maximum temperature exceeding 750°C and that adequate shutdown margins are always available. This temperature limit for the safe operation of the TRIGA fuel under steady state conditions has been set by General Atomics[10] and is a conservative limit based upon radiation and fission-product-pressure-induced swelling of the erbium-uranium-zirconium hydride matrix. In addition to the experimental studies, numerous safety parameters (power peaking factors, temperature coefficients, control rod worths, shutdown margins, and kinetic parameters) are being calculated and thermal hydraulic studies will be undertaken as part of an ANL/INR joint study program.

After a shutdown period of two years, full power operation of the SSR is expected to resume in the near future with a mixed HEU/LEU core configuration.

REFERENCES

1. W. L. Whittemore, "Startup Testing of the Romania Dual-Core Test Reactor," Seventh Biennial U.S. TRIGA Users' Conference, San Diego, CA, March 2-5, 1980, GA TOC-12.

2. C. Iorgulis, Institute for Nuclear Research, Pitesti, Romania, private communication, February 1991.

3. M. Ciocanescu, Institute for Nuclear Research, Pitesti, Romania, private communication, March 27, 1991.

4. G. Iorgulis and C. Costescu, "Tridimensional Neutronic Calculation Methods Used in Romanian 14 MW TRIGA-SSR Fuel Management and Irradiation Experiments Evaluation," Eleventh European TRIGA Users' Conference, Heidelberg, Germany, September 11-13, 1990, GA TOC-22.

5. M. Ciocanescu, Institute for Nuclear Research, Pitesti, Romania, private communication, August 13, 1991.

6. G. B. West, General Atomics, private communication, August, 1991.

7. B. A. Zolotar, et al., "EPRI-CELL Code Description," Advanced Recycle Methodology Program System Documentation, Part II, Chapter 5, September 1977.

8. R. E. MacFarlane, et al., "The NJOY Nuclear Data Processing System: User's Manual," LA-7584-M, December 1978.

9. H. Henryson III, B. J. Toppel, and C. G. Stenberg, "MC^2-2: A Code to Calculate Fast Neutron Spectra and Multigroup Cross Sections," Argonne National Laboratory Report ANL-8144 (June 1976).

10. "Safety Analysis Report of the TRIGA Steady-State Research/Materials and Testing Reactor for the Institute of Nuclear Technologies, Bucharest, Romania," General Atomic Company, E-117-323, Vol. II, February 1974.

11. K. L. Derstine, "DIF3D: A Code to Solve One, Two, and Three-Dimensional Finite-Difference Diffusion Theory Problems," Argonne National Laboratory Report ANL-82-64 (April 1984).

12. B. J. Toppel, "A User's Guide for the REBUS-3 Fuel Cycle Analysis Capability," Argonne National Laboratory Report ANL-83-2 (March 1983).

13. R. N. Bloomquist, "VIM - A Continuous Energy Neutronics and Photon Transport Code," pp. 222-224, ANS Proceedings of the Topical Meeting on Advances in Reactor Computations, Salt Lake City, Utah, March 28-31, 1983.

TABLE 1. SSR HEU FUEL CLUSTER EXPOSURES AND U-235 BURNUPS (BASED ON GAMMA-SCANNING MEASUREMENTS)

Fuel Cluster	MWD's Exposure	% U-235 Burnup
H11	495	62.4
H05	479	60.4
H25	475	59.9
H01	464	58.6
H04	460	58.0
H07	453	57.3
H13	453	57.2
H23	448	56.6
H22	447	56.5
H21	444	56.2
H02	440	55.6
H12	438	55.4
H18	433	54.8
H24	433	54.7
H26	431	54.6
H03	431	54.5
H17	428	54.1
H27	427	54.0
H29	426	53.9
H19	426	53.8
H08	425	53.8
H15	422	53.4
H10	422	53.4
H20	418	52.9
H28	416	52.7
H06	411	52.0
H09	406	51.5
H16	399	50.5
H14	393	49.8
H31	246	31.4
H30	245	31.3
H32	195	24.9
H35	35	4.5
H34	33	4.2
H33	29	3.7

TABLE 2. SSR TRIGA FUEL PIN LOADINGS
(Before Alloy Formation and Subsequent Hydration)

Element/Isotope		35 HEU Clusters Wt %		5 LEU Clusters Wt %		14 LEU Clusters Wt %	
U		10.0		45.0		45.0	
	234		1.00		0.15		0.15
	235		93.09		19.79		19.7
	236		0.43		0.25		0.25
	238		5.48		79.81		79.9
Er		2.8		1.1		1.3	
	162		0.14		0.14		0.14
	164		1.58		1.58		1.58
	166		33.33		33.33		33.33
	167		22.90		22.90		22.90
	168		26.91		26.91		26.91
	170		15.14		15.14		15.14
Zr (Balance)		87.2		53.9		53.7	

TABLE 3. ATOM DENSITIES IN FRESH SSR TRIGA FUEL PINS

Element/Isotope		35 HEU Clusters atoms/barn-cm	5 LEU Clusters atoms/barn-cm	14 LEU Clusters atoms/barn-cm
H		5.4867 E-02	4.4714 E-02	4.4700 E-02
C		1.2154 E-03	1.6517 E-03	1.6535 E-03
Zr		3.4747 E-02	2.9201 E-02	2.9111 E-02
Er		6.0841 E-04	3.1719 E-04	3.8436 E-04
	166	2.0442 E-04	1.0658 E-04	1.2915 E-04
	167	1.3963 E-04	7.2796 E-05	8.8211 E-05
U		1.5455 E-03	9.3621 E-03	9.3723 E-03
	234	1.5532 E-05	1.4247 E-05	1.4263 E-05
	235	1.4396 E-03	1.8720 E-03	1.8652 E-03
	236	6.6680 E-06	2.3543 E-05	2.3569 E-05
	238	8.3664 E-05	7.4523 E-03	7.4693 E-03
Pin Density: (g/cc)		6.1522	8.3106	8.3196

TABLE 4. BROAD GROUP STRUCTURES

Group Number	Upper Energy of Group (eV)	
	3-Group Set	8-Group Set
1	1.00000E+07	1.00000E+07
2	5.53084E+03	6.39279E+05
3	1.85539E+00	9.11882E+03
4		1.85539E+00
5		1.16643E+00
6		4.17035E-01
7		1.45728E-01
8		5.69250E-02

TABLE 5. SSR EIGENVALUE COMPARISONS FOR A 29-CLUSTER HEU TRIGA CORE

Quantity	VIM-Monte Carlo	DIF3D-DIFFUSION			
No. of Groups:	Continous	8	3	8	3
Explicit Shroud Reg.?	Yes	No	No	Yes	Yes
K-Effective:	1.0306±0.0019	1.0427	1.0426	1.0300	1.0360

TABLE 6. SSR EIGENVALUES AND EXCESS REACTIVITIES FOR ROOM-TEMPERATURE, FRESH FUEL, HEU CRITICALS

QUANTITY	17-CLUSTER CORE	29-CLUSTER CORE
k_{eff} (C)	1.0103	1.0597
k_{eff} (E)	1.0101	1.0576
ρ_{excess} (C)	$1.45	$8.05
ρ_{excess} (E)	$1.43	$7.78

Note: Results are based on β_{eff}=0.007 (Ref. 10).

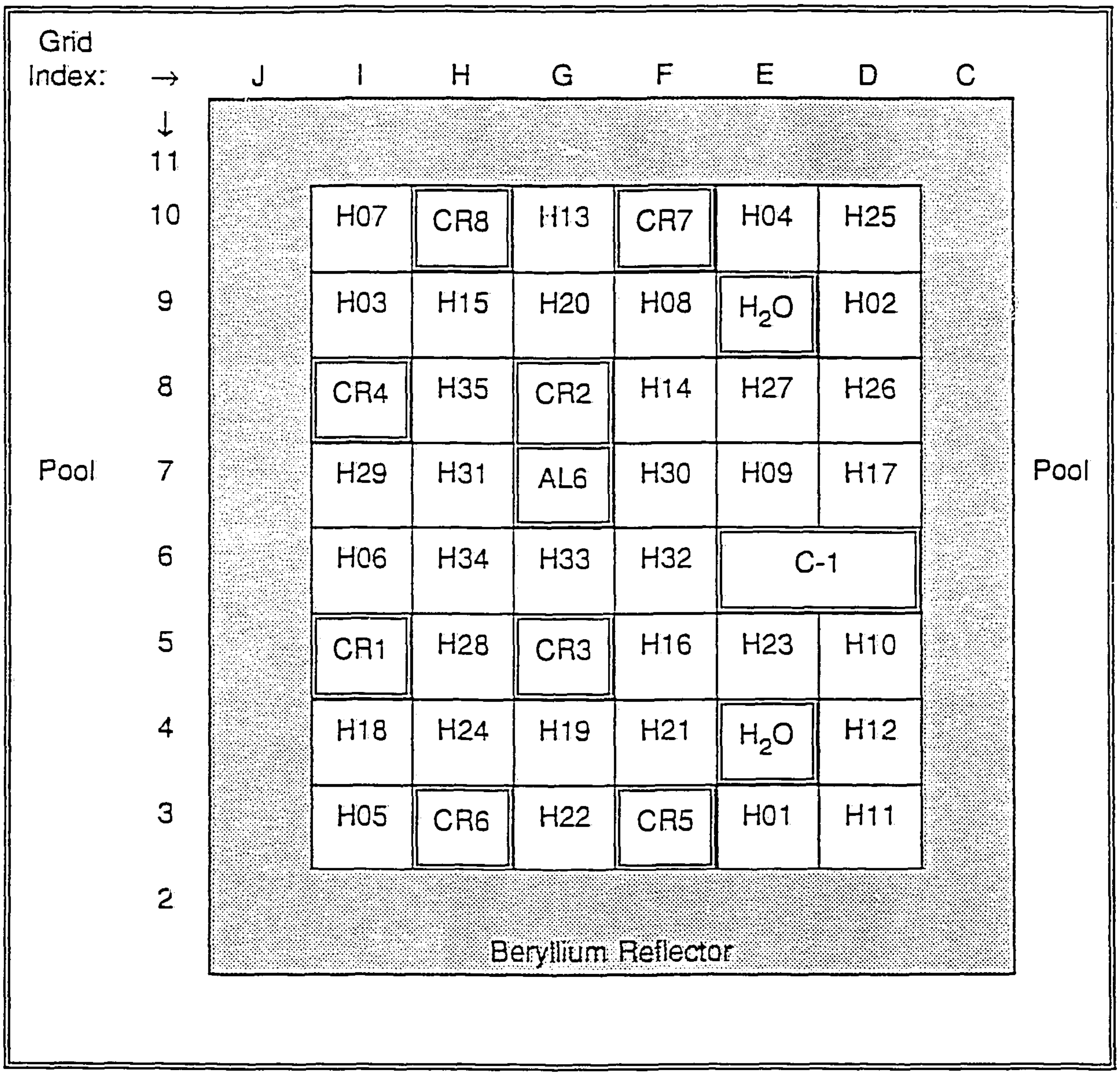
TABLE 7. FUEL SHUFFLING FOR THE 29-CLUSTER SSR CORE BEGINNING WITH 14 UNIRRADIATED LEU TRIGA ELEMENTS

Grid Location	Figure 10 Core	Fuel Cluster Location for Each 126 FPD Subcycle				
		Cycle 1	Cycle 2	Cycle 3	Cycle 4	Cycle 5
D4	H16	H07	H21	H30	H01	H01
D5	L09	H22	H23	H22	H25	H05
D8	L08	H23	H07	H23	H31	H31
D9	H23	H21	H22	H21	H11	H11
E3	H28	H33	H34	H35	H33	H33
E5	L07	L02	L03	L03	L02	L03
E8	L06	H35	H32	H32	H32	H32
E10	L10	H32	H30	H07	H05	H25
F4	L11	L17	L06	L06	L14	L08
F5	L04	L11	L14	L13	L07	L16
F6	H32	L14	L11	L07	L12	L14
F7	H30	L18	L07	L17	L10	L18
F8	L05	L05	L04	L05	L03	L01
F9	L12	H34	H33	H34	H35	H35
G3	H14	L04	L05	L04	L05	L04
G4	L13	L13	L12	L16	L09	L19
G6	H33	L06	L17	L10	L18	L12
G9	L16	L03	L02	L02	L04	L02
G10	L14	H31	H31	H31	H04	H04
H4	L15	L16	L09	L19	L13	L11
H5	L03	L07	L18	L11	L17	L13
H6	H34	L08	L19	L14	L08	L15
H7	H31	L10	L15	L12	L19	L10
H8	H35	L15	L10	L08	L15	L06
H9	L18	H30	H35	H33	H34	H34
I4	L17	L19	L08	L18	L11	L17
I6	L02	L09	L16	L15	L06	L09
I7	L01	L12	L13	L09	L16	L07
I9	L19	L01	L01	L01	L01	L05

TABLE 8. FINAL BURNUP STATUS OF THE SSR FUEL FOLLOWING 546 FPD'S OPERATION OF THE FIG. 12 CORE

HEU Fuel		LEU Fuel	
Cluster	% U-235 Burnup	Cluster	% U-235 Burnup
H01	63.0	L01	32.8
H02	60.0	L02	35.4
H03	59.2	L03	34.7
H04	62.8	L04	34.1
H05	67.1	L05	35.9
H06	60.1	L06	31.5
H07	65.0	L07	31.7
H08	59.5	L08	31.8
H09	60.6	L09	30.9
H10	69.4	L10	33.0
H11	65.1	L11	31.1
H12	60.4	L12	32.1
H13	60.2	L13	31.1
H14	58.8	L14	31.5
H15	59.1	L15	32.5
H16	60.6	L16	31.0
H17	60.4	L17	32.3
H18	59.4	L18	32.3
H19	60.6	L19	31.4
H20	59.2		
H21	69.2		
H22	68.0		
H23	73.9		
H24	62.3		
H25	63.8		
H26	59.3		
H27	62.7		
H28	61.2		
H29	62.6		
H30	60.1		
H31	63.1		
H32	60.1		
H33	46.8		
H34	47.6		
H35	43.9		

SSR CURRENT CORE CONFIGURATION
(35 HEU Fuel Clusters)



Grid Index: →	J	I	H	G	F	E	D	C
11								
10		H07	CR8	H13	CR7	H04	H25	
9		H03	H15	H20	H08	H_2O	H02	
8		CR4	H35	CR2	H14	H27	H26	
7		H29	H31	AL6	H30	H09	H17	
6		H06	H34	H33	H32	C-1		
5		CR1	H28	CR3	H16	H23	H10	
4		H18	H24	H19	H21	H_2O	H12	
3		H05	CR6	H22	CR5	H01	H11	
2								

CR = Control rod assemblies

AL6 = Experiment loop with 6 Candu fuel elements

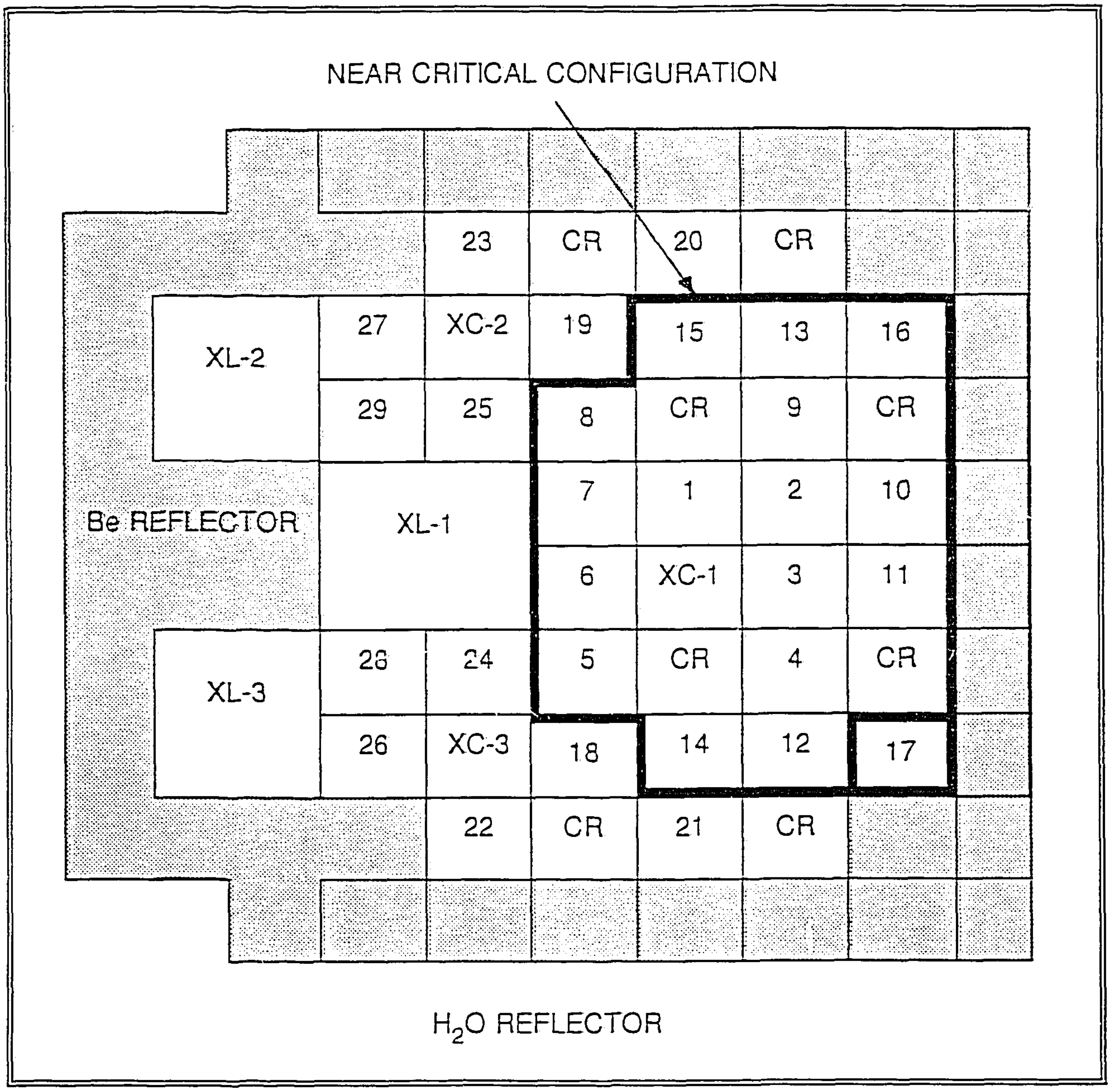
C-1 = C1 type capsule with 1 Candu fuel element

H_2O = Water hole

H01 = HEU fuel cluster

FIGURE 1

SSR GRID LOCATIONS FOR THE INITIAL CORE LOADING SEQUENCE



ALL EXPERIMENT LOCATIONS (XL-1, 2, 3; XC-1, 2, 3; AND THE HOLES IN THE BERYLLIUM REFLECTOR) ARE FILLED WITH WATER

FIGURE 2

SSR BURNUP–DEPENDENT ATOM DENSITIES IN HEU FUEL

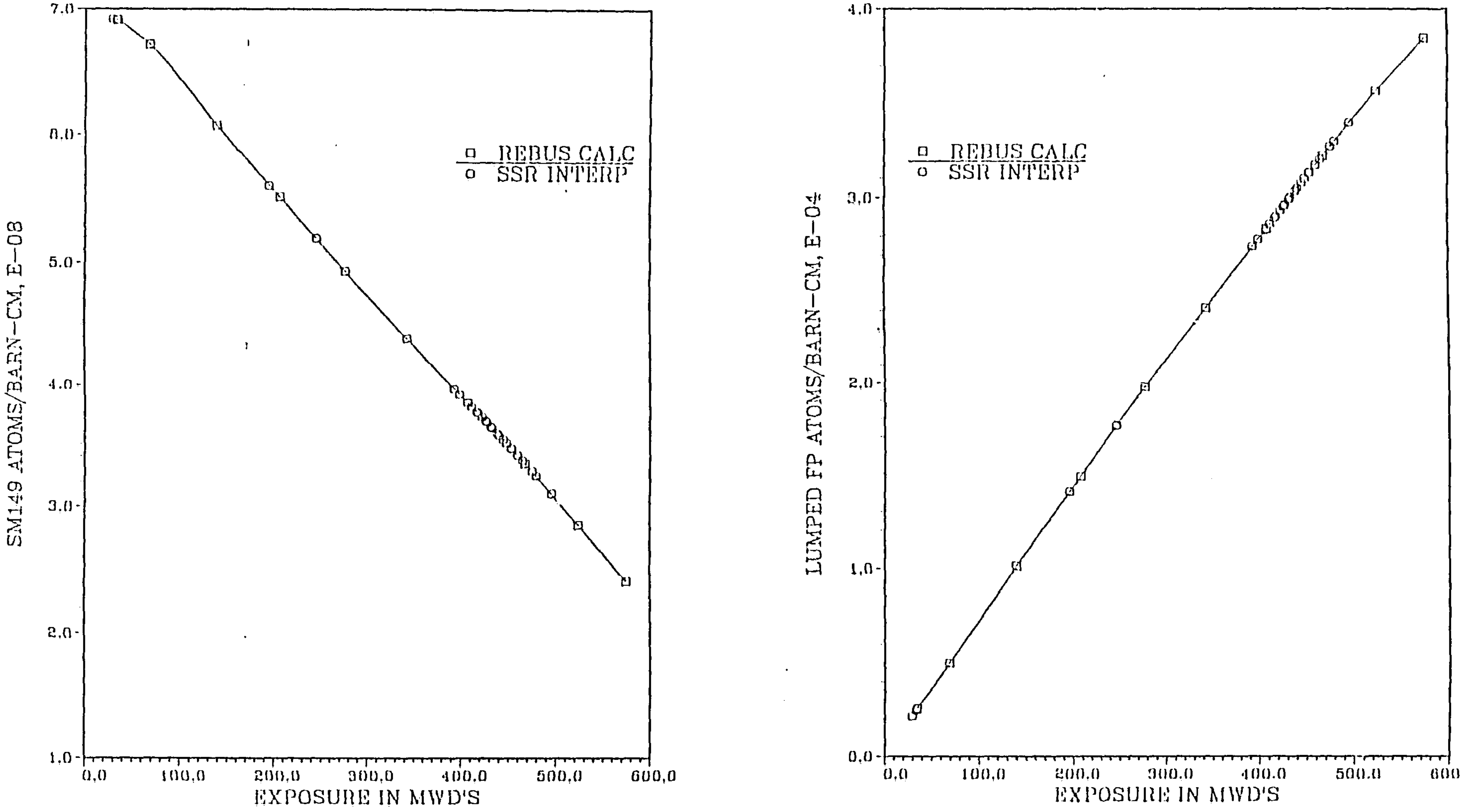


FIGURE 3

SSR BURNUP–DEPENDENT ATOM DENSITIES IN HEU FUEL

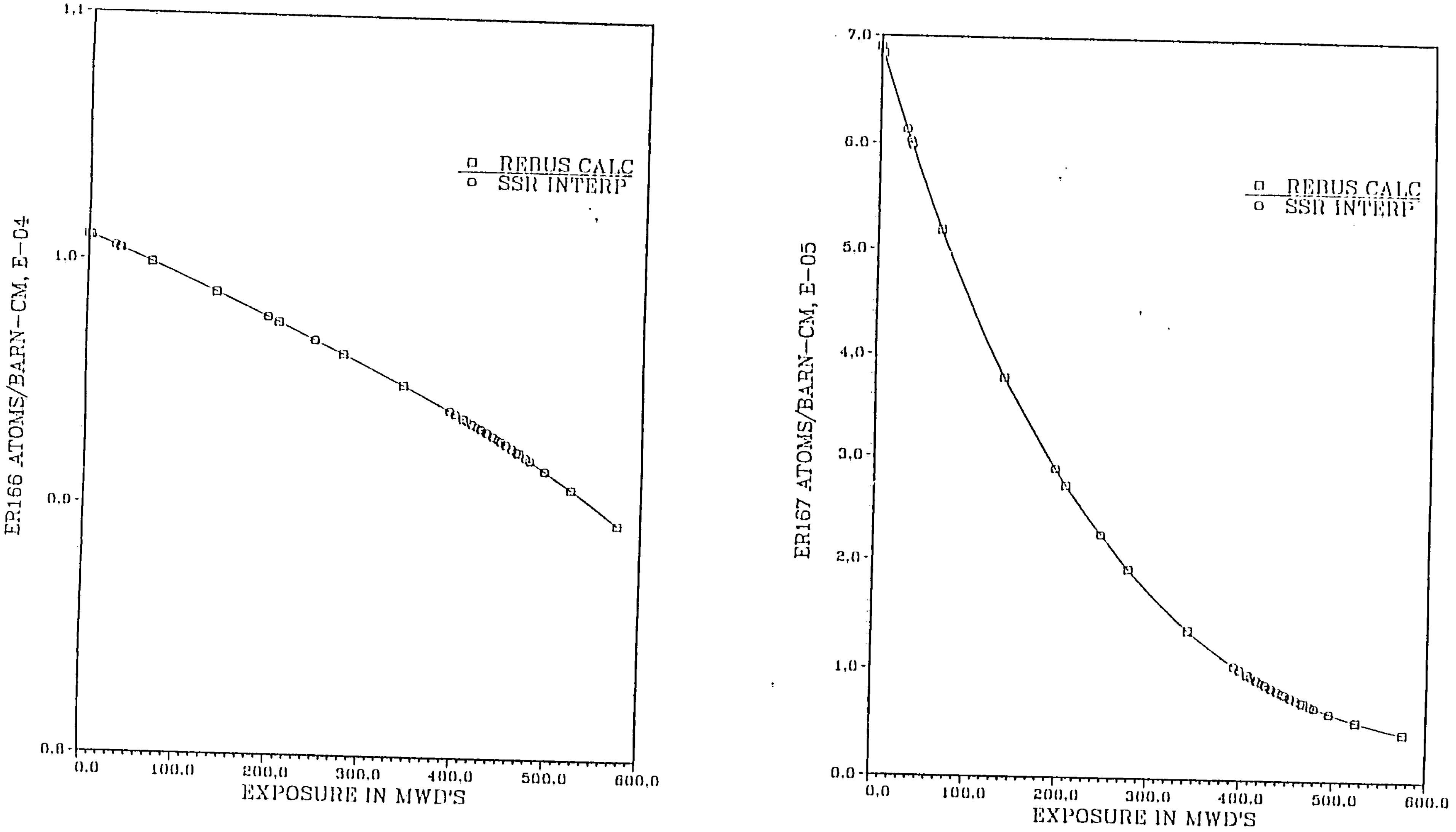


FIGURE 4

SSR BURNUP–DEPENDENT ATOM DENSITIES IN HEU FUEL

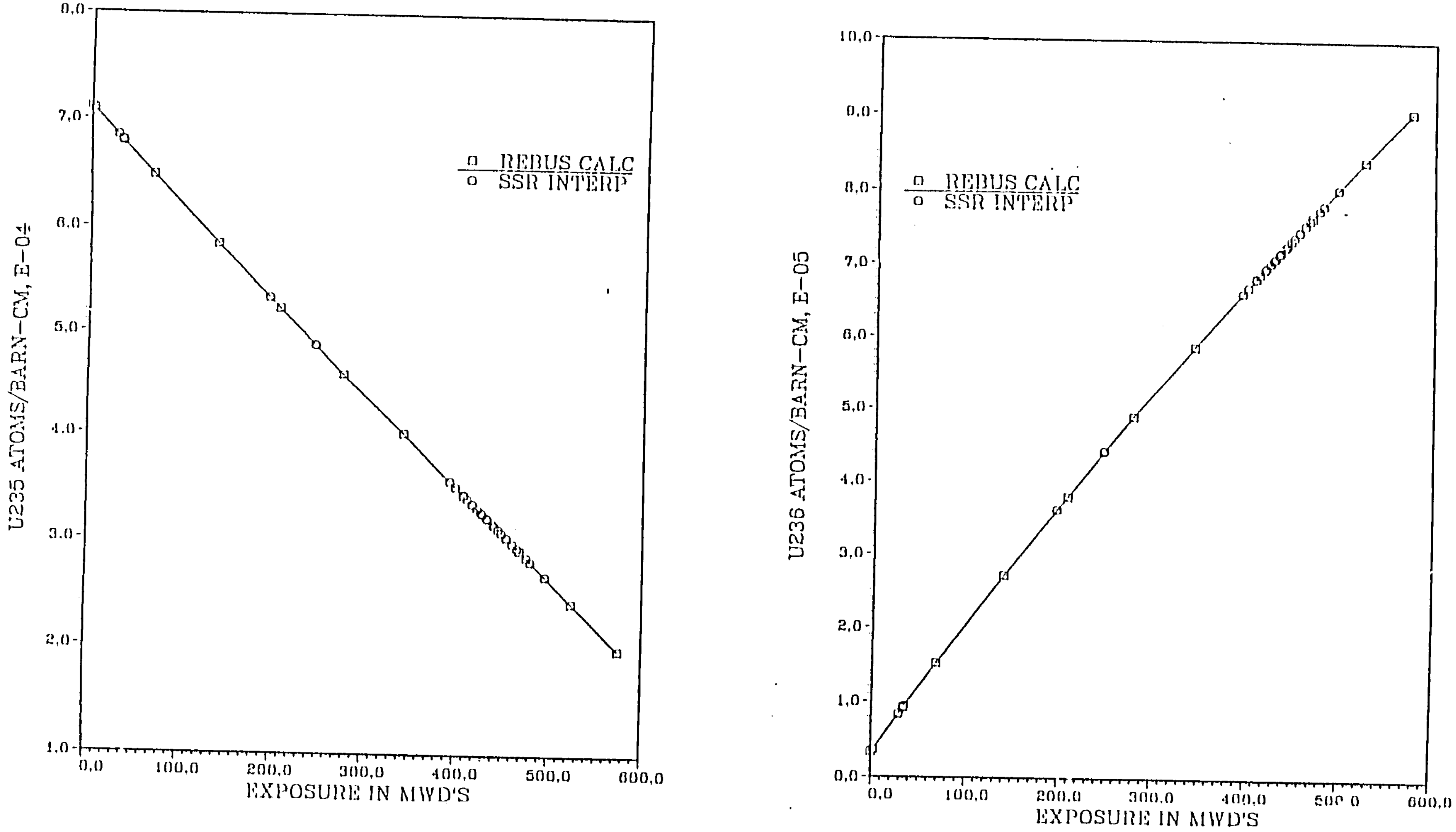


FIGURE 5

SSR BURNUP–DEPENDENT ATOM DENSITIES IN HEU FUEL

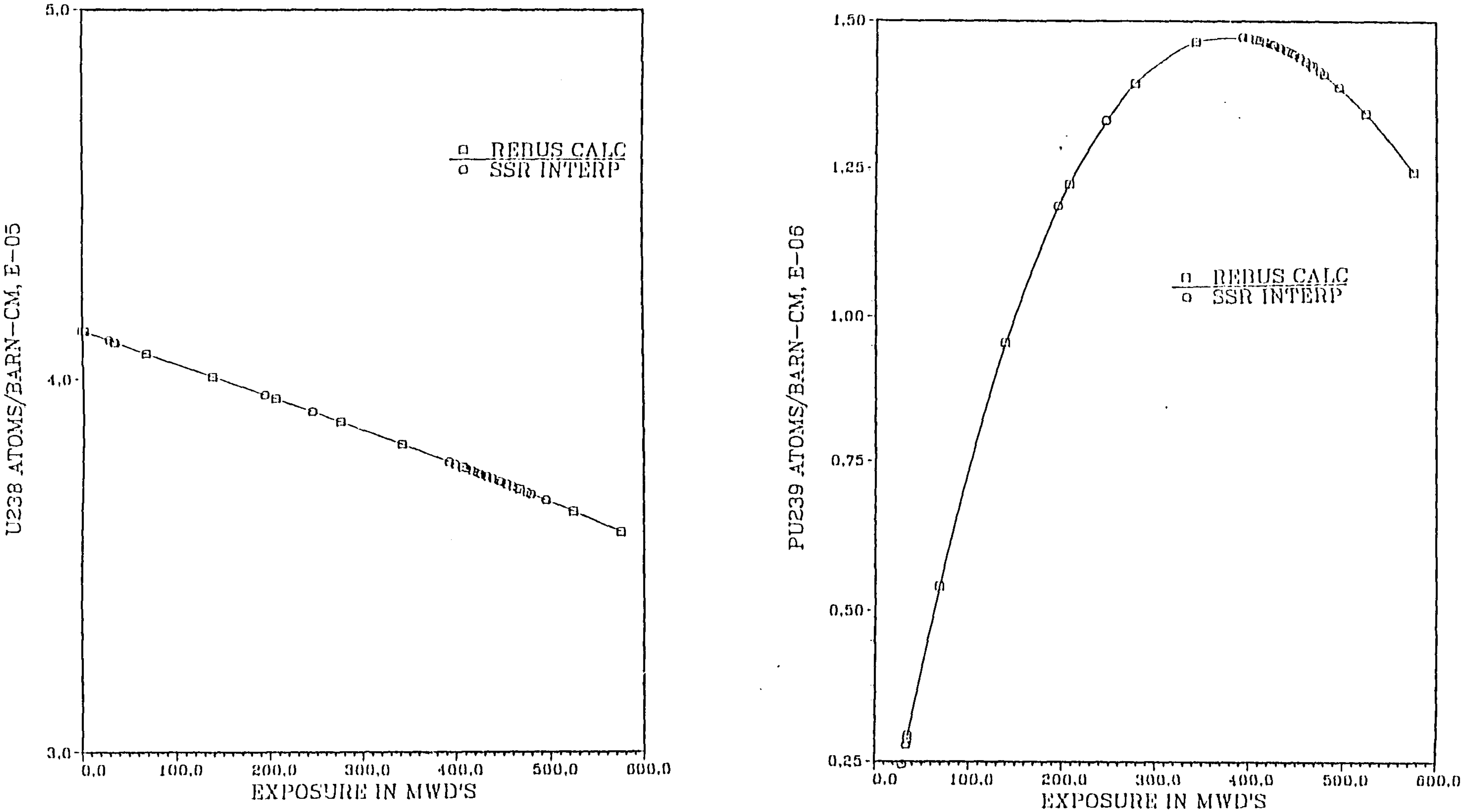


FIGURE 6

SSR BURNUP–DEPENDENT ATOM DENSITIES IN HEU FUEL

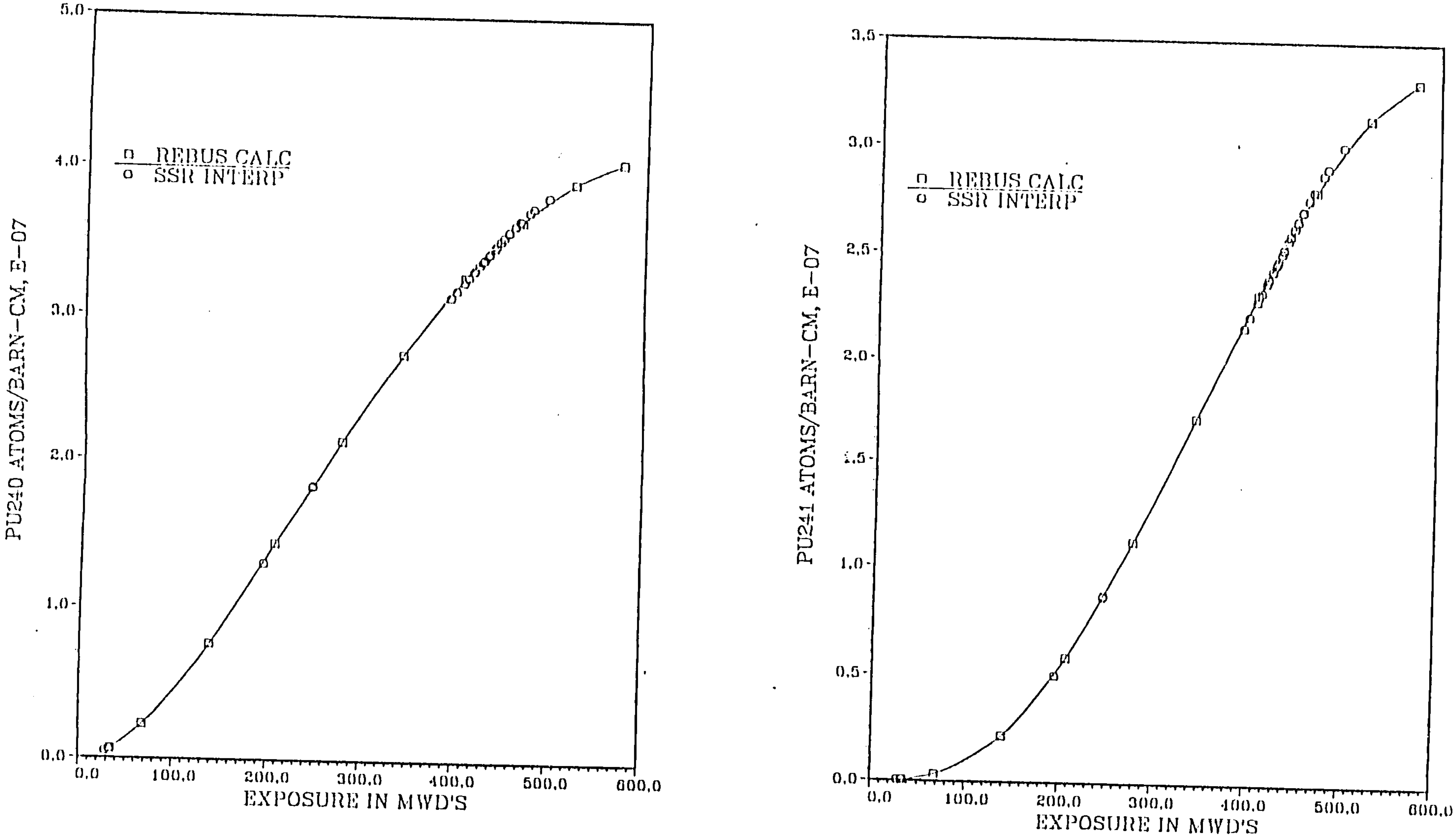
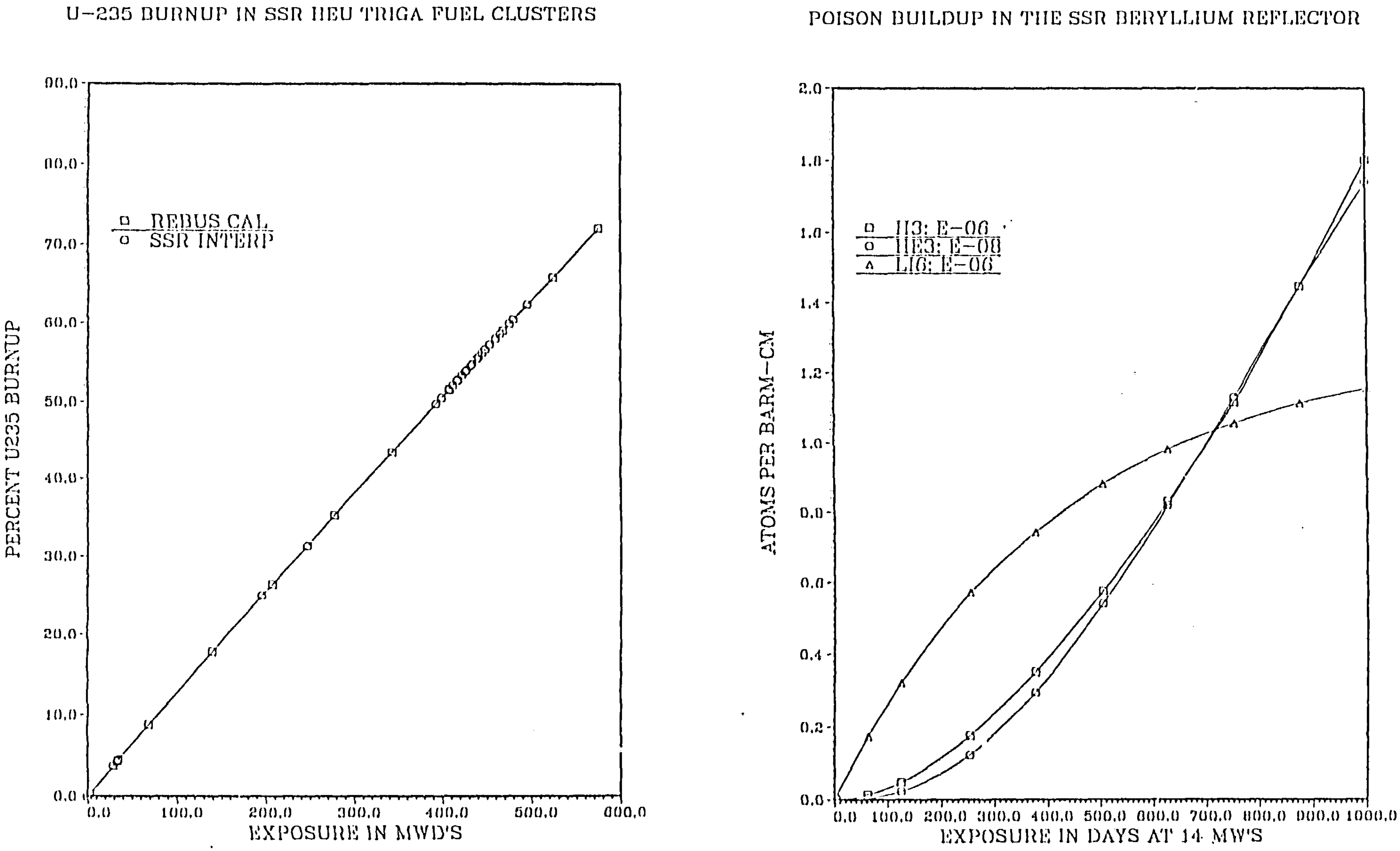


FIGURE 7


U-235 BURNUP IN SSR HEU TRIGA FUEL CLUSTERS

POISON BUILDUP IN THE SSR BERYLLIUM REFLECTOR

FIGURE 8

AXIAL BURNUP DISTRIBUTION IN SSR HEU FUEL PINS

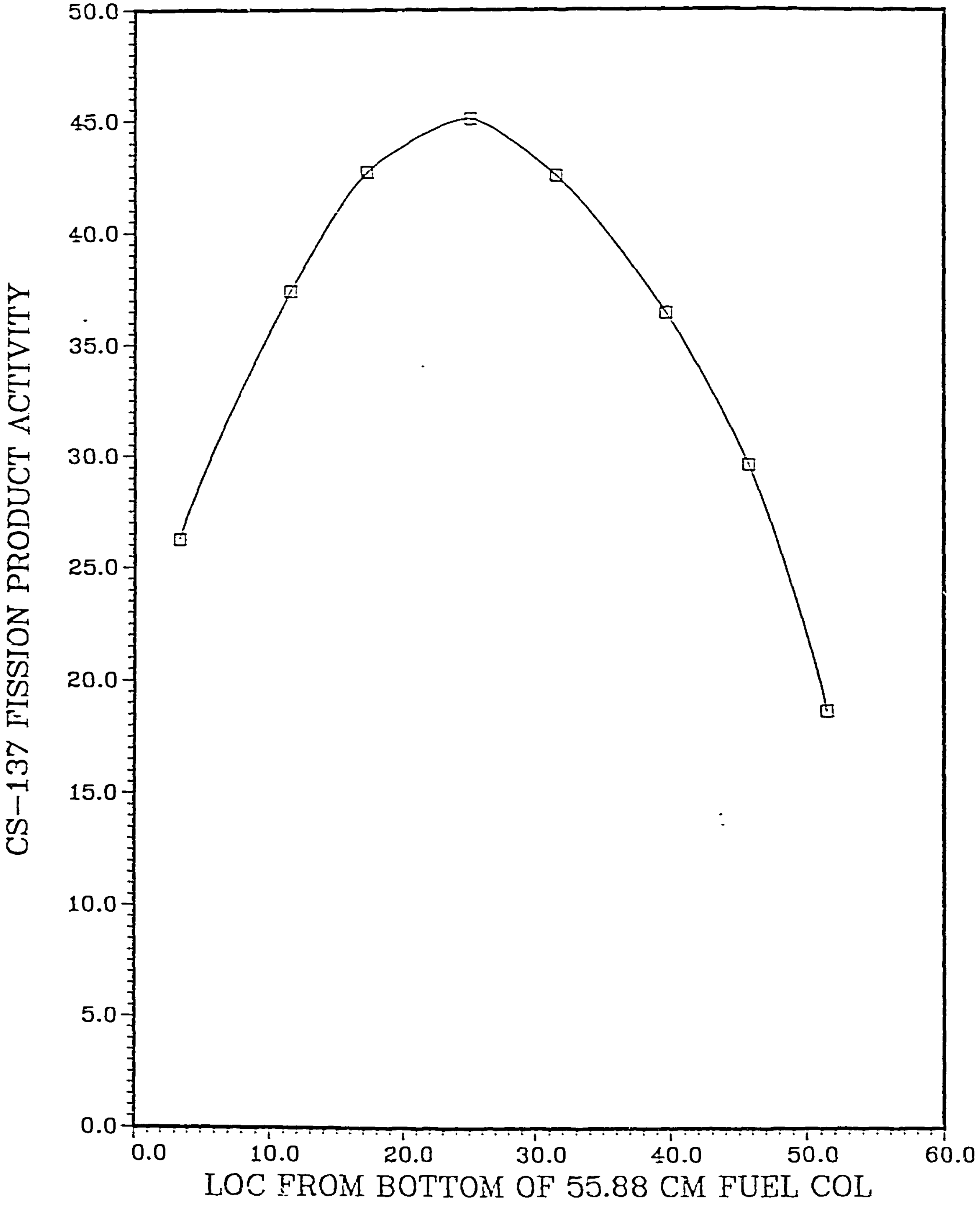
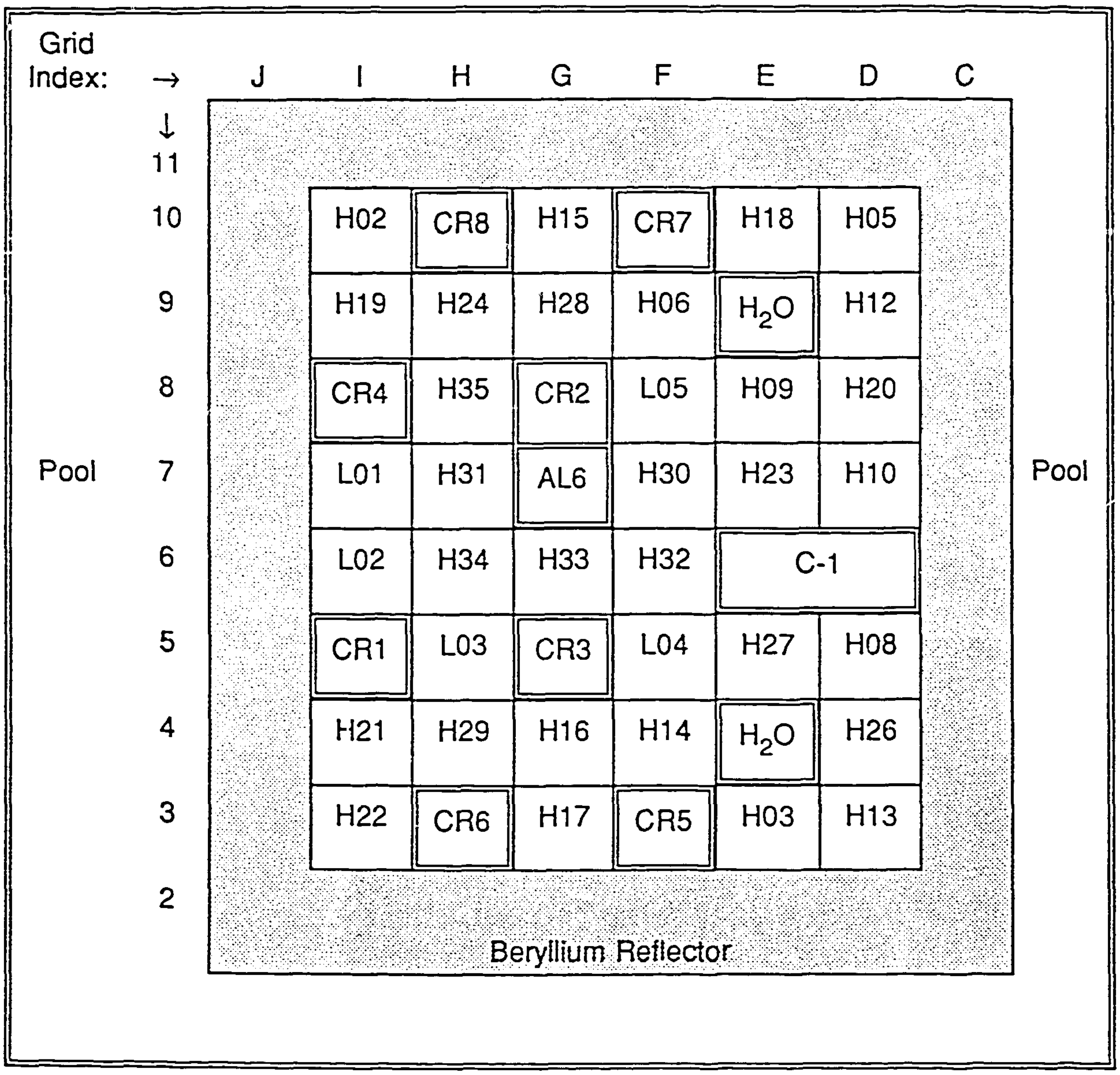


FIGURE 9

PROPOSED SSR CORE CONFIGURATION WITH 5 FRESH LEU AND 30 BURNED HEU FUEL CLUSTERS



Grid Index:	J	I	H	G	F	E	D	C
11								
10		H02	CR8	H15	CR7	H18	H05	
9		H19	H24	H28	H06	H_2O	H12	
8		CR4	H35	CR2	L05	H09	H20	
7		L01	H31	AL6	H30	H23	H10	
6		L02	H34	H33	H32	C-1		
5		CR1	L03	CR3	L04	H27	H08	
4		H21	H29	H16	H14	H_2O	H26	
3		H22	CR6	H17	CR5	H03	H13	
2								

CR = Control rod assemblies

AL6 = Experiment loop with 6 Candu fuel elements

C-1 = C1 type capsule with 1 Candu fuel element

H_2O = Water hole

L01 = LEU fuel cluster

FIGURE 10

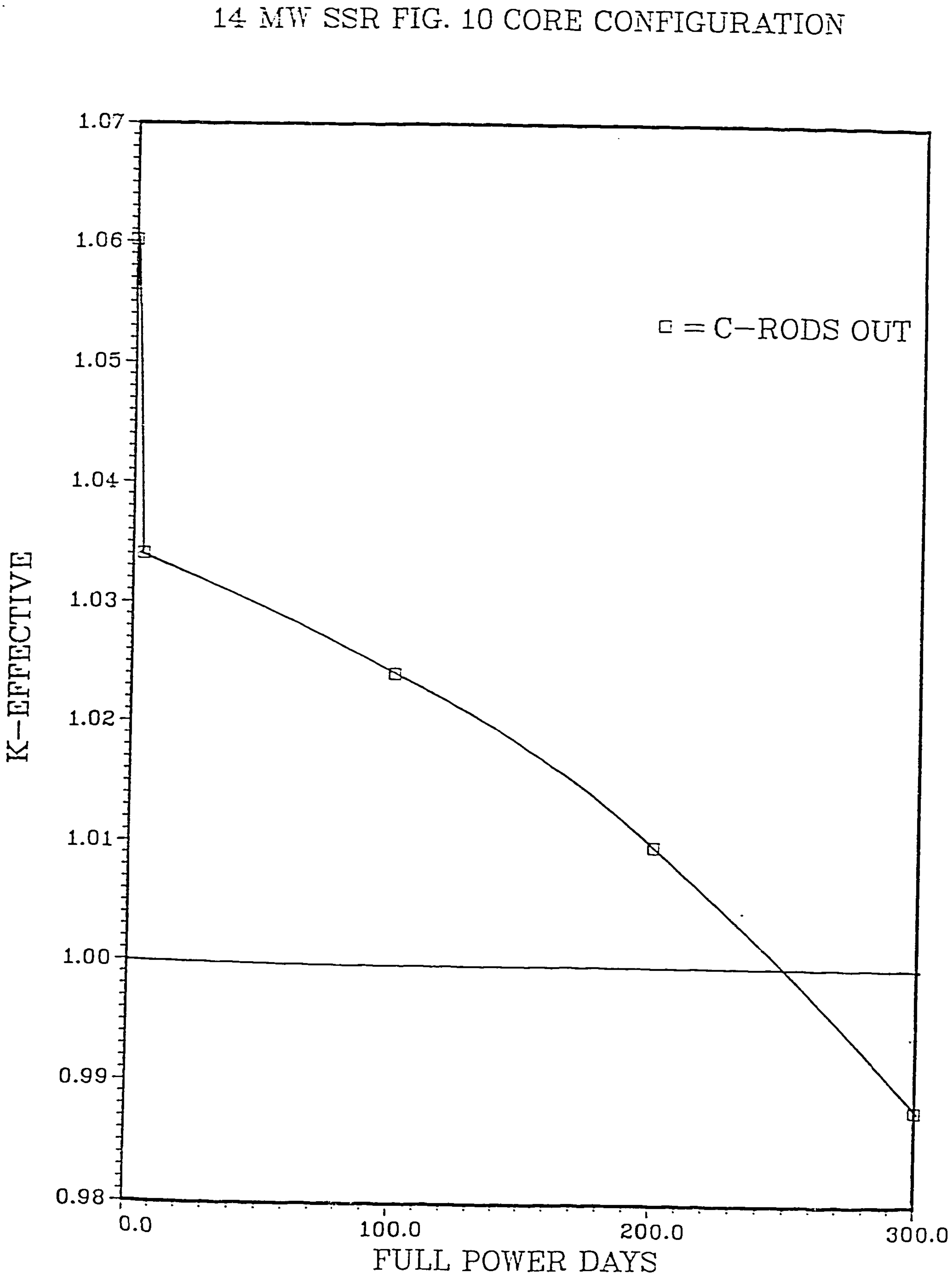
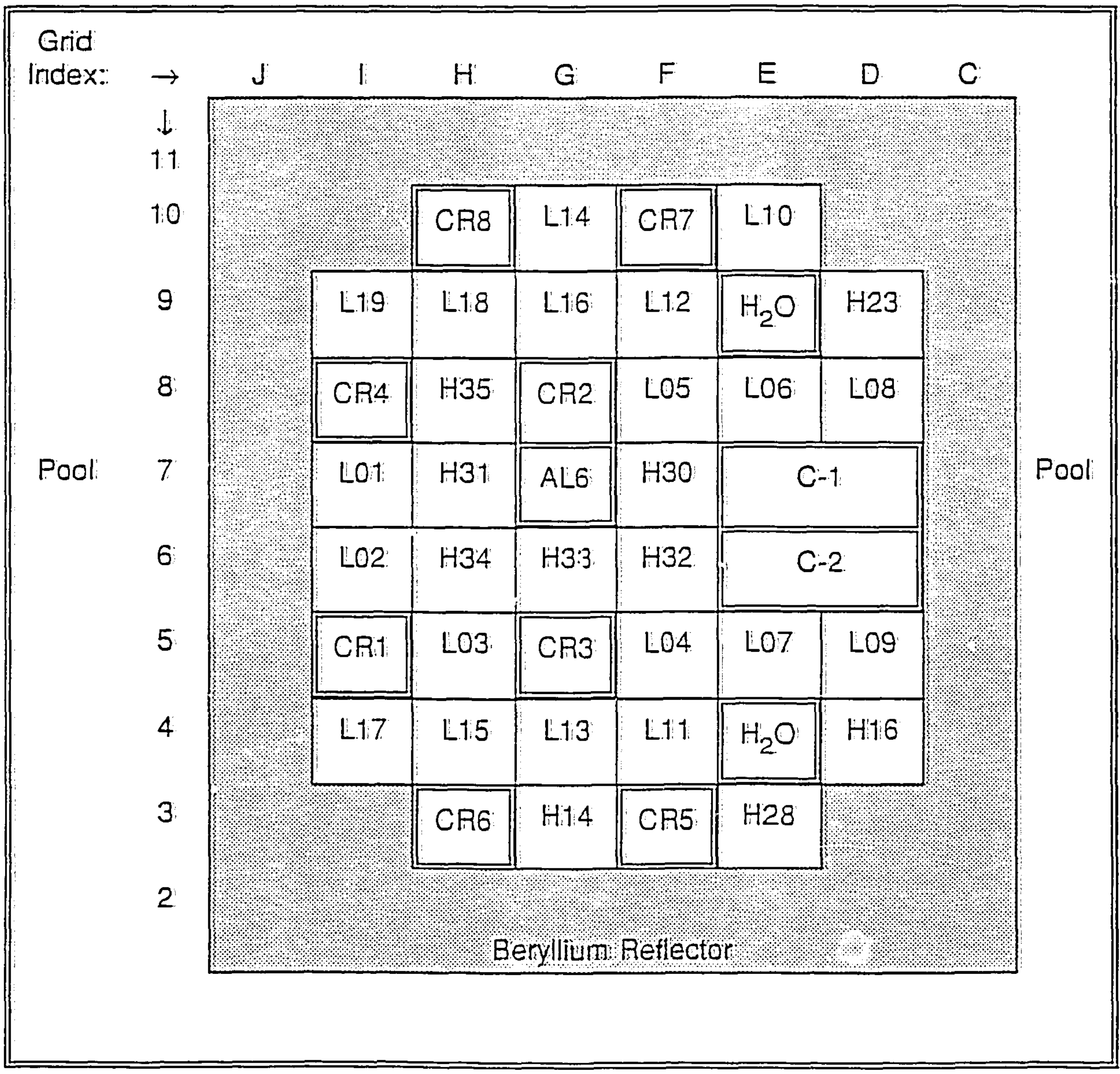

14 MW SSR FIG. 10 CORE CONFIGURATION
□ = C−RODS OUT
K−EFFECTIVE
1.07
1.06
1.05
1.04
1.03
1.02
1.01
1.00
0.99
0.98
0.0
100.0
200.0
300.0
FULL POWER DAYS

FIGURE 11

PROPOSED SSR CORE CONFIGURATION WITH 14 FRESH LEU, 5 BURNED LEU, AND 10 BURNED HEU FUEL CLUSTERS



	Grid Index: → ↓	J	I	H	G	F	E	D	C	
	11									
	10			CR8	L14	CR7	L10			
	9		L19	L18	L16	L12	H_2O	H23		
	8		CR4	H35	CR2	L05	L06	L08		
Pool	7		L01	H31	AL6	H30	C-1			Pool
	6		L02	H34	H33	H32	C-2			
	5		CR1	L03	CR3	L04	L07	L09		
	4		L17	L15	L13	L11	H_2O	H16		
	3			CR6	H14	CR5	H28			
	2									

Beryllium Reflector

CR = Control rod assemblies

AL6 = Experiment loop with 6 Candu fuel elements

C-1=C-2 = C1 type capsule with 1 Candu fuel element

H_2O = Water hole

H30 = HEU fuel cluster

L19 = LEU fuel cluster

FIGURE 12

14 MW SSR FIG. 12 CORE WITH CONTROL RODS WITHDRAWN

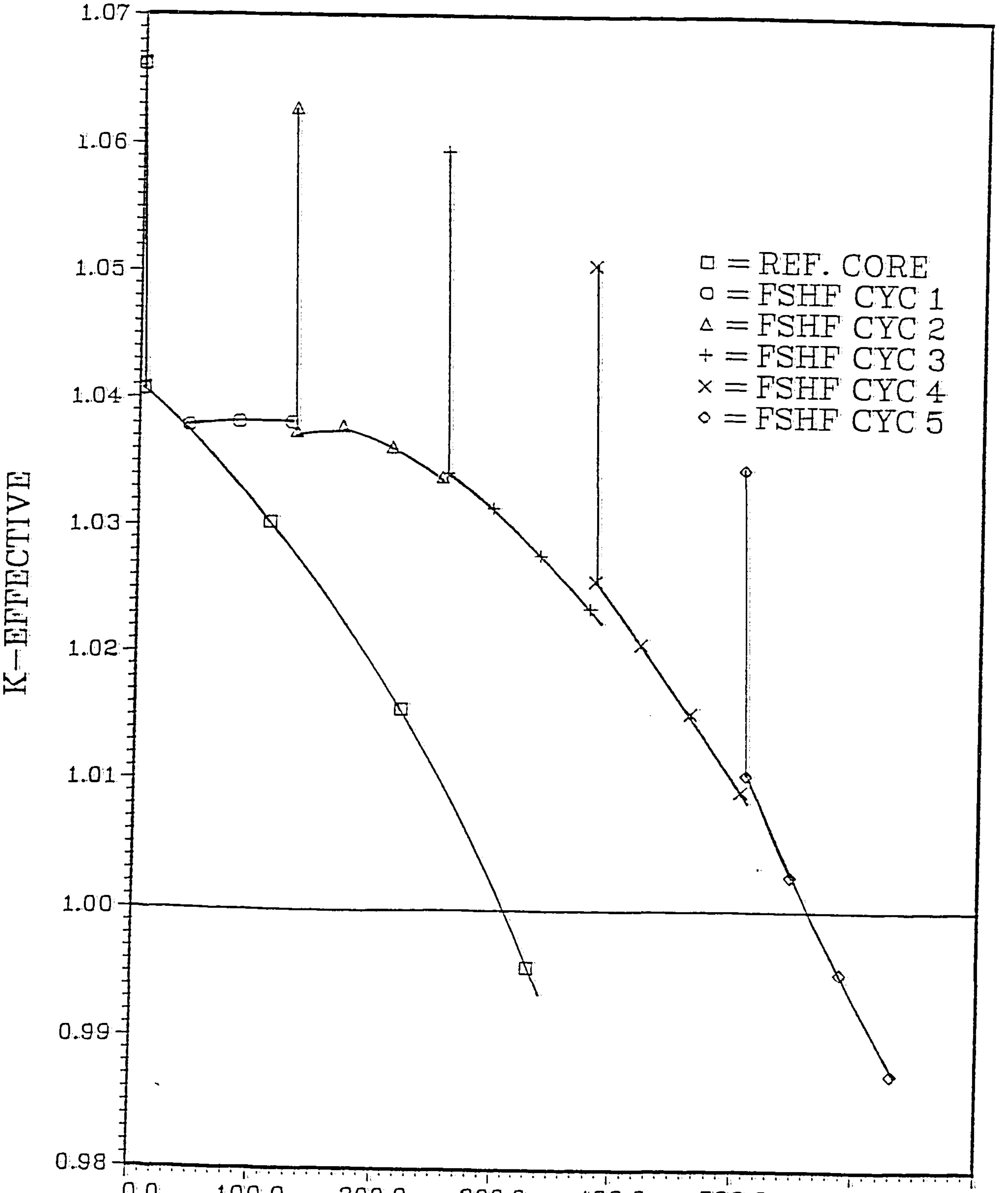


FIGURE 13