CONF-870102--14

Log Nr. 008

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CONF-870102--14

DE87 006985

MANUSCRIPT prepared for:

OPEN SESSIONS
Fourth Symposium
on Space Nuclear Power Systems
Albuquerque, New Mexico
12-16 January 1987

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Title running head: EFFECTS OF SPACE-DEPENDENT CROSS SECTIONS

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*Work supported by the U.S. Department of Energy, "Nuclear Energy Programs under Contract W-31-109-Eng-38.

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ABSTRACT

Effects of multigroup neutron cross section generation procedures on core physics parameters for compact fast spectrum reactors have been examined. Homogeneous and space-dependent multigroup cross section sets were generated in 11 and 27 groups for a representative fast reactor core. These cross sections were used to compute various reactor physics parameters for the reference core. Coarse group structure and neglect of space-dependence in the generation procedure resulted in inaccurate computations of reactor flux and power distributions and in significant errors regarding estimates of core reactivity and control system worth. Delayed neutron fraction was insensitive to cross section treatment, and computed reactivity coefficients were only slightly sensitive. However, neutron lifetime was found to be very sensitive to cross section treatment. Deficiencies in multigroup cross sections are reflected in core nuclear design and, consequently, in system mechanical design.

INTRODUCTION

Generation of accurate multigroup cross sections is a necessary aspect of nuclear reactor design. The basic measured cross section data can be processed into a very fine or essentially continuous energy structure for use in several different Monte Carlo codes, but these codes have limited applicability in many phases of reactor design. While accurate Monte Carlo calculations are essential for benchmark purposes and for certain types of analyses, there are many types of analyses for which Monte Carlo is either unsuited or unnecessary. Examples in this category are fuel cycle analyses, kinetics and safety calculations, perturbation calculations, and many routine or repetitive early and intermediate phase design calculations. For these types of calculations, the basic cross section data must be processed into the conventional multigroup format. This processing can be done with varying degrees of rigor. This paper examines the effects that multigroup cross section generation procedures can have on computed core physics parameters for compact fast spectrum space power reactors.

The reactor analyzed for this work was a small fast spectrum reactor with uranium nitride (UN) fuel in conventional pin form. The coolant was lithium; the cladding and structural material was a W-Re alloy. The core was divided into three radial enrichment zones and had an outer radius of 0.17 m. The reactor was controlled by external reflector/control drums. However, the conclusions drawn here should be valid for compact fast spectrum reactors in general. In particular, these conclusions should apply equally to cermet-fueled cores. A pin form was chosen for this analysis because cross section processing codes are structured for cells based on fuel pin geometry. Unit cell geometry for cermets is "inside out" and requires some code modifications which are presently being made.

CROSS SECTION GENERATION METHODOLOGY

Two broad group structures were used to test the effects of energy group structure on core physics. A coarse eleven group structure appropriate for preliminary and intermediate design calculations was produced. For comparison, a finer twenty-seven group cross section set appropriate for more detailed types of calculations was also produced.

Two different generation procedures were used here. All cross sections were processed through the MC^2-2 (Henryson et. al., 1976) and SDX (Toppel et. al., 1978) code packages. The basic cross section treatment was based on a fundamental mode calculation using MC^2-2. In its simplest form, MC^2-2 performs an ultra-fine (approximately 2000) group calculation for a homogeneous composition using either the fission spectrum for a particular isotope, the spectrum resulting from the mixture of materials being computed (provided that fissionable materials are present in the mixture), or an imposed spectrum input by the user. For these calculations, the spectrum resulting from the mixture of materials was used when possible; for non-fissionable compositions, the U-235 spectrum was used. A group-independent buckling search to critical is then performed, and the cross sections are then collapsed to the desired broad group structure. These calculations were performed in the consistent P-1 approximation with improved Greuling-Goertzel continuous slowing down parameters. This procedure was used to generate multigroup cross section sets adequate for preliminary scoping analyses.

The adequacy of cross sections resulting from this procedure depends on how well the single composition and spectrum used represent the core as a whole. For compact fast spectrum reactors, particularly those controlled by external reflection or control drums, core conditions may exhibit substantial spatial variations. Spatial effects were included by processing cross sections through the SDX code. First, MC^2-2 was used as described above for each core and reflector spatial region (or composition) to generate an intermediate group (230 groups) cross section library containing smooth cross sections but excluding resonance cross sections. SDX was used to compute heavy and intermediate element resonance cross sections appropriate to the composition, unit cell structure, and temperature of each region, including a resonance heterogeneity treatment within the fuel pins. These resonance cross sections were merged onto the base library. Finally, a one-dimensional multigroup diffusion calculation was performed for the reactor at the 230 group level, and the resulting fluxes were used to do the final collapsing to produce space-dependent broad group cross sections in the eleven and twenty-seven group structures.

CALCULATIONAL RESULTS

These multigroup cross section sets were used in a number of RZ diffusion and one-dimensional transport calculations with DIF3D (Derstine, 1984) and ONEDANT (O'Dell et. al., 1982) for the reference configuration described above. First, the effects of higher order scattering were examined. the reference configuration was computed with ONEDANT for cases with the control drum poison segment rotated outward as far from the fuel as possible and inward as close to the fuel as possible. These cases were computed with P-1 and P-0 cross sections. For the case with the poison segment rotated outward, the difference between the P-1 and P-0 eigenvalues was 0.0007 Δk; the corresponding difference with the poison segment rotated inward was -0.0028 Δk. This indicates that higher order scattering has only a minor effect in the present core configuration which is confirmed by comparing the computed powers for corresponding configurations in the P-1 and P-0 cases. The computed powers in these cases are nearly identical as one would expect from the close eigenvalue agreement. Because higher order scattering seems to be a small effect in the present context, all subsequent calculations were performed with P-0 cross sections.

Table 1 lists key core parameters computed with the various cross section sets. Both spatial dependence and group structure have significant effects on core reactivity. At the 27-group level, the change from homogeneous to space-dependent cross sections is worth 0.0075 Δk or $1.11; the corresponding change at the 11-group level is worth -0.0116 Δk or -$1.72. For space-dependent cross sections, the transition from 11 to 27 groups is worth 0.0013 Δk or $0.19 while the corresponding change for homogeneous cross sections is worth -0.0178 Δk or -$2.64.

Table 1 also shows the effects of different cross section sets on computed control drum worth. At the 27-group level, the change from homogeneous to space-dependent cross sections changes computed drum worth by 0.0043 Δk or $0.64. For the 11-group cases, this transition reduces drum worth by 0.0031 Δk or $0.46. While these differences are not large in the absolute sense, they could become significant if available control margins are narrow.

Inaccurate calculation of core reactivity and control system worth affect system design in several ways. Beginning-of-life (BOL) and end-of-life (EOL) core reactivity and the resultant burnup reactivity swing are major factors governing control system design. Inaccurate computation of these parameters results in an increased excess reactivity margin to cover uncertainties. An excessive reactivity margin places additional demands on the control system and may result in the need for a secondary control system. In addition, inaccurate calculation of these reactivity parameters may affect the expected operational lifetime of the system. Finally, inaccurate computations of BOL k, EOL k, and depletion will result in inaccurate computation of reactivity coefficients at EOL even if the BOL values are reasonably accurate.

The computed core power distribution is an important factor in core nuclear and mechanical design. Figure 1 shows radial power profiles computed with the four cross section sets used to determine Table 1. These profiles are for the configuration with the control drum poison segment rotated outward, away from the fuel. Figure 1 shows that space-dependence and group structure have little effect in the inner part of the core. In the outer part of the core, however, the effects of spatial dependence and group structure are apparent in Figure 1. When the poison segment is inside, next to the core, the profiles for all four cross section sets are nearly indistinguishable because the strong poison absorption dominates the region where space-dependence and group structure are most important in this reactor.

Power peaking is an important factor in core nuclear and mechanical design because the degree of peaking governs peak fuel temperature and burnup and such factors as thermal stresses in the fuel. For present purposes, power peaking is defined as peak power divided by the core centerline power. Table 1 and Figure 1 show power peaking factors for each of the cross section sets. Space-dependent cross sections result in higher computed peaking factors than homogeneous sets produce, and finer group structure results in higher peaks than those computed with a coarser group structure.

Table 2 shows computed reactivity coefficients for the two space-dependent cross section sets. The differences shown in Table 2 are not likely to have a significant effect on kinetics or safety calculations. These differences are smaller than one would expect based on experience in conventional LMR design and analysis, and the reason for this discrepancy is being investigated.

Delayed neutron fractions are shown in Table 1 for the four cross section sets. These delayed neutron fractions are insensitive to cross section treatment for two reasons. First, the amount of delayed neutron data available is quite limited, and this scarcity of data is reflected in the calculation. Second, delayed neutrons are emitted by specific fission fragment nuclei. Each fissionable isotope has a characteristic distribution of these nuclei among its fission products. Because reactor fuel consists of a particular mixture of the fissionable isotopes, a given fuel composition also has some characteristic fission fragment distribution. This distribution and the resulting delayed neutron fraction vary only to the extent that they are sensitive to the energy of the fissioning neutron.

The last column in Table 1 shows computed generation times for the reference core using the various cross section sets. Differences in computed generation times are much more dramatic than any other differences in this table. The order-of-magnitude changes shown here can be very important to reactor kinetics, control, and safety during certain severe transients.

Differences in generation time shown here result from more accurate treatment of the lower energy range in the space-dependent set and in the finer group structure. The contribution of an energy group to generation time is proportional to the product of the flux and adjoint flux in the group divided by the average velocity in the group. The inverse dependence on neutron velocity emphasizes the contribution of the lower energy range. Finer group structure results in more accurate computation of the flux and adjoint. In the present cases, this is particularly true of the lower energy range.

Space-dependent cross sections more accurately reflect varying conditions in different core regions than do homogeneous cross sections. In a large conventional reactor, core conditions are at least reasonably uniform over large regions. In compact fast spectrum space reactors, especially those controlled by drums or other reflector control, core composition and flux spectrum can change significantly over the small distance from the core centerline to the outer boundary of the reflector. Homogeneous cross sections only represent conditions at one point; space-dependent cross sections can represent the varying conditions in each core region. In addition, space-dependent cross sections provide a more accurate treatment of leakage and spectral effects than is possible with the fundamental mode approximation.

Figures 2 and 3 show the product of the flux and adjoint divided by the average neutron velocity in each energy group for the inner and outer core regions. There is a clear spectral shift between the inner and outer core. In the outer core, the lower energy groups make the most important contribution to generation time. The contributions of the lower energy range and the outer portion of the core are emphasized because computation of generation time involves a volume integral; the outer portion of the core and the reflector have much larger volumes than does the inner core. Spatially, the outermost fuel zone and the reflector make the most important contributions to generation time, and it is precisely these regions which most require space-dependent cross sections rather than a fundamental mode treatment.

CONCLUSIONS

Cross section generation procedures affect a number of core physics parameters. The order of scattering had only a small effect in the cases considered here, but higher order scattering might show a larger effect in another configuration. The difference of -0.0028 Δk between P-1 and P-0 with the poison segment rotated outward is on the borderline of significance for some situations.

Cross section generation procedures affect computation of reactivity and control worth. Transition from a coarse group structure to a finer one can change computed reactivity by \$1 or more in either direction; transition from homogeneous to space-dependent cross sections has similar worth. Spatial effects also changed computed control drum worth by \$0.64. Changes of this magnitude are relevant to reactor control and to core design because reactivity requirements are one governing factor for core size, control system design, and core mass.

Over most of the core, group structure and spatial treatment had little effect on power shape. However, near the outer fuel boundary where the composition and spectrum change sharply, cross section treatment has a large (15%-30%) effect on power peaking. Cross section treatment influences core mechanical design through its effect on power shape. Radial and axial power distributions are governing factors for heat transfer design considerations. Power peaking determines peak fuel temperature and burnup which in turn is one determinant of core size. Power gradients determine the severity of bowing and stress which also affect core lifetime. Coarse group structure and neglect of space dependence can result in inaccurate data for mechanical design purposes and, ultimately, to poor core mechanical design.

Cross section treatment has a strong effect on kinetics and safety-related parameters. The reactivity coefficients shown in Table 2 were only slightly affected. The delayed neutron fraction is a fundamental characteristic of the fuel and is not likely to be affected by the cross section treatment.

However, neutron generation time is very strongly affected. Finer group structure provides more accurate estimation of the flux and adjoint. An accurate calculation of the flux and adjoint is particularly important in the lower energy range because the inverse dependence of generation time on neutron velocity emphasizes the contribution of the lower energy range to generation time. Inclusion of spatial effects in cross sections more accurately reflects overall core conditions than does a fundamental mode treatment. Spatial effects are particularly important in the type of core considered here because its small size and sharp composition and spectral changes near the outer outer core boundary make fundamental mode approximations particularly bad here.

Inadequate neutron cross sections result in inaccurate computations of reactor power and flux distributions and in large uncertainties regarding reactivity estimates. Calculation of important safety parameters such as reactivity feedback coefficients and neutron lifetime can be highly sensitive to the cross sections used in calculations. Furthermore, because core nuclear and mechanical designs are closely interrelated, the deficiencies in multigroup cross sections used for nuclear design can be reflected in system mechanical design.

Acknowledgments

This work was performed at Argonne National Laboratory, supported by the U.S. Department of Energy, Nuclear Energy Programs under contract W-31-109-Eng-38.

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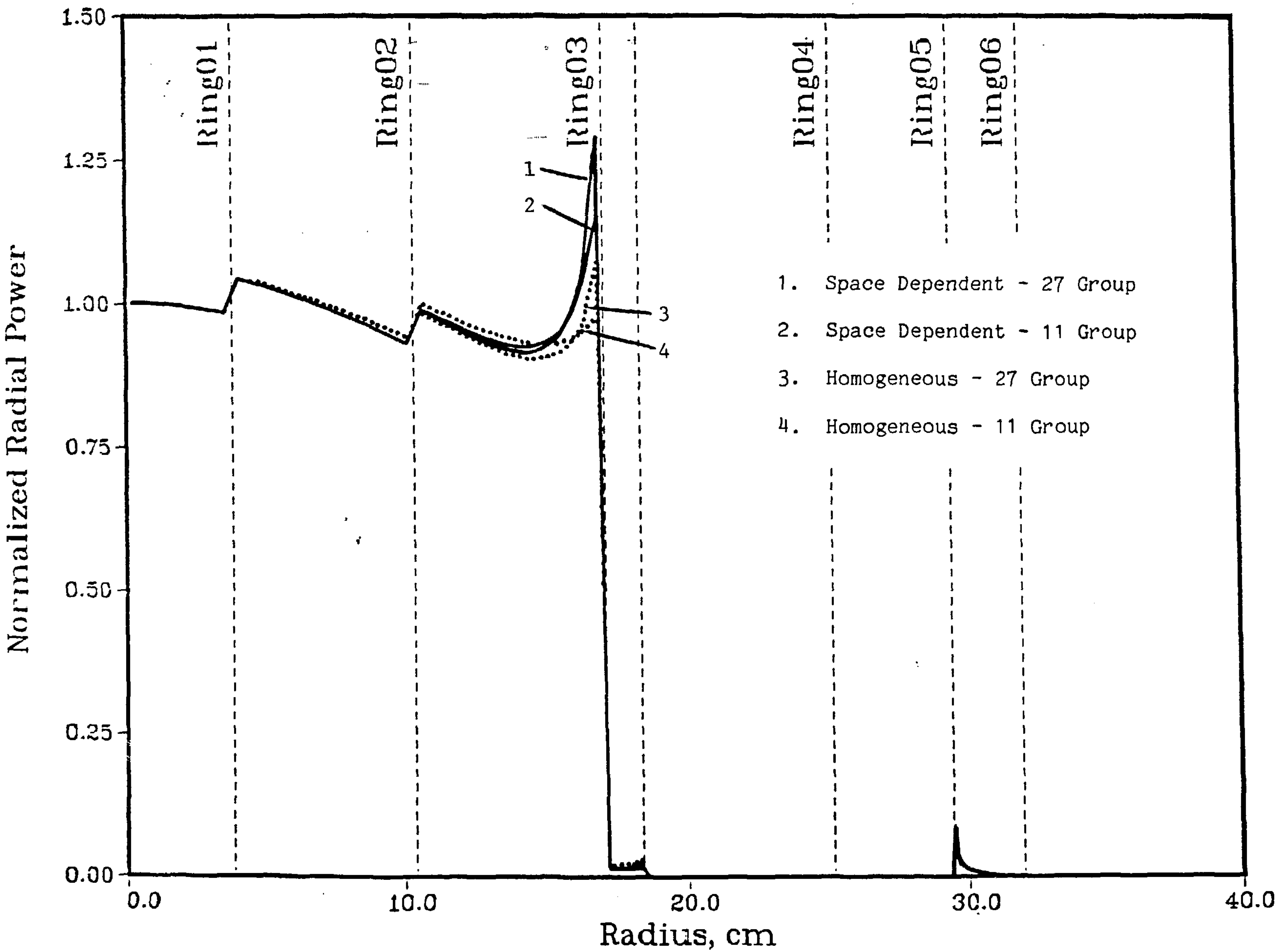
Table 1. Core Physics Parameters as a Function of Cross Generation Procedure

Case	Cross Section Procedure	Group Structure	k	Control Drum Worth, Δk	Power Peaking Factor	Delayed Neutron Fraction, β	Generation Time, μs
1	Space Dependent	27	1.1238	0.1095	1.359	0.00676	2.465
2	Space Dependent	11	1.1225	0.1095	1.214	0.00674	0.338
3	Homogeneous	27	1.1163	0.1052	1.188	0.00676	0.307
4	Homogeneous	11	1.1341	0.1126	1.017	0.00675	0.084

Table 2. Reactivity Coefficients as a Function of Group Structure for Space-Dependent Cross Sections

	11-Group	27-Group
Fuel Doppler ($/K)	0.000207	0.000200
Coolant ($/K)	0.000144	0.000141
Fuel Radial Expansion ($/K)	0.000339	0.000341
Fuel Axial Expansion ($/K)	0.000225	0.000222

FIGURE 1. Effects of Cross Section Generation Procedure on Radial Power Distribution.



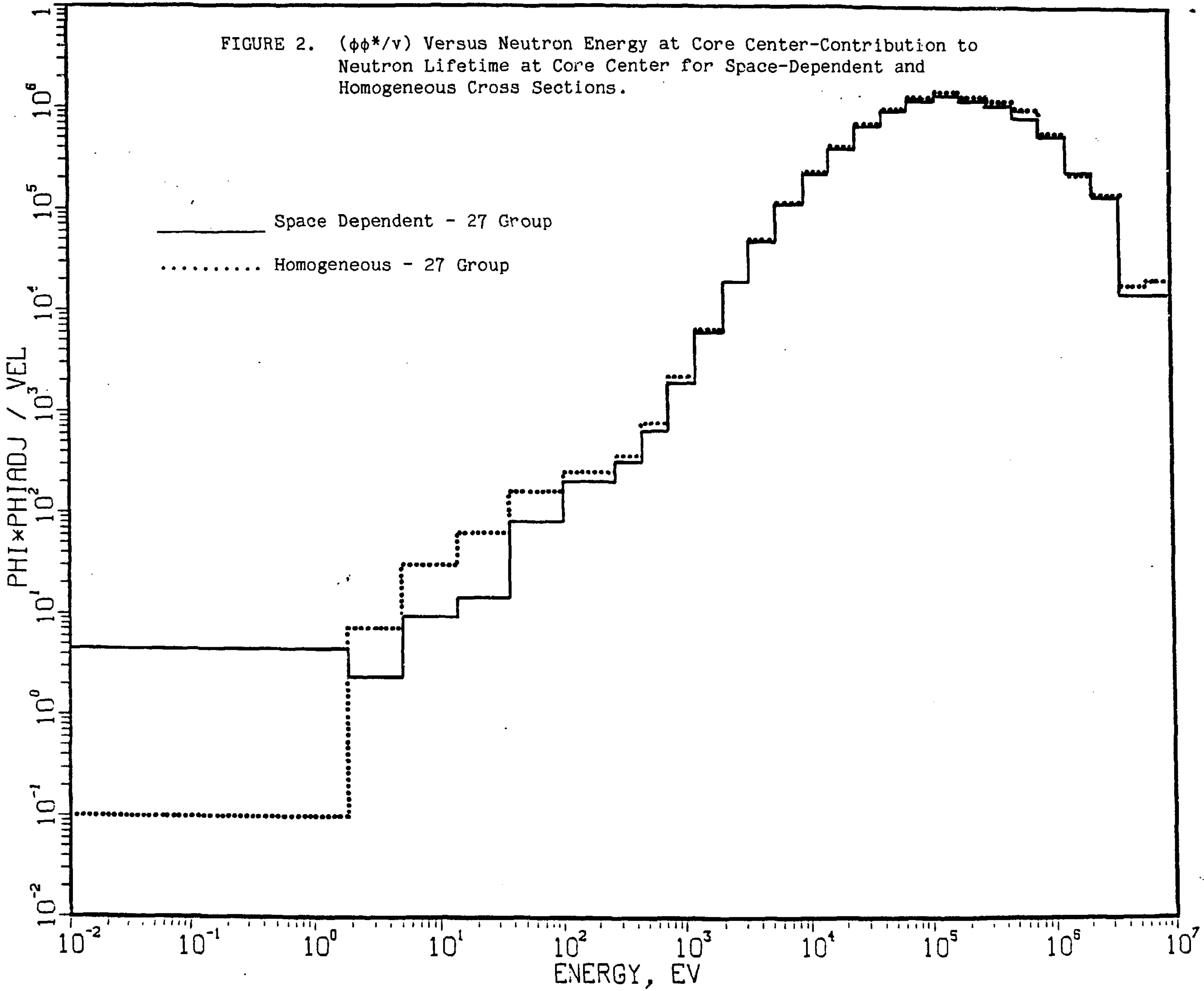
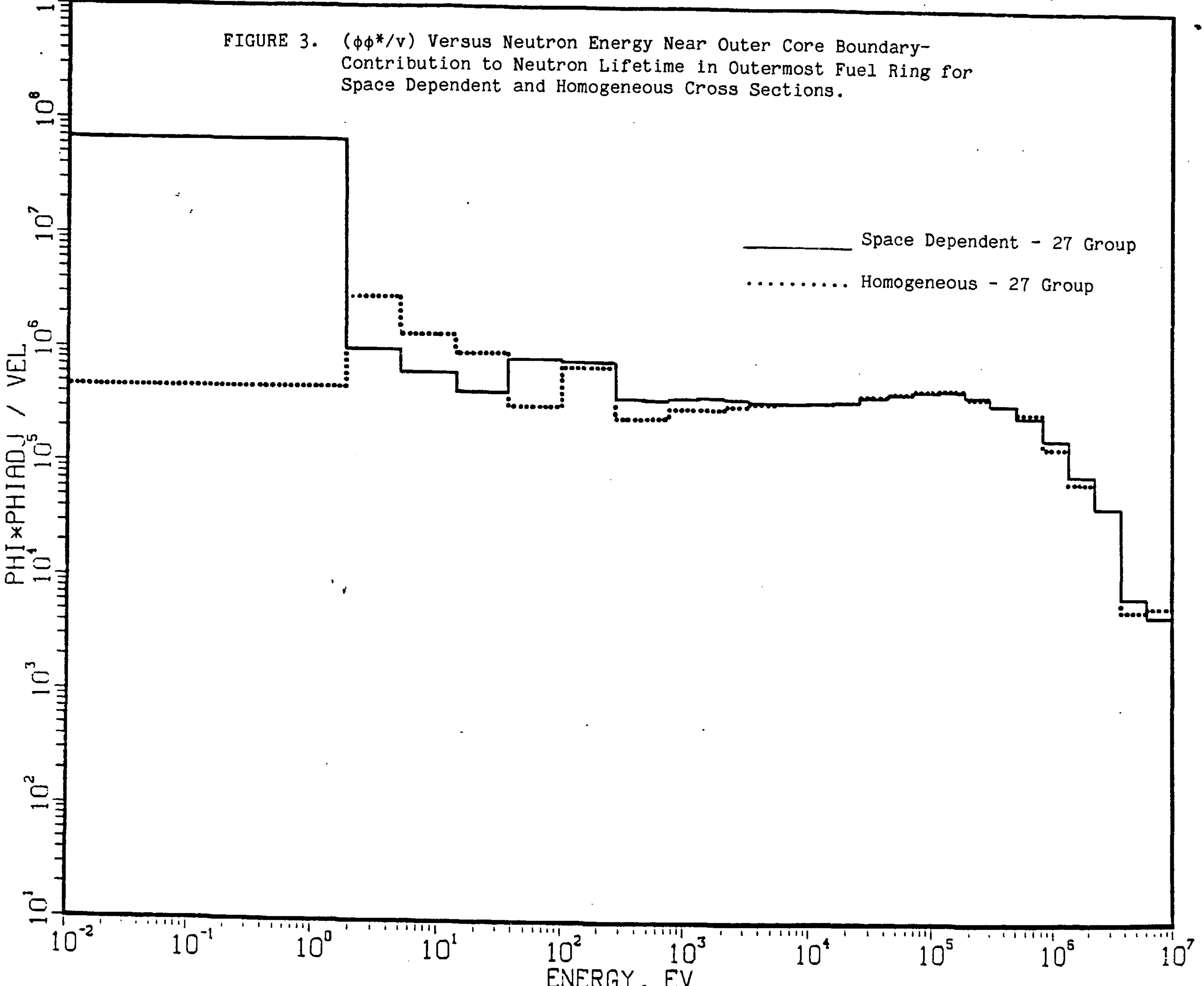


FIGURE 2. (ϕϕ*/v) Versus Neutron Energy at Core Center-Contribution to Neutron Lifetime at Core Center for Space-Dependent and Homogeneous Cross Sections.
Space Dependent - 27 Group
Homogeneous - 27 Group
PHI*PHIADJ / VEL
ENERGY, EV
10^{-2}
10^{-1}
10^{0}
10^{1}
10^{2}
10^{3}
10^{4}
10^{5}
10^{6}
10^{7}
1

FIGURE 3. $(\phi\phi^*/v)$ Versus Neutron Energy Near Outer Core Boundary-Contribution to Neutron Lifetime in Outermost Fuel Ring for Space Dependent and Homogeneous Cross Sections.



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