PROJECTION OF GRAPHITE BEHAVIOR
IN VERSION III
OF THE ORNL CONCEPTUAL FUSION REACTOR

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by

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Ceramics and Graphite Section
Fuels and Materials Department

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INTRODUCTION

With the prospect that scientific feasibility of power from nuclear fusion will be demonstrated within the next few years, a number of groups of scientists in the United States have begun to examine the problem of how an actual power-producing reactor would be built. The purpose of this conceptual design work is to bring much more clearly into focus the many engineering problems which will be encountered, to determine what engineering and material information is required that is not currently available, and to see where the design and economic trade-offs lie.

Most conceptual fusion reactor designs include graphite in the blanket and/or shield regions. Its purpose is to moderate and reflect the high-energy neutrons back into the tritium breeding portion of the blanket and/or to moderate the high-energy neutrons so that they can be more readily absorbed before penetrating into the superconducting magnet region. The graphite behavior in these conceptual designs is being evaluated by Battelle-Northwest under programs supported by the Controlled Thermonuclear Research Division of the AEC and PNL special studies funds. While we have been involved to only a very limited extent in conceptual fusion reactor design work per se, we do have many years experience studying radiation effects in graphite in the fission reactor programs and have been actively involved in national and international efforts to relate radiation damage rates in graphite between the various neutron energy spectra in the different reactor types.

A brief review describing the various types of graphite that can be obtained, radiation effects in graphite in general, and a comparison of the fusion reactors we have evaluated will be issued separately.⁽¹⁾ The present

paper covers our analysis of the graphite performance in Version III of the ORNL conceptual fusion reactor⁽²⁾ in more detail.

BACKGROUND

A general review of the effects of high-energy neutron irradiation on graphite will be published elsewhere.⁽¹⁾ Briefly, large changes in dimensions, strength, elastic modulus, thermal conductivity, thermal expansion, and other less important physical properties occur at rates and magnitudes which are dependent on the high-energy neutron fluence, the neutron energy spectrum, irradiation temperature, and the type of graphite. Of these, the dimensional changes are probably the most important for both fission and fusion reactor applications and these are the changes that are considered in the most detail here.

In addition to the property changes mentioned above, graphite undergoes large irradiation-induced creep at temperatures far below those ($\sim 2000^\circ\text{C}$) where thermal creep becomes appreciable. This radiation creep occurs in two stages: 1) Transient creep which occurs rapidly but soon saturates at a value about equal to the elastic strain; and 2) steady-state creep which is linear with fluence and stress at least up to fluences of about 7×10^{21} n/cm².⁽³⁾ Radiation creep is important in reactor design considerations because it serves to relieve stresses induced by thermal gradients and by fluence gradients which give rise to gradients in the radiation-induced dimensional change rates. As a corollary, the dimensional changes can be altered by stresses and the accompanying radiation creep.

REACTOR DESIGN

Design details for Version III of the ORNL conceptual fusion reactor have been given by Fraas⁽²⁾ and Figures 1 and 2 have been reproduced here with his permission. Figure 1 shows an overall view of the reactor while Figure 2 shows a cross sectional view through one of the lithium/graphite segments. Other pertinent design data are given in Table I.

TABLE I

ORNL CONCEPTUAL CTR, VERSION III - DESIGN DATA

TYPE:	Pulsed, Low-Beta, Torodial, Tokamak-Type
MINOR RADIUS:	3.5 meters
MAJOR RADIUS:	10.5 meters
SURFACE AREA OF FIRST WALL:	1450 m ²
WALL LOADING:	0.43 MW/m ² (14.1 MeV Neutrons)
FIRST WALL:	0.25 cm Niobium
POWER:	1000 MW(t), 560 MW(e)
OPERATING CYCLE:	172 sec, 86% on-time
BLANKET:	Lithium/Graphite
GRAPHITE DENSITY:	8.04×10^{22} atoms/cm ³ (1.6 g/cm ³)

NEUTRONICS

Normalization Factors

John Kriese of ORNL kindly furnished a complete punched card listing of the fluxes, by energy group and space zone, from a S_{12} - P_3 ANISN calculation.^(4,5) The neutron cross sections used in these calculations were processed from ENDF/B-III in the 100 energy group structure shown in Table II. The space zone schematic, shown in Table III, represents a radial cross section of the cylindrical geometry used for the ANISN calculations. The zones are evenly spaced within any given region but are not necessarily the same spacing in the different regions.

The source conditions for the ANISN calculations was one neutron at 14 MeV uniformly distributed throughout zone 1, the plasma region; this is equivalent to 1 fusion event per sec. per cm. of length measured along the major circumference of the torus. Using the accepted definition for the neutron wall loading ($1 \text{ MW/m}^2 \equiv 4.43 \times 10^{13} \text{ n/m}^2 \text{ sec}$)*, we have renormalized the ANISN results to a wall loading of 1 MW/m^2 for comparison with other fusion reactors we are evaluating^{6,7,8)} and, using the total thermal energy generation per fusion event, we have also renormalized the results to the design power of 1000 MW (thermal); the renormalization procedures are illustrated in Table IV.

Neutron Fluence Calculations

Projections of graphite behavior in different fusion reactor designs must currently be made on the basis of data accumulated from irradiation tests in fission reactors. Data on the radiation-induced changes in nuclear graphites, accumulated from irradiation tests in the fission reactors, are currently being converted to an internationally consistent fluence scale,** "equivalent fission fluence for damage in graphite" (Φ_G), which is defined by:

* $(4.43 \times 10^{13} \text{ n/cm}^2 \text{ sec}) (14.06 \text{ MeV/n}) (1.602 \times 10^{-19} \text{ MW sec/MeV}) (10^4 \text{ cm}^2/\text{m}^2) \equiv 1 \text{ MW/m}^2$

** Recommended by the delegates (representing the I.A.E.A., EURATOM, U.S.A.E.C. and individual laboratories from six countries) attending the "Specialist's Meeting on Radiation Damage in Graphite", which was held on October 30, 1972, at the Battelle Seattle Research Center under the cosponsorship of the International Atomic Energy Agency and Battelle Memorial Institute.

TABLE II. NEUTRON ENERGY AND LETHARGY LIMITS FOR 100 GROUP STRUCTURE
(ENERGY LIMITS ARE GIVEN IN ELECTRON VOLTS)

GROUP NUMBER	UPPER ENERGY	LOWER ENERGY	LOWER LETHARGY	GROUP NUMBER	UPPER ENERGY	LOWER ENERGY	LOWER LETHARGY
1	1.4918+07*	1.3499+07	0.10	51	8.6517+04	6.7379+04	5.40
2	1.3499+07	1.2214+07	0.20	52	6.7379+04	5.2475+04	5.65
3	1.2214+07	1.1052+07	0.30	53	5.2475+04	4.0868+04	5.90
4	1.1052+07	1.0000+07	0.40	54	4.0868+04	3.1823+04	6.15
5	1.0000+07	9.0484+06	0.50	55	3.1828+04	2.4788+04	6.40
6	9.0484+06	8.1873+06	0.60	56	2.4788+04	1.9305+04	6.65
7	8.1873+06	7.4082+06	0.70	57	1.9305+04	1.5034+04	6.90
8	7.4082+06	6.7032+06	0.80	58	1.5034+04	1.1709+04	7.15
9	6.7032+06	6.0653+06	0.90	59	1.1709+04	9.1188+03	7.40
10	6.0653+06	5.4881+06	1.00	60	9.1188+03	7.1017+03	7.65
11	5.4881+06	4.9659+06	1.10	61	7.1017+03	5.5308+03	7.90
12	4.9659+06	4.4933+06	1.20	62	5.5308+03	4.3074+03	8.15
13	4.4933+06	4.0657+06	1.30	63	4.3074+03	3.3546+03	8.40
14	4.0657+06	3.6788+06	1.40	64	3.3546+03	2.6126+03	8.65
15	3.6788+06	3.3287+06	1.50	65	2.6126+03	2.0347+03	8.90
16	3.3287+06	3.0119+06	1.60	66	2.0347+03	1.5846+03	9.15
17	3.0119+06	2.7253+06	1.70	67	1.5846+03	1.2341+03	9.40
18	2.7253+06	2.4660+06	1.80	68	1.2341+03	9.6112+02	9.65
19	2.4660+06	2.2313+06	1.90	69	9.6112+02	7.4852+02	9.90
20	2.2313+06	2.0190+06	2.00	70	7.4852+02	5.8295+02	10.15
21	2.0190+06	1.8268+06	2.10	71	5.8295+02	4.5400+02	10.40
22	1.8268+06	1.6530+06	2.20	72	4.5400+02	3.5358+02	10.65
23	1.6530+06	1.4957+06	2.30	73	3.5358+02	2.7536+02	10.90
24	1.4957+06	1.3534+06	2.40	74	2.7536+02	2.1445+02	11.15
25	1.3534+06	1.2246+06	2.50	75	2.1445+02	1.6702+02	11.40
26	1.2246+06	1.1080+06	2.60	76	1.6702+02	1.3007+02	11.65
27	1.1080+06	1.0026+06	2.70	77	1.3007+02	1.0130+02	11.90
28	1.0026+06	9.0718+05	2.80	78	1.0130+02	7.8893+01	12.15
29	9.0718+05	8.2085+05	2.90	79	7.8893+01	6.1442+01	12.40
30	8.2085+05	7.4274+05	3.00	80	6.1442+01	4.7851+01	12.65
31	7.4274+05	6.7206+05	3.10	81	4.7851+01	3.7267+01	12.90
32	6.7206+05	6.0810+05	3.20	82	3.7267+01	2.9023+01	13.15
33	6.0810+05	5.5023+05	3.30	83	2.9023+01	2.2603+01	13.40
34	5.5023+05	4.9787+05	3.40	84	2.2603+01	1.7603+01	13.65
35	4.9787+05	4.5049+05	3.50	85	1.7603+01	1.3710+01	13.90
36	4.5049+05	4.0762+05	3.60	86	1.3710+01	1.0677+01	14.15
37	4.0762+05	3.6883+05	3.70	87	1.0677+01	8.3153+00	14.40
38	3.6883+05	3.3373+05	3.80	88	8.3153+00	6.4760+00	14.65
39	3.3373+05	3.0197+05	3.90	89	6.4760+00	5.0435+00	14.90
40	3.0197+05	2.7324+05	4.00	90	5.0435+00	3.9279+00	15.15
41	2.7324+05	2.4724+05	4.10	91	3.9279+00	3.0590+00	15.40
42	2.4724+05	2.2371+05	4.20	92	3.0590+00	2.3824+00	15.65
43	2.2371+05	2.0242+05	4.30	93	2.3824+00	1.8554+00	15.90
44	2.0242+05	1.8316+05	4.40	94	1.8554+00	1.4450+00	16.15
45	1.8316+05	1.6573+05	4.50	95	1.4450+00	1.1254+00	16.40
46	1.6573+05	1.4966+05	4.60	96	1.1254+00	8.7643-01	16.65
47	1.4966+05	1.3569+05	4.70	97	8.7643-01	6.8256-01	16.90
48	1.3569+05	1.2277+05	4.80	98	6.8256-01	5.3158-01	17.15
49	1.2277+05	1.1109+05	4.90	99	5.3158-01	4.1399-01	17.40
50	1.1109+05	8.6517+04	5.15	100	4.1399-01	(THERMAL)	*****

* ±N REPRESENTS 10^{±N}

TABLE III

SPACE ZONE SCHEMATIC

<u>Region</u>	<u>Material</u>	<u>Zone No.</u>	<u>Outer Radius (cm)</u>
Plasma	D-T Plasma	1	280
Vacuum	Vacuum	2	350
First Wall	Nb	3-5	350.25
First Li Blanket	99% Li 1% Nb	6-35	380.25
Structure	Nb	36-38	380.5
Graphite Moderator	Graphite	39-58	420.5
Structure	Nb	59-61	420.75
Second Li Blanket	99% Li 1% Nb	62-76	450.75
Structure	Nb	79-79	451.0

TABLE IV: FLUX NORMALIZATION

RESULTS NORMALIZED TO 1 MW/m²

Average neutron generation rate
per cm of length along the major
circumference of the torus

$$: (4.43 \times 10^{13} \text{ n/cm}^2\text{sec}) (2\pi) (350 \text{ cm}) = 9.742 \times 10^{16} \text{ n/cm sec.}$$

Average fusion rate in plasma
region

$$: (9.742 \times 10^{16} \text{ n/cm sec}) (2\pi) (1050 \text{ cm}) = 6.43 \times 10^{20} \text{ n/sec.}$$

Total thermal energy generation
in CTR**

$$: (6.43 \times 10^{20} \text{ n/sec}) (22.4 \text{ MeV/n}) (1.602 \times 10^{-19} \text{ MW sec/MeV}) = 2,306 \text{ MW(t).}$$

Lethargy group #1 flux at
first wall

$$: 7.92 \times 10^{13} \text{ n/cm}^2\text{sec.}$$

Lethargy group #1 flux at
inner surface of graphite

$$: 6.58 \times 10^{12} \text{ n/cm}^2\text{sec.}$$

RESULTS NORMALIZED TO DESIGN POWER

Average wall loading

$$: 1 \text{ MW/m}^2 \frac{1000 \text{ MW(t)}}{2306 \text{ MW(t)}} = 0.43 \text{ MW/m}^2.$$

Peak Wall Loading

$$: (0.43 \text{ MW/m}^2) / (86\% \text{ on-time}) = 0.50 \text{ MW/m}^2$$

Group #1 flux at first
wall, average
peak

$$: (7.92 \times 10^{13} \text{ n/cm}^2\text{sec}) (0.43) = 3.43 \times 10^{13} \text{ n/cm}^2\text{sec}$$

$$: (7.92 \times 10^{13} \text{ n/cm}^2\text{sec}) (0.50) = 3.99 \times 10^{13} \text{ n/cm}^2\text{sec}$$

Group #1 flux at inner surface of
graphite, average
peak

$$: (6.58 \times 10^{12} \text{ n/cm}^2\text{sec}) (0.43) = 2.85 \times 10^{12} \text{ n/cm}^2\text{sec}$$

$$: (6.58 \times 10^{12} \text{ n/cm}^2\text{sec}) (0.50) = 3.32 \times 10^{12} \text{ n/cm}^2\text{sec}$$

* At one neutron per fusion event

** The value of 22.4 MeV/fusion is an estimate, but is in reasonable agreement with the calculations.⁽⁴⁾

$$\Phi_G = \frac{\int_0^{t_2} \int_0^{\infty} \sigma_s(E) p(E) \phi(E,t) dE dt}{\int_0^{\infty} \sigma_s(E) p(E) \chi(E) dE / \int_0^{\infty} \chi(E) dE} \quad (1)$$

Where $\phi(E,t)$ is the absolute flux density spectrum at time t
 $\sigma_s(E)$ is the scattering cross section of carbon
 $p(E)$ is the Thompson and Wright form of the atomic displacement weighting function, and
 $\chi(E)$ is the fission spectrum

The upper integral of equation (1) is the atomic displacement rate integrated over the time of the irradiation and the lower ratio is merely a normalization factor, $\bar{D}_f$, which converts the units of Φ_G into n/cm^2 and, at the same time, tends to reduce the dependence of Φ_G on the particular form used for the weighting function, $p(E)$. Jenkins⁽⁹⁾ has used the Thompson and Wright form of the weighting function, $p(E)$, and neutron scattering cross section, $\sigma_s(E)$, processed from ENDF/B library files to calculate group averaged displacement cross section, $\sigma_d(\Delta U)_i$, for carbon.

$$\sigma_d(\Delta U)_i = \frac{\int_{E_i}^{E_i+\Delta E_i} \sigma_s(E) p(E) \phi_n(E) dE}{\int_{E_i}^{E_i+\Delta E_i} \phi_n(E) dE} \quad (2)$$

Where $\sigma_d(\Delta U)_i$ is the group averaged atomic displacement cross section for the i^{th} lethargy group of width ΔU
 ΔU is $\ln(E_i+\Delta E_i/E_i)$
 E_i and $(E_i+\Delta E_i)$ are the lower and upper energy limits for the lethargy group; and
 $\phi_n(E)$ is merely the particular spectrum used for normalization. [The values derived from $\sigma_d(\Delta U)_i$ are practically insensitive to the form used for $\phi_n(E)$ if $(\Delta E_i < E_i)$.]

These group average displacement cross sections have been used previously for calculations of dpa and for comparisons of graphite irradiation data; we have, therefore, used the values derived by Jenkins⁽⁹⁾ for our first series of comparisons.

The calculation of the normalization factor,

$$\bar{D}_f = \frac{\sum_{i=1}^{\infty} \sigma_d(\Delta U)_i \chi(\Delta U)_i \Delta U_i}{\sum_{i=1}^{\infty} \chi(\Delta U)_i \Delta U_i}, \quad (3)$$

is illustrated in Table V, and Tables VI to X illustrate the calculations of atomic displacement rates (normalized for a wall loading of 1 MW/m²) at various positions in the graphite region of the CTR blanket.

The integral displacement rates from Tables VI to X have been divided by the normalization factor from Table V to yield the equivalent fission flux, ϕ_G^* , as a function of position in the graphite which is plotted in Figure 3. The integral flux above 10 MeV, $\phi(E > 10 \text{ MeV})$, has also been plotted so that one can see how the neutron energy is degraded as one progresses radially outward through the blanket. These same two fluxes are also plotted for both the inner and outer lithium regions. It can be seen that the flux attenuation is well represented by an equation of the form

$$\phi_x = \phi_0 e^{-ax}$$

The values derived for the flux attenuation factors, a , in the different regions are listed in Table XI.

* It should be noted that the fluence scale for all previous graphite radiation data from this laboratory has been in units of $\phi(E > 0.18 \text{ MeV})$. This is related to ϕ_G by: (11)

$$\phi_G = 1.18 \phi(E > 0.18 \text{ MeV})$$

TABLE IV. NORMALIZATION FACTOR FOR EQUIVALENT FISSION FLUENCE
U-235 FISSION NEUTRON FLUX NORMALIZED TO 1.0 NEUTRON
PER SQ. CM. OVER THE INTEGRAL

LETHARGY GROUP NUMBER	DISPL. (1) CROSS SECTION	NEUTRON FLUX PER GROUP (2)	INTEGRAL NEUTRON FLUX (3)	DISPL. RATE PER GROUP (4)	INTEGRAL DISPL. RATE (5)
1	270.360	4.1217-05 *	4.1217-05	1.1143-26	1.1143-26
2	297.460	1.1332-04	1.5454-04	3.3708-26	4.4852-26
3	213.600	2.7702-04	4.3157-04	2.6377-26	1.8173-26
4	266.210	6.0915-04	1.0407-03	1.6216-25	2.9358-26
5	247.490	1.2175-03	2.2582-03	3.0132-25	5.9521-26
6	254.730	2.2331-03	4.4913-03	5.6295-25	1.1642-24
7	436.360	3.7910-03	8.2323-03	1.8438-24	3.0079-24
8	295.630	6.0034-03	1.4286-02	1.7148-24	4.7227-24
9	348.690	8.9304-03	2.3216-02	3.1139-24	7.3366-24
10	345.670	1.2558-02	3.5774-02	4.3409-24	1.2172-23
11	418.070	1.6791-02	5.2565-02	7.0198-24	1.9197-23
12	603.610	2.1454-02	7.4019-02	1.0804-23	2.0002-23
13	643.560	2.6320-02	1.0034-01	1.6938-23	4.6940-23
14	746.060	3.1131-02	1.3147-01	2.3226-23	7.0166-23
15	844.780	3.5636-02	1.6711-01	3.0105-23	1.0027-22
16	566.740	3.9513-02	2.0672-01	2.2450-23	1.2272-22
17	708.670	4.2891-02	2.4961-01	3.0396-23	1.5312-22
18	525.210	4.5257-02	2.9497-01	2.3822-23	1.7694-22
19	601.860	4.6962-02	3.4193-01	2.3568-23	2.3051-22
20	575.110	4.7712-02	3.8964-01	2.7440-23	2.2795-22
21	522.260	4.7560-02	4.3730-01	2.4891-23	2.5284-22
22	538.270	4.6890-02	4.8419-01	2.5244-23	2.7808-22
23	556.400	4.5509-02	5.2970-01	2.5321-23	3.0340-22
24	574.480	4.3634-02	5.7333-01	2.5067-23	3.2847-22
25	591.510	4.1381-02	6.1472-01	2.4477-23	3.5295-22
26	607.090	3.8863-02	6.5352-01	2.3891-23	3.7654-22
27	620.040	3.6179-02	6.8976-01	2.2466-23	3.9901-22
28	632.700	3.3416-02	7.2317-01	2.1142-23	4.2015-22
29	642.250	3.0647-02	7.5382-01	1.9683-23	4.3983-22
30	649.470	2.7930-02	7.8175-01	1.8140-23	4.5797-22
31	654.210	2.5310-02	8.0706-01	1.6558-23	4.7453-22
32	656.520	2.2818-02	8.2988-01	1.4980-23	4.8951-22
33	656.350	2.0478-02	8.5036-01	1.3441-23	5.0295-22
34	653.030	1.8302-02	8.6866-01	1.1968-23	5.1492-22
35	649.100	1.6297-02	8.8496-01	1.0578-23	5.2550-22
36	641.830	1.4463-02	8.9942-01	9.2828-24	5.3476-22
37	632.210	1.2797-02	9.1222-01	8.0954-24	5.4287-22
38	620.100	1.1293-02	9.2351-01	7.0028-24	5.4987-22
39	603.770	9.9408-03	9.3345-01	6.0020-24	5.5587-22
40	585.210	8.7316-03	9.4218-01	5.1098-24	5.6098-22
41	565.290	7.6544-03	9.4983-01	4.3263-24	5.6551-22
42	544.020	6.6982-03	9.5653-01	3.6440-24	5.6895-22
43	521.810	5.8521-03	9.6239-01	3.0537-24	5.7201-22
44	498.830	5.1054-03	9.6749-01	2.5467-24	5.7455-22
45	475.410	4.4482-03	9.7194-01	2.1147-24	5.7667-22
46	451.620	3.8711-03	9.7581-01	1.7483-24	5.7842-22
47	427.820	3.3652-03	9.7918-01	1.4397-24	5.7986-22
48	404.210	2.9227-03	9.8210-01	1.1814-24	5.8104-22
49	380.810	2.5361-03	9.8463-01	9.6577-25	5.8200-22
50	341.050	4.9589-03	9.8959-01	1.6912-24	5.8370-22

* ±N REPRESENTS 10^{±N}

TABLE V. (CONT.)

LETHARGY GROUP NUMBER	DISPL. (1) CROSS SECTION	NEUTRON FLUX PER GROUP (2)	INTEGRAL NEUTRON FLUX (3)	DISPL. RATE PER GROUP (4)	INTEGRAL DISPL. RATE (5)
51	287.510	4.4578-03	9.9305-01	9.9415-25	5.8469-22
52	238.810	2.4034-03	9.9345-01	5.7396-25	5.8526-22
53	195.480	1.6653-03	9.9312-01	3.2573-25	5.8559-22
54	157.580	1.1531-03	9.9327-01	1.8171-25	5.8577-22
55	125.510	7.8669-04	9.9307-01	1.0007-25	5.8587-22
56	99.899	5.4981-04	9.9362-01	5.4425-26	5.8593-22
57	79.263	3.7909-04	1.0000+00	3.0025-26	5.8596-22
58	63.004	0.0000	1.0000+00	0.0000	5.8596-22
59	49.956	0.0000	1.0000+00	0.0000	5.8596-22
60	39.642	0.0000	1.0000+00	0.0000	5.8596-22
61	31.423	0.0000	1.0000+00	0.0000	5.8596-22
62	24.936	0.0000	1.0000+00	0.0000	5.8596-22
63	19.729	0.0000	1.0000+00	0.0000	5.8596-22
64	15.520	0.0000	1.0000+00	0.0000	5.8596-22
65	12.372	0.0000	1.0000+00	0.0000	5.8596-22
66	9.778	0.0000	1.0000+00	0.0000	5.8596-22
67	7.733	0.0000	1.0000+00	0.0000	5.8596-22
68	6.166	0.0000	1.0000+00	0.0000	5.8596-22
69	4.887	0.0000	1.0000+00	0.0000	5.8596-22
70	3.871	0.0000	1.0000+00	0.0000	5.8596-22
71	2.962	0.0000	1.0000+00	0.0000	5.8596-22
72	2.311	0.0000	1.0000+00	0.0000	5.8596-22
73	1.587	0.0000	1.0000+00	0.0000	5.8596-22
74	.651	0.0000	1.0000+00	0.0000	5.8596-22
75	.006	0.0000	1.0000+00	0.0000	5.8596-22
76	.000	0.0000	1.0000+00	0.0000	5.8596-22
77	.000	0.0000	1.0000+00	0.0000	5.8596-22
78	.000	0.0000	1.0000+00	0.0000	5.8596-22
79	.000	0.0000	1.0000+00	0.0000	5.8596-22
80	.000	0.0000	1.0000+00	0.0000	5.8596-22
81	.000	0.0000	1.0000+00	0.0000	5.8596-22
82	.000	0.0000	1.0000+00	0.0000	5.8596-22
83	.000	0.0000	1.0000+00	0.0000	5.8596-22
84	.000	0.0000	1.0000+00	0.0000	5.8596-22
85	.000	0.0000	1.0000+00	0.0000	5.8596-22
86	.000	0.0000	1.0000+00	0.0000	5.8596-22
87	.000	0.0000	1.0000+00	0.0000	5.8596-22
88	.000	0.0000	1.0000+00	0.0000	5.8596-22
89	.000	0.0000	1.0000+00	0.0000	5.8596-22
90	.000	0.0000	1.0000+00	0.0000	5.8596-22
91	.000	0.0000	1.0000+00	0.0000	5.8596-22
92	.000	0.0000	1.0000+00	0.0000	5.8596-22
93	.000	0.0000	1.0000+00	0.0000	5.8596-22
94	.000	0.0000	1.0000+00	0.0000	5.8596-22
95	.000	0.0000	1.0000+00	0.0000	5.8596-22
96	.000	0.0000	1.0000+00	0.0000	5.8596-22
97	.000	0.0000	1.0000+00	0.0000	5.8596-22
98	.000	0.0000	1.0000+00	0.0000	5.8596-22
99	.000	0.0000	1.0000+00	0.0000	5.8596-22
100	.000	0.0000	1.0000+00	0.0000	5.8596-22

- 1) CARBON ATOM DISPLACEMENT CROSS SECTION (BARNS)
- 2) NEUTRONS PER SQUARE CENTIMETER PER SECOND IN THIS LETHARGY GROUP
- 3) NEUTRONS PER SQUARE CENTIMETER PER SECOND WITH ENERGIES GREATER THAN THE LOWER ENERGY FOR THIS GROUP
- 4) ATOMS DISPLACED PER ATOM PER SECOND BY NEUTRONS WITHIN THIS LETHARGY GROUP
- 5) TOTAL ATOMS DISPLACED PER ATOM PER SECOND BY NEUTRONS WITH ENERGIES GREATER THAN THE LOWER ENERGY LIMIT FOR THIS GROUP

TABLE VI. NEUTRON FLUX AND ATOMIC DISPLACEMENT RATES FOR GRAPHITE
IN ZONE 39 OF THE O.R.N.L. CONCEPTUAL C.T.R., VERSION-3
DATA NORMALIZED TO 1.0 MEGAWATT PER SQ. METER WALL LOADING

LETHARGY GROUP NUMBER	DISPL.(1) CROSS SECTION	NEUTRON FLUX PER GROUP (2)	INTEGRAL NEUTRON FLUX (3)	DISPL. RATE PER GROUP (4)	INTEGRAL DISPL. RATE (5)
1	270.360	6.5774+12*	6.5774+12	1.7783-09	1.7783-09
2	297.460	3.7154+12	1.0293+13	1.1052-09	2.8835-09
3	313.600	2.0838+12	1.2377+13	6.5348-10	3.5269-09
4	266.210	1.8173+12	1.4194+13	4.8391-10	4.0208-09
5	247.490	1.3622+12	1.5557+13	3.3714-10	4.3580-09
6	254.780	1.3721+12	1.6929+13	3.4957-10	4.7076-09
7	486.360	1.1743+12	1.8103+13	5.7113-10	5.2787-09
8	285.630	1.2510+12	1.9354+13	3.5731-10	5.6360-09
9	348.690	9.9251+11	2.0346+13	3.4608-10	5.9821-09
10	345.670	1.0924+12	2.1439+13	3.7760-10	6.3597-09
11	418.070	9.5787+11	2.2397+13	4.0046-10	6.7601-09
12	503.610	1.0180+12	2.3415+13	5.1269-10	7.2728-09
13	643.560	1.0021+12	2.4417+13	6.4489-10	7.9177-09
14	746.060	9.7956+11	2.5396+13	7.3081-10	8.6485-09
15	844.780	9.1293+11	2.6309+13	7.7123-10	9.4198-09
16	566.740	1.1408+12	2.7450+13	6.4653-10	1.0065-08
17	708.670	1.2681+12	2.8718+13	8.9868-10	1.0965-08
18	525.210	1.6231+12	3.0341+13	8.5248-10	1.1817-08
19	501.860	1.6792+12	3.2021+13	8.4274-10	1.2660-08
20	575.110	1.7412+12	3.3762+13	1.0014-09	1.3662-08
21	522.260	1.8314+12	3.5593+13	9.5647-10	1.4618-08
22	538.370	1.8501+12	3.7443+13	9.9604-10	1.5614-08
23	556.400	1.9498+12	3.9393+13	1.0848-09	1.6699-08
24	574.480	1.9825+12	4.1375+13	1.1389-09	1.7838-08
25	591.510	1.9960+12	4.3372+13	1.1807-09	1.9018-08
26	607.090	2.0293+12	4.5401+13	1.2319-09	2.0250-08
27	620.940	2.0500+12	4.7451+13	1.2729-09	2.1523-08
28	632.700	2.0966+12	4.9547+13	1.3265-09	2.2850-08
29	642.250	2.1519+12	5.1699+13	1.3821-09	2.4232-08
30	649.470	2.2233+12	5.3923+13	1.4440-09	2.5676-08
31	654.210	2.3245+12	5.6247+13	1.5207-09	2.7197-08
32	656.520	2.4278+12	5.8675+13	1.5939-09	2.8791-08
33	656.350	2.5305+12	6.1205+13	1.6609-09	3.0451-08
34	653.930	2.6171+12	6.3823+13	1.7114-09	3.2163-08
35	649.100	2.6661+12	6.6489+13	1.7306-09	3.3893-08
36	641.830	2.6273+12	6.9116+13	1.6863-09	3.5580-08
37	632.210	2.5566+12	7.1673+13	1.6163-09	3.7196-08
38	620.100	2.3889+12	7.4061+13	1.4814-09	3.8677-08
39	603.770	2.1007+12	7.6162+13	1.2683-09	3.9946-08
40	585.210	1.7606+12	7.7923+13	1.0303-09	4.0976-08
41	565.200	1.4839+12	7.9407+13	8.3870-10	4.1815-08
42	544.020	1.4549+12	8.0861+13	7.9148-10	4.2606-08
43	521.810	1.7583+12	8.2620+13	9.1752-10	4.3524-08
44	498.830	2.1946+12	8.4814+13	1.0947-09	4.4618-08
45	475.410	2.6767+12	8.7491+13	1.2725-09	4.5891-08
46	451.620	3.0369+12	9.0528+13	1.3715-09	4.7262-08
47	427.820	3.1640+12	9.3692+13	1.3536-09	4.8616-08
48	404.210	3.2216+12	9.6914+13	1.3022-09	4.9918-08
49	380.810	3.2634+12	1.0018+14	1.2427-09	5.1161-08
50	341.050	8.0085+12	1.0819+14	2.7313-09	5.3892-08

* $\pm n$ represents $10^{\pm n}$

TABLE VI. (CONT.)

LETHARGY GROUP NUMBER	DISPL. (1) CROSS SECTION	NEUTRON FLUX PER GROUP (2)	INTEGRAL NEUTRON FLUX (3)	DISPL. RATE PER GROUP (4)	INTEGRAL DISPL. RATE (5)
51	287.510	7.6096+12	1.1580+14	2.1878-09	5.6080-08
52	238.810	7.1732+12	1.2297+14	1.7130-09	5.7793-08
53	195.480	6.7364+12	1.2970+14	1.3168-09	5.9110-08
54	157.580	6.3197+12	1.3602+14	9.9586-10	6.0106-08
55	125.610	5.9262+12	1.4195+14	7.4438-10	6.0850-08
56	99.899	5.5556+12	1.4751+14	5.5500-10	6.1405-08
57	79.363	5.2041+12	1.5271+14	4.1301-10	6.1818-08
58	63.004	4.8698+12	1.5758+14	3.0682-10	6.2125-08
59	49.956	4.5509+12	1.6213+14	2.2734-10	6.2352-08
60	39.642	4.2478+12	1.6638+14	1.6839-10	6.2521-08
61	31.423	3.9559+12	1.7033+14	1.2431-10	6.2645-08
62	24.906	3.7007+12	1.7404+14	9.2169-11	6.2737-08
63	19.729	3.4004+12	1.7744+14	6.7087-11	6.2804-08
64	15.620	3.1189+12	1.8055+14	4.8717-11	6.2853-08
65	12.373	2.9160+12	1.8347+14	3.6079-11	6.2889-08
66	9.778	2.6658+12	1.8614+14	2.6067-11	6.2915-08
67	7.733	2.4305+12	1.8857+14	1.8795-11	6.2934-08
68	6.106	2.2068+12	1.9077+14	1.3474-11	6.2947-08
69	4.807	2.0097+12	1.9278+14	9.6597-12	6.2957-08
70	3.801	1.8233+12	1.9461+14	6.9306-12	6.2964-08
71	2.963	1.6708+12	1.9628+14	4.9496-12	6.2969-08
72	2.301	1.5085+12	1.9779+14	3.4706-12	6.2972-08
73	1.587	1.3825+12	1.9917+14	2.1936-12	6.2975-08
74	.651	1.2731+12	2.0044+14	8.2827-13	6.2976-08
75	.006	1.1512+12	2.0159+14	6.7922-15	6.2976-08
76	.000	1.0728+12	2.0267+14	0.0000	6.2976-08
77	.000	9.8618+11	2.0365+14	0.0000	6.2976-08
78	.000	9.2199+11	2.0457+14	0.0000	6.2976-08
79	.000	8.6243+11	2.0544+14	0.0000	6.2976-08
80	.000	8.0891+11	2.0625+14	0.0000	6.2976-08
81	.000	7.5934+11	2.0700+14	0.0000	6.2976-08
82	.000	7.1426+11	2.0772+14	0.0000	6.2976-08
83	.000	6.7495+11	2.0839+14	0.0000	6.2976-08
84	.000	6.3896+11	2.0903+14	0.0000	6.2976-08
85	.000	6.0625+11	2.0964+14	0.0000	6.2976-08
86	.000	5.7638+11	2.1022+14	0.0000	6.2976-08
87	.000	5.4904+11	2.1076+14	0.0000	6.2976-08
88	.000	5.2398+11	2.1129+14	0.0000	6.2976-08
89	.000	5.0095+11	2.1179+14	0.0000	6.2976-08
90	.000	4.7972+11	2.1227+14	0.0000	6.2976-08
91	.000	4.6009+11	2.1273+14	0.0000	6.2976-08
92	.000	4.4197+11	2.1317+14	0.0000	6.2976-08
93	.000	4.2509+11	2.1360+14	0.0000	6.2976-08
94	.000	4.0932+11	2.1401+14	0.0000	6.2976-08
95	.000	3.9457+11	2.1440+14	0.0000	6.2976-08
96	.000	3.8079+11	2.1478+14	0.0000	6.2976-08
97	.000	3.6787+11	2.1515+14	0.0000	6.2976-08
98	.000	3.5573+11	2.1550+14	0.0000	6.2976-08
99	.000	3.4428+11	2.1585+14	0.0000	6.2976-08
100	.000	1.4778+13	2.3063+14	0.0000	6.2976-08

1) CARBON ATOM DISPLACEMENT CROSS SECTION (BARNs)

2) NEUTRONS PER SQUARE CENTIMETER PER SECOND IN THIS LETHARGY GROUP

3) NEUTRONS PER SQUARE CENTIMETER PER SECOND WITH ENERGIES GREATER THAN THE LOWER ENERGY FOR THIS GROUP

4) ATOMS DISPLACED PER ATOM PER SECOND BY NEUTRONS WITHIN THIS LETHARGY GROUP

5) TOTAL ATOMS DISPLACED PER ATOM PER SECOND BY NEUTRONS WITH ENERGIES GREATER THAN THE LOWER ENERGY LIMIT FOR THIS GROUP

TABLE VII. NEUTRON FLUX AND ATOMIC DISPLACEMENT RATES FOR GRAPHITE
IN ZONE 43 OF THE O.R.N.L. CONCEPTUAL C.T.R., VERSION-3
DATA NORMALIZED TO 1.0 MEGAWATT PER SQ. METER WALL LOADING

LETHARGY GROUP NUMBER	DISPL. (1) CROSS SECTION	NEUTRON FLUX PER GROUP (2)	INTEGRAL NEUTRON FLUX (3)	DISPL. RATE PER GROUP (4)	INTEGRAL DISPL. RATE (5)
1	270.360	3.0292+12*	3.0292+12	8.1897-10	8.1897-10
2	297.460	1.8687+12	4.8979+12	5.5587-10	1.3748-09
3	313.600	1.1067+12	6.0046+12	3.4706-10	1.7219-09
4	266.210	9.5426+11	6.9588+12	2.5403-10	1.9759-09
5	247.490	6.8043+11	7.6393+12	1.6840-10	2.1443-09
6	254.780	8.7070+11	8.5100+12	2.2184-10	2.3662-09
7	486.360	6.4219+11	9.1522+12	3.1234-10	2.6785-09
8	285.630	9.1766+11	1.0070+13	2.6211-10	2.9406-09
9	348.690	6.3704+11	1.0707+13	2.2213-10	3.1627-09
10	345.670	8.0003+11	1.1507+13	2.7655-10	3.4393-09
11	418.070	6.8390+11	1.2191+13	2.8592-10	3.7252-09
12	503.610	7.4826+11	1.2939+13	3.7683-10	4.1020-09
13	643.560	7.1685+11	1.3656+13	4.6133-10	4.5634-09
14	746.060	6.4895+11	1.4305+13	4.8416-10	5.0475-09
15	844.780	5.2274+11	1.4828+13	4.4160-10	5.4891-09
16	566.740	7.3402+11	1.5562+13	4.1600-10	5.9051-09
17	708.670	7.5842+11	1.6320+13	5.3747-10	6.4426-09
18	525.210	1.0703+12	1.7390+13	5.6211-10	7.0047-09
19	501.860	1.1147+12	1.8505+13	5.5941-10	7.5641-09
20	575.110	1.1436+12	1.9649+13	6.5770-10	8.2218-09
21	522.260	1.2401+12	2.0889+13	6.4763-10	8.8695-09
22	538.370	1.2232+12	2.2112+13	6.5854-10	9.5280-09
23	556.400	1.2826+12	2.3394+13	7.1366-10	1.0242-08
24	574.480	1.2842+12	2.4679+13	7.3774-10	1.0979-08
25	591.510	1.2683+12	2.5947+13	7.5022-10	1.1730-08
26	607.090	1.2654+12	2.7212+13	7.6820-10	1.2498-08
27	620.940	1.2454+12	2.8458+13	7.7333-10	1.3271-08
28	632.700	1.2359+12	2.9694+13	7.8193-10	1.4053-08
29	642.250	1.2251+12	3.0919+13	7.8679-10	1.4840-08
30	649.470	1.2192+12	3.2138+13	7.9184-10	1.5632-08
31	654.210	1.2153+12	3.3353+13	7.9507-10	1.6427-08
32	656.520	1.2132+12	3.4566+13	7.9647-10	1.7223-08
33	656.350	1.2164+12	3.5783+13	7.9837-10	1.8022-08
34	653.930	1.2199+12	3.7003+13	7.9772-10	1.8819-08
35	649.100	1.2221+12	3.8225+13	7.9329-10	1.9613-08
36	641.830	1.2211+12	3.9446+13	7.8371-10	2.0396-08
37	632.210	1.2178+12	4.0664+13	7.6994-10	2.1166-08
38	620.100	1.2094+12	4.1873+13	7.4993-10	2.1916-08
39	603.770	1.1924+12	4.3066+13	7.1995-10	2.2636-08
40	585.210	1.1687+12	4.4234+13	6.8396-10	2.3320-08
41	565.200	1.1427+12	4.5377+13	6.4587-10	2.3966-08
42	544.020	1.1223+12	4.6499+13	6.1054-10	2.4577-08
43	521.810	1.1152+12	4.7614+13	5.8191-10	2.5158-08
44	498.830	1.1203+12	4.8735+13	5.5885-10	2.5717-08
45	475.410	1.1348+12	4.9870+13	5.3952-10	2.6257-08
46	451.620	1.1538+12	5.1023+13	5.2110-10	2.6778-08
47	427.820	1.1721+12	5.2196+13	5.0143-10	2.7279-08
48	404.210	1.1904+12	5.3386+13	4.8116-10	2.7761-08
49	380.810	1.2105+12	5.4596+13	4.6099-10	2.8221-08
50	341.050	3.1288+12	5.7725+13	1.0671-09	2.9289-08

* 10^{+n} represents 10^{-n}

TABLE VII. (CONT.)

LETHARGY GROUP NUMBER	DISPL. (1) CROSS SECTION	NEUTRON FLUX PER GROUP (2)	INTEGRAL NEUTRON FLUX (3)	DISPL. RATE PER GROUP (4)	INTEGRAL DISPL. RATE (5)
51	287.510	3.2515+12	6.0977+13	9.3484-10	3.0223-08
52	238.810	3.3549+12	6.4332+13	8.0120-10	3.1025-08
53	195.480	3.4377+12	6.7769+13	6.7199-10	3.1697-08
54	157.580	3.4980+12	7.1267+13	5.5121-10	3.2248-08
55	125.610	3.5374+12	7.4805+13	4.4434-10	3.2692-08
56	99.800	3.5583+12	7.8363+13	3.5547-10	3.3048-08
57	79.363	3.5619+12	8.1925+13	2.8268-10	3.3330-08
58	63.004	3.5502+12	8.5475+13	2.2368-10	3.3554-08
59	49.956	3.5249+12	8.9000+13	1.7609-10	3.3730-08
60	39.642	3.4885+12	9.2489+13	1.3829-10	3.3868-08
61	31.423	3.4423+12	9.5931+13	1.0817-10	3.3977-08
62	24.906	3.3878+12	9.9319+13	8.4376-11	3.4061-08
63	19.729	3.3247+12	1.0264+14	6.5592-11	3.4127-08
64	15.620	3.2549+12	1.0590+14	5.0842-11	3.4177-08
65	12.373	3.1807+12	1.0908+14	3.9354-11	3.4217-08
66	9.778	3.1012+12	1.1218+14	3.0324-11	3.4247-08
67	7.733	3.0181+12	1.1520+14	2.3338-11	3.4270-08
68	6.106	2.9317+12	1.1813+14	1.7900-11	3.4288-08
69	4.807	2.8428+12	1.2097+14	1.3664-11	3.4302-08
70	3.801	2.7525+12	1.2373+14	1.0463-11	3.4312-08
71	2.963	2.6620+12	1.2639+14	7.8862-12	3.4320-08
72	2.301	2.5714+12	1.2896+14	5.9158-12	3.4326-08
73	1.587	2.4822+12	1.3144+14	3.9384-12	3.4330-08
74	.651	2.3947+12	1.3384+14	1.5580-12	3.4332-08
75	.006	2.3089+12	1.3614+14	1.3622-14	3.4332-08
76	.000	2.2259+12	1.3837+14	0.0000	3.4332-08
77	.000	2.1454+12	1.4052+14	0.0000	3.4332-08
78	.000	2.0682+12	1.4258+14	0.0000	3.4332-08
79	.000	1.9942+12	1.4458+14	0.0000	3.4332-08
80	.000	1.9235+12	1.4650+14	0.0000	3.4332-08
81	.000	1.8559+12	1.4836+14	0.0000	3.4332-08
82	.000	1.7917+12	1.5015+14	0.0000	3.4332-08
83	.000	1.7305+12	1.5188+14	0.0000	3.4332-08
84	.000	1.6722+12	1.5355+14	0.0000	3.4332-08
85	.000	1.6168+12	1.5517+14	0.0000	3.4332-08
86	.000	1.5640+12	1.5673+14	0.0000	3.4332-08
87	.000	1.5137+12	1.5825+14	0.0000	3.4332-08
88	.000	1.4659+12	1.5971+14	0.0000	3.4332-08
89	.000	1.4204+12	1.6113+14	0.0000	3.4332-08
90	.000	1.3769+12	1.6251+14	0.0000	3.4332-08
91	.000	1.3354+12	1.6384+14	0.0000	3.4332-08
92	.000	1.2959+12	1.6514+14	0.0000	3.4332-08
93	.000	1.2581+12	1.6640+14	0.0000	3.4332-08
94	.000	1.2220+12	1.6762+14	0.0000	3.4332-08
95	.000	1.1875+12	1.6881+14	0.0000	3.4332-08
96	.000	1.1544+12	1.6996+14	0.0000	3.4332-08
97	.000	1.1227+12	1.7109+14	0.0000	3.4332-08
98	.000	1.0923+12	1.7218+14	0.0000	3.4332-08
99	.000	1.0631+12	1.7324+14	0.0000	3.4332-08
100	.000	4.9031+13	2.2227+14	0.0000	3.4332-08

- 1) CARBON ATOM DISPLACEMENT CROSS SECTION (BARNS)
- 2) NEUTRONS PER SQUARE CENTIMETER PER SECOND IN THIS LETHARGY GROUP
- 3) NEUTRONS PER SQUARE CENTIMETER PER SECOND WITH ENERGIES GREATER THAN THE LOWER ENERGY FOR THIS GROUP
- 4) ATOMS DISPLACED PER ATOM PER SECOND BY NEUTRONS WITHIN THIS LETHARGY GROUP
- 5) TOTAL ATOMS DISPLACED PER ATOM PER SECOND BY NEUTRONS WITH ENERGIES GREATER THAN THE LOWER ENERGY LIMIT FOR THIS GROUP

TABLE VIII. NEUTRON FLUX AND ATOMIC DISPLACEMENT RATES FOR GRAPHITE
IN ZONE 48 OF THE O.R.N.L. CONCEPTUAL C.T.R., VERSION-3
DATA NORMALIZED TO 1.0 MEGAWATT PER SQ. METER WALL LOADING

LETHARGY GROUP NUMBER	DISPL. (1) CROSS SECTION	NEUTRON FLUX PER GROUP (2)	INTEGRAL NEUTRON FLUX (3)	DISPL. RATE PER GROUP (4)	INTEGRAL DISPL. RATE (5)
1	270.360	1.1782+12*	1.1782+12	3.1854-10	3.1854-10
2	297.460	8.1238+11	1.9906+12	2.4165-10	5.6019-10
3	313.600	5.0427+11	2.4948+12	1.5814-10	7.1832-10
4	266.210	4.4000+11	2.9357+12	1.1737-10	8.3570-10
5	247.490	3.0989+11	3.2456+12	7.6695-11	9.1239-10
6	254.780	4.4603+11	3.6917+12	1.1364-10	1.0260-09
7	486.360	2.9129+11	3.9829+12	1.4167-10	1.1677-09
8	285.650	5.2257+11	4.5055+12	1.4926-10	1.3170-09
9	348.590	5.3704+11	4.8426+12	1.1752-10	1.4345-09
10	345.670	4.5935+11	5.3019+12	1.5878-10	1.5933-09
11	418.070	3.9454+11	5.6964+12	1.6495-10	1.7582-09
12	503.610	4.3443+11	6.1309+12	2.1874-10	1.9770-09
13	641.550	4.1551+11	6.5465+12	2.6747-10	2.2445-09
14	746.060	3.6714+11	6.9136+12	2.7391-10	2.5164-09
15	644.750	2.5339+11	7.1970+12	2.5940-10	2.7778-09
16	566.740	4.0474+11	7.5013+12	2.2938-10	2.9812-09
17	708.670	4.0570+11	8.0085+12	2.8622-10	3.2754-09
18	525.210	5.9139+11	8.2995+12	3.1060-10	3.5860-09
19	501.860	6.2075+11	8.2206+12	3.1155-10	3.8975-09
20	575.110	6.5787+11	7.8503+12	3.6864-10	4.2644-09
21	522.260	7.0001+11	1.0555+13	3.6559-10	4.6299-09
22	555.570	6.5961+11	1.1246+13	3.7137-10	5.0013-09
23	556.400	7.2254+11	1.1971+13	4.0202-10	5.4035-09
24	574.480	7.2200+11	1.2695+13	4.1477-10	5.8161-09
25	591.510	7.1015+11	1.3403+13	4.2006-10	6.2502-09
26	607.090	7.0471+11	1.4108+13	4.2782-10	6.6860-09
27	620.940	6.8906+11	1.4797+13	4.2707-10	7.0955-09
28	632.700	6.7750+11	1.5474+13	4.2685-10	7.5225-09
29	642.250	6.6416+11	1.6136+13	4.2636-10	7.9491-09
30	645.470	6.5256+11	1.6791+13	4.2382-10	8.3729-09
31	654.210	6.4050+11	1.7451+13	4.1689-10	8.7918-09
32	656.520	6.2829+11	1.8080+13	4.1249-10	9.2045-09
33	656.550	6.1855+11	1.8678+13	4.0598-10	9.6102-09
34	653.930	6.0945+11	1.9288+13	3.9852-10	1.0009-08
35	649.100	6.0054+11	1.9888+13	3.9000-10	1.0399-08
36	641.850	5.9297+11	2.0461+13	3.8055-10	1.0779-08
37	632.210	5.8615+11	2.1057+13	3.7056-10	1.1150-08
38	620.100	5.8000+11	2.1647+13	3.5966-10	1.1510-08
39	603.770	5.7411+11	2.2222+13	3.4805-10	1.1856-08
40	585.270	5.6870+11	2.2790+13	3.3251-10	1.2189-08
41	565.200	5.6381+11	2.3354+13	3.1866-10	1.2508-08
42	544.020	5.5955+11	2.3914+13	3.0441-10	1.2812-08
43	521.810	5.5622+11	2.4470+13	2.9024-10	1.3102-08
44	498.850	5.5361+11	2.5024+13	2.7626-10	1.3379-08
45	475.410	5.5212+11	2.5576+13	2.6246-10	1.3641-08
46	451.620	5.5116+11	2.6127+13	2.4892-10	1.3890-08
47	427.820	5.5095+11	2.6678+13	2.3571-10	1.4126-08
48	404.210	5.5113+11	2.7229+13	2.2277-10	1.4346-08
49	380.810	5.5194+11	2.7781+13	2.1019-10	1.4559-08
50	341.050	1.5667+12	2.9170+13	4.1562-10	1.5652-08

* $\pm n$, represents $10^{\pm n}$

TABLE VIII. (CONT.)

LEAD GROUP NUMBER	DISPL. (1) CROSS SECTION	NEUTRON FLUX PER GROUP (2)	INTEGRAL NEUTRON FLUX (3)	DISPL. RATE PER GROUP (4)	INTEGRAL DISPL. RATE (5)
51	287.510	1.4064+12	3.0076+13	4.0434-10	1.3431-03
52	238.810	1.4305+12	3.2007+13	3.4102-10	1.3778-03
53	195.480	1.4599+12	3.3467+13	2.8539-10	1.0064-06
54	157.580	1.4928+12	3.4999+13	2.3523-10	1.6299-06
55	125.610	1.5280+12	3.6487+13	1.9194-10	1.6491-08
56	99.899	1.5649+12	3.8052+13	1.5033-10	1.6647-08
57	79.353	1.6019+12	3.9654+13	1.2713-10	1.6774-08
58	63.004	1.6382+12	4.1292+13	1.0321-10	1.6877-08
59	49.936	1.6731+12	4.2965+13	8.3561-11	1.6961-08
60	37.642	1.7062+12	4.4672+13	6.7638-11	1.7029-08
61	31.423	1.7371+12	4.6409+13	5.4305-11	1.7083-08
62	24.905	1.7651+12	4.8174+13	4.3980-11	1.7127-08
63	17.729	1.7900+12	4.9964+13	3.5315-11	1.7163-08
64	13.620	1.8117+12	5.1775+13	2.8299-11	1.7191-08
65	12.373	1.8302+12	5.3590+13	2.2643-11	1.7213-08
66	9.776	1.8452+12	5.5411+13	1.8043-11	1.7232-08
67	7.733	1.8570+12	5.7236+13	1.4360-11	1.7246-08
68	6.106	1.8655+12	5.9063+13	1.1503-11	1.7257-08
69	4.807	1.8702+12	6.0903+13	8.9632-12	1.7265-08
70	3.801	1.8717+12	6.2743+13	7.1126-12	1.7273-08
71	2.933	1.8702+12	6.4586+13	5.5412-12	1.7279-08
72	2.301	1.8652+12	6.6431+13	4.2933-12	1.7283-08
73	1.837	1.8572+12	6.8277+13	3.2977-12	1.7286-08
74	.991	1.8436+12	7.0121+13	2.4034-12	1.7287-08
75	.000	1.8317+12	7.1963+13	1.6443-12	1.7287-08
76	.000	1.8205+12	7.3802+13	0.0000	1.7287-08
77	.000	1.8073+12	7.5627+13	0.0000	1.7287-08
78	.000	1.7897+12	7.7439+13	0.0000	1.7287-08
79	.000	1.7703+12	7.9290+13	0.0000	1.7287-08
80	.000	1.7497+12	8.1159+13	0.0000	1.7287-08
81	.000	1.7278+12	8.2867+13	0.0000	1.7287-08
82	.000	1.7049+12	8.4572+13	0.0000	1.7287-08
83	.000	1.6813+12	8.6253+13	0.0000	1.7287-08
84	.000	1.6570+12	8.7910+13	0.0000	1.7287-08
85	.000	1.6322+12	8.9542+13	0.0000	1.7287-08
86	.000	1.6069+12	9.1149+13	0.0000	1.7287-08
87	.000	1.5814+12	9.2731+13	0.0000	1.7287-08
88	.000	1.5556+12	9.4286+13	0.0000	1.7287-08
89	.000	1.5298+12	9.5816+13	0.0000	1.7287-08
90	.000	1.5038+12	9.7320+13	0.0000	1.7287-08
91	.000	1.4779+12	9.8798+13	0.0000	1.7287-08
92	.000	1.4520+12	1.0025+14	0.0000	1.7287-08
93	.000	1.4264+12	1.0166+14	0.0000	1.7287-08
94	.000	1.4009+12	1.0308+14	0.0000	1.7287-08
95	.000	1.3755+12	1.0443+14	0.0000	1.7287-08
96	.000	1.3504+12	1.0580+14	0.0000	1.7287-08
97	.000	1.3256+12	1.0713+14	0.0000	1.7287-08
98	.000	1.3010+12	1.0843+14	0.0000	1.7287-08
99	.000	1.2763+12	1.0971+14	0.0000	1.7287-08
100	.000	8.3331+13	1.1026+14	0.0000	1.7287-08

- 1) CARBON ATOM DISPLACEMENT CROSS SECTION (BARNES)
- 2) NEUTRONS PER SQUARE CENTIMETER PER SECOND IN THIS LEAD GROUP
- 3) NEUTRONS PER SQUARE CENTIMETER PER SECOND WITH ENERGIES GREATER THAN THE LOWER ENERGY FOR THIS GROUP
- 4) ATOMS DISPLACED PER ATOM PER SECOND BY NEUTRONS WITHIN THIS LEAD GROUP
- 5) TOTAL ATOMS DISPLACED PER ATOM PER SECOND BY NEUTRONS WITH ENERGIES GREATER THAN THE LOWER ENERGY LIMIT FOR THIS GROUP

TABLE IX. NEUTRON FLUX AND ATOMIC DISPLACEMENT RATES FOR GRAPHITE IN ZONE 55 OF THE O.R.N.L. CONCEPTUAL C.T.R., VERSION 5. DATA NORMALIZED TO 1.0 MEGAWATT PER SQ. METER WALL LOADING

LETHARGY GROUP NUMBER	DISPL. (1) CROSS SECTION	NEUTRON FLUX PER GROUP (2)	INTEGRAL NEUTRON FLUX (3)	DISPL. RATE PER GROUP (4)	INTEGRAL DISPL. RATE (5)
1	270.360	4.6706+11*	4.6706+11	1.2627-10	1.2627-10
2	297.460	3.5505+11	8.2211+11	1.0501-10	2.3187-10
3	315.600	2.2971+11	1.0518+12	7.2056-11	3.0392-10
4	265.210	2.0455+11	1.2587+12	5.4537-11	3.5846-10
5	247.470	1.4355+11	1.4002+12	3.5522-11	3.7396-10
6	254.750	2.2198+11	1.6222+12	5.6551-11	4.5053-10
7	435.300	1.7144+11	1.7558+12	6.5527-11	5.1446-10
8	255.830	2.0151+11	2.0557+12	6.0409-11	5.9487-10
9	348.870	1.7212+11	1.2072+12	6.0017-11	6.5489-10
10	345.670	2.4723+11	2.4545+12	6.5461-11	7.4035-10
11	418.070	2.1427+11	2.6537+12	5.5576-11	8.2992-10
12	503.310	2.2473+11	2.0557+12	1.1652-10	9.4626-10
13	643.560	2.2431+11	3.1283+12	1.4488-10	1.0927-09
14	746.060	1.9648+11	3.3250+12	1.4658-10	1.2595-09
15	844.760	1.4944+11	3.4744+12	1.2625-10	1.3658-09
16	566.740	2.1431+11	3.6887+12	1.2146-10	1.4872-09
17	708.670	2.1215+11	3.9009+12	1.5055-10	1.6376-09
18	525.210	3.1189+11	4.2128+12	1.6581-10	1.8014-09
19	501.860	3.2915+11	4.5419+12	1.6519-10	1.9666-09
20	575.110	3.5881+11	4.8807+12	1.9485-10	2.1614-09
21	522.260	3.7336+11	5.2541+12	1.9499-10	2.3564-09
22	538.370	3.6644+11	5.6225+12	1.9656-10	2.5548-09
23	556.400	3.8578+11	6.0085+12	2.1469-10	2.7694-09
24	574.460	3.8574+11	6.3941+12	2.2160-10	2.9910-09
25	591.510	3.7945+11	6.7735+12	2.2445-10	3.2155-09
26	637.090	3.7652+11	7.1495+12	2.2646-10	3.4439-09
27	620.940	3.6790+11	7.5177+12	2.2844-10	3.6724-09
28	632.700	3.6128+11	7.8790+12	2.2858-10	3.9009-09
29	642.250	3.5562+11	8.2326+12	2.2712-10	4.1261-09
30	647.470	3.4662+11	8.5795+12	2.2525-10	4.3535-09
31	654.210	3.3952+11	8.9190+12	2.2212-10	4.5754-09
32	656.520	3.3227+11	9.2512+12	2.1814-10	4.7956-09
33	656.350	3.2614+11	9.5774+12	2.1406-10	5.0076-09
34	653.930	3.2052+11	9.8977+12	2.0946-10	5.2171-09
35	649.100	3.1473+11	1.0212+13	2.0429-10	5.4214-09
36	641.830	3.0957+11	1.0522+13	1.9869-10	5.6201-09
37	632.210	3.0498+11	1.0827+13	1.9281-10	5.8129-09
38	620.100	3.0082+11	1.1128+13	1.8654-10	5.9994-09
39	603.770	2.9688+11	1.1425+13	1.7925-10	6.1767-09
40	585.210	2.9322+11	1.1718+13	1.7160-10	6.3503-09
41	565.200	2.8989+11	1.2008+13	1.6385-10	6.5141-09
42	544.020	2.8703+11	1.2295+13	1.5615-10	6.6703-09
43	521.810	2.8479+11	1.2580+13	1.4861-10	6.8189-09
44	498.830	2.8307+11	1.2863+13	1.4121-10	6.9601-09
45	475.410	2.8175+11	1.3144+13	1.3395-10	7.0940-09
46	451.620	2.8077+11	1.3425+13	1.2680-10	7.2208-09
47	427.820	2.8012+11	1.3705+13	1.1984-10	7.3407-09
48	404.210	2.7961+11	1.3985+13	1.1302-10	7.4537-09
49	380.810	2.7935+11	1.4264+13	1.0638-10	7.5601-09
50	341.050	6.9879+11	1.4963+13	2.3832-10	7.7964-09

* $\pm n$ represents $10^{\pm n}$

TABLE IX. (CONT.)

LETHARGY GROUP NUMBER	DISPL. (1) CROSS SECTION	NEUTRON FLUX PER GROUP (2)	INTEGRAL NEUTRON FLUX (3)	DISPL. RATE PER GROUP (4)	INTEGRAL DISPL. RATE (5)
53	287.510	7.0174+11	1.5065+13	2.0216-10	8.0002-09
54	255.810	7.0700+11	1.6372+13	1.8884-10	8.1690-09
55	193.450	7.1718+11	1.7033+13	1.3960-10	8.3006-09
56	157.230	7.2271+11	1.7809+13	1.1333-10	8.4225-09
57	125.510	7.3249+11	1.8541+13	8.2008-11	8.5145-09
58	95.850	7.4345+11	1.9281+13	7.4270-11	8.5888-09
59	72.560	7.5575+11	2.0040+13	5.9738-11	8.6487-09
60	63.004	7.6750+11	2.0808+13	4.8374-11	8.6971-09
61	49.956	7.8087+11	2.1589+13	3.9009-11	8.7361-09
62	39.642	7.9449+11	2.2383+13	3.1495-11	8.7676-09
63	31.423	8.0857+11	2.3192+13	2.5405-11	8.7930-09
64	24.906	8.2275+11	2.4014+13	2.0491-11	8.8135-09
65	19.729	8.3701+11	2.4851+13	1.6513-11	8.8300-09
66	15.620	8.5122+11	2.5703+13	1.3276-11	8.8433-09
67	12.373	8.6528+11	2.6568+13	1.0706-11	8.8540-09
68	9.778	8.7906+11	2.7447+13	8.5935-12	8.8626-09
69	7.733	8.9246+11	2.8339+13	6.9012-12	8.8695-09
70	6.106	9.0529+11	2.9245+13	5.5270-12	8.8750-09
71	4.807	9.1759+11	3.0162+13	4.4073-12	8.8794-09
72	3.801	9.2876+11	3.1091+13	3.5203-12	8.8830-09
73	2.963	9.3950+11	3.2030+13	2.8527-12	8.8857-09
74	2.301	9.4900+11	3.2975+13	2.3822-12	8.8877-09
75	1.837	9.5711+11	3.3927+13	1.9917-12	8.8894-09
76	.651	9.6482+11	3.4801+13	1.6733-12	8.8901-09
77	.006	9.7247+11	3.5695+13	1.4275-12	8.8901-09
78	.000	9.7929+11	3.6611+13	0.0000	8.8901-09
79	.000	9.8507+11	3.7550+13	0.0000	8.8901-09
80	.000	9.9086+11	3.8513+13	0.0000	8.8901-09
81	.000	9.9672+11	3.9501+13	0.0000	8.8901-09
82	.000	10.0254+11	4.0514+13	0.0000	8.8901-09
83	.000	10.0832+11	4.1552+13	0.0000	8.8901-09
84	.000	10.1413+11	4.2615+13	0.0000	8.8901-09
85	.000	10.1995+11	4.3704+13	0.0000	8.8901-09
86	.000	10.2578+11	4.4818+13	0.0000	8.8901-09
87	.000	10.3164+11	4.5957+13	0.0000	8.8901-09
88	.000	10.3754+11	4.7121+13	0.0000	8.8901-09
89	.000	10.4347+11	4.8311+13	0.0000	8.8901-09
90	.000	10.4943+11	4.9527+13	0.0000	8.8901-09
91	.000	10.5543+11	5.0769+13	0.0000	8.8901-09
92	.000	10.6147+11	5.2038+13	0.0000	8.8901-09
93	.000	10.6754+11	5.3334+13	0.0000	8.8901-09
94	.000	10.7364+11	5.4657+13	0.0000	8.8901-09
95	.000	10.7978+11	5.6007+13	0.0000	8.8901-09
96	.000	10.8597+11	5.7384+13	0.0000	8.8901-09
97	.000	10.9221+11	5.8789+13	0.0000	8.8901-09
98	.000	10.9850+11	6.0222+13	0.0000	8.8901-09
99	.000	11.0484+11	6.1683+13	0.0000	8.8901-09
100	.000	11.1124+11	6.3173+13	0.0000	8.8901-09

1) CARBON ATOM DISPLACEMENT CROSS SECTION (BARNs)

2) NEUTRONS PER SQUARE CENTIMETER PER SECOND IN THIS LETHARGY GROUP

3) NEUTRONS PER SQUARE CENTIMETER PER SECOND WITH ENERGIES GREATER THAN THE LOWER ENERGY FOR THIS GROUP

4) ATOMS DISPLACED PER ATOM PER SECOND BY NEUTRONS WITHIN THIS LETHARGY GROUP

5) TOTAL ATOMS DISPLACED PER ATOM PER SECOND BY NEUTRONS WITH ENERGIES GREATER THAN THE LOWER ENERGY LIMIT FOR THIS GROUP

TABLE A.

NEUTRON FLUX AND ATOMIC DISPLACEMENT RATES FOR GRAPHITE
IN ZONE 58 OF THE O.R.N.L. CONCEPTUAL C.T.R., VERSION-3
DATA NORMALIZED TO 1.0 MEGAWATT PER C.C. PETER WALL LOADING.

LETHARGY GROUP NUMBER	DISPL.(1) CROSS SECTION	NEUTRON FLUX PER GROUP (2)	INTEGRAL NEUTRON FLUX (3)	DISPL. RATE PER GROUP (4)	INTEGRAL DISPL. RATE (5)
1	270.360	1.8809+11*	1.8809+11	5.0852-11	5.0852-11
2	297.460	1.5519+11	3.4328+11	4.6163-11	9.7015-11
3	313.600	1.0203+11	4.4531+11	3.1996-11	1.2901-10
4	266.210	9.2669+10	5.3796+11	2.4689-11	1.5388-10
5	247.490	6.7355+10	6.0531+11	1.6685-11	1.7034-10
6	234.760	1.0707+11	7.1238+11	2.7280-11	1.9782-10
7	400.350	5.0103+10	7.7249+11	2.7234-11	2.2608-10
8	203.030	1.0415+11	7.1424+11	4.0007-11	2.5755-10
9	740.670	4.3707+10	7.7314+11	2.7222-11	2.7677-10
10	343.670	1.1000+11	1.1170+12	4.1007-11	3.3788-10
11	418.070	1.0544+11	1.2274+12	4.4247-11	3.8210-10
12	503.510	1.1127+11	1.3347+12	5.6046-11	4.3815-10
13	543.350	1.0595+11	1.4407+12	6.8175-11	5.0633-10
14	746.000	7.3035+10	1.5543+12	6.9871-11	5.7620-10
15	844.760	7.3037+10	1.6094+12	6.3390-11	6.3939-10
16	566.740	1.0543+11	1.7157+12	6.0264-11	6.9985-10
17	708.670	1.0348+11	1.8192+12	7.3333-11	7.7310-10
18	525.210	1.4452+11	1.9667+12	7.8230-11	8.5171-10
19	501.860	1.5709+11	2.1278+12	7.9837-11	9.3155-10
20	575.110	1.5223+11	2.2900+12	9.3302-11	1.0249-09
21	522.260	1.7676+11	2.4668+12	9.2314-11	1.1172-09
22	538.370	1.7666+11	2.6434+12	9.5109-11	1.2123-09
23	556.400	1.6504+11	2.8285+12	1.0296-10	1.3152-09
24	574.480	1.7567+11	3.0152+12	1.0724-10	1.4225-09
25	591.510	1.6585+11	3.2010+12	1.0723-10	1.5324-09
26	607.070	1.5775+11	3.3810+12	1.1222-10	1.6455-09
27	520.940	1.5336+11	3.5602+12	1.1625-10	1.7595-09
28	612.700	1.5307+11	3.7370+12	1.1954-10	1.8754-09
29	642.230	1.7185+11	3.9227+12	1.1630-10	1.9922-09
30	677.470	1.5717+11	4.1115+12	1.1732-10	2.1100-09
31	634.210	1.5131+11	4.2985+12	1.1681-10	2.2286-09
32	550.340	1.6077+11	4.4816+12	1.1633-10	2.3474-09
33	556.350	1.6088+11	4.6605+12	1.1672-10	2.4662-09
34	652.930	1.8025+11	4.8407+12	1.1787-10	2.5840-09
35	649.100	1.7820+11	5.0189+12	1.1567-10	2.6997-09
36	641.830	1.7385+11	5.1928+12	1.1158-10	2.8113-09
37	532.210	1.5715+11	5.3617+12	1.0394-10	2.9182-09
38	620.100	1.6175+11	5.5257+12	1.0050-10	3.0185-09
39	605.770	1.5046+11	5.6741+12	9.0850-11	3.1074-09
40	585.210	1.5051+11	5.8105+12	7.9770-11	3.1871-09
41	565.200	1.2365+11	5.9341+12	6.7384-11	3.2590-09
42	544.070	1.2368+11	6.0580+12	6.7373-11	3.3284-09
43	521.610	1.3034+11	6.1765+12	7.2297-11	3.3987-09
44	456.610	1.3442+11	6.3005+12	7.7050-11	3.4707-09
45	475.410	1.6505+11	6.4200+12	6.0309-11	3.5381-09
46	431.620	1.7034+11	6.5395+12	7.9711-11	3.6060-09
47	427.820	1.7437+11	6.6713+12	7.4600-11	3.7106-09
48	404.210	1.7004+11	7.0715+12	6.6575-11	3.7796-09
49	360.610	1.6544+11	7.2104+12	6.4743-11	3.8437-09
50	341.050	4.0292+11	7.6135+12	1.3145-10	3.9812-09

* In represents 10^{1n}

TABLE A. (CONT.)

LETHARGY GROUP- NUMBER	1) SPL. (1) CROSS SECTION	NEUTRON FLUX PER GROUP (2)	INTEGRAL NEUTRON FLUX (3)	DISPL. RATE PER GROUP (4)	INTEGRAL DISPL. RATE (5)
51	287.510	3.7956+11	7.9429+12	1.0915-10	4.0903-09
52	238.810	3.6042+11	8.3553+12	8.6075-11	4.1784-09
53	195.480	3.4468+11	8.6987+12	8.7416-11	4.2430-09
54	157.580	3.3243+11	9.0306+12	9.2384-11	4.2982-09
55	129.510	3.2216+11	9.3530+12	9.0492-11	4.3387-09
56	97.597	3.1424+11	9.6872+12	9.1392-11	4.3680-09
57	73.103	3.0776+11	9.9740+12	9.4407-11	4.3972-09
58	63.004	3.0207+11	1.0277+13	9.9032-11	4.4115-09
59	49.556	2.9748+11	1.0574+13	1.4881-11	4.4265-09
60	39.642	2.9371+11	1.0868+13	1.1643-11	4.4380-09
61	31.942	2.9202+11	1.1160+13	9.1760-12	4.4472-09
62	24.905	2.9007+11	1.1450+13	7.2244-12	4.4544-09
63	19.729	2.8565+11	1.1736+13	5.6355-12	4.4600-09
64	15.620	2.8193+11	1.2018+13	4.4048-12	4.4644-09
65	12.373	2.8121+11	1.2302+13	3.9120-12	4.4677-09
66	9.778	2.8012+11	1.2588+13	3.1391-12	4.4707-09
67	7.157	2.7754+11	1.2889+13	2.4440-12	4.4726-09
68	5.106	2.7480+11	1.3189+13	1.8717-12	4.4745-09
69	4.807	2.7148+11	1.3480+13	1.3040-12	4.4759-09
70	3.501	2.6777+11	1.3803+13	1.0176-12	4.4768-09
71	2.965	2.6334+11	1.3937+13	7.8784-13	4.4778-09
72	2.301	2.5240+11	1.4201+13	6.0386-13	4.4782-09
73	1.587	2.4043+11	1.4467+13	4.1320-13	4.4786-09
74	.651	2.2742+11	1.4721+13	1.8875-13	4.4788-09
75	.006	2.5461+11	1.4776+13	1.5022-13	4.4788-09
76	.000	2.5502+11	1.5231+13	0.0000	4.4788-09
77	.000	2.5138+11	1.5763+13	0.0000	4.4788-09
78	.000	2.5194+11	1.5733+13	0.0000	4.4788-09
79	.000	2.5129+11	1.5586+13	0.0000	4.4788-09
80	.000	2.5049+11	1.6237+13	0.0000	4.4788-09
81	.000	2.4915+11	1.6486+13	0.0000	4.4788-09
82	.000	2.4760+11	1.6735+13	0.0000	4.4788-09
83	.000	2.4647+11	1.6980+13	0.0000	4.4788-09
84	.000	2.4510+11	1.7225+13	0.0000	4.4788-09
85	.000	2.4377+11	1.7469+13	0.0000	4.4788-09
86	.000	2.4229+11	1.7711+13	0.0000	4.4788-09
87	.000	2.4075+11	1.7952+13	0.0000	4.4788-09
88	.000	2.3912+11	1.8191+13	0.0000	4.4788-09
89	.000	2.3740+11	1.8428+13	0.0000	4.4788-09
90	.000	2.3583+11	1.8664+13	0.0000	4.4788-09
91	.000	2.3384+11	1.8898+13	0.0000	4.4788-09
92	.000	2.3200+11	1.9130+13	0.0000	4.4788-09
93	.000	2.3012+11	1.9360+13	0.0000	4.4788-09
94	.000	2.2818+11	1.9588+13	0.0000	4.4788-09
95	.000	2.2613+11	1.9814+13	0.0000	4.4788-09
96	.000	2.2406+11	2.0038+13	0.0000	4.4788-09
97	.000	2.2188+11	2.0260+13	0.0000	4.4788-09
98	.000	2.1969+11	2.0480+13	0.0000	4.4788-09
99	.000	2.1741+11	2.0697+13	0.0000	4.4788-09
100	.000	1.2635+12	3.3331+13	0.0000	4.4788-09

- 1) CARBON ATOM DISPLACEMENT CROSS SECTION (BARNS)
- 2) NEUTRONS PER SQUARE CENTIMETER PER SECOND IN THIS LETHARGY GROUP
- 3) NEUTRONS PER SQUARE CENTIMETER PER SECOND WITH ENERGIES GREATER THAN THE LOWER ENERGY FOR THIS GROUP
- 4) ATOMS DISPLACED PER ATOM PER SECOND BY NEUTRONS WITHIN THIS LETHARGY GROUP
- 5) TOTAL ATOMS DISPLACED PER ATOM PER SECOND BY NEUTRONS WITH ENERGIES GREATER THAN THE LOWER ENERGY LIMIT FOR THIS GROUP

TABLE XI. Flux Attenuation Factors

Region	Flux Attenuation Factor, a (cm^{-1})	
	For ϕ ($E > 10$ MeV)	For ϕ_G
First Li Blanket	0.0536	0.0210
Graphite	0.0855	0.0675
Second Li Blanket	0.0408	0.0355

The fluxes shown in Figure 4 are normalized to a total reactor power of 1000 MW thermal. Figure 4 also shows the heat generation rate^{*} in the graphite which was determined by Don Steiner⁽¹²⁾ of ORNL assuming a graphite density of 1.6 g/cm^3 .

Graphite Behavior

Graphite temperatures in Figure 5 were calculated from the heat generation data shown in Figure 4. The lithium temperature was taken to be 1025°C which is midway between the extremes of 996 and 1056°C given by Fraas.⁽²⁾ Both curves in Figure 5 were calculated assuming that heat transfer takes place only radially inward and outward and using a thermal conductivity value for graphite of $0.1 \text{ cal/sec-cm-}^\circ\text{C}$ which is fairly typical for graphite irradiated near 1000°C .⁽¹³⁾ Early in the operating life of the reactor, before the thermal conductivity of the graphite decreases to the assumed value, the temperature gradient through the graphite will be less. The lower curve was calculated assuming perfect heat transfer between the graphite and the lithium while the upper curve was calculated assuming only radiative heat transfer between the graphite and the lithium. The upper curve is reasonable from the standpoint that the graphite operates in a vacuum and, due to radiation-induced dimensional changes, will shrink away from the niobium structure containing the lithium. However, there will always be some contact so the actual graphite temperatures will lie somewhere between the two extreme curves shown.

Radiation-induced dimensional changes in graphite tend to be most severe over the temperature range 1000 to 1200°C from the standpoint that

* The heat generation rates given here are for 0.43 MW/m^2 neutron wall loading whereas those given in Ref. 2 are for 1 MW/m^2 neutron wall loading.

the expansion back to its original length (or volume) occurs quicker than at either lower or higher temperatures.⁽¹⁴⁾ This is equivalent to saying that the useful lifetime of most graphites is shortest in this temperature range.

Looking at the design of the blanket shown in Figure 2, one can see that the graphite should not be allowed to expand much, if at all, beyond its original dimensions because this would give rise to a circumferential compressive stress which would tend to make the whole blanket expand. The critical point is the inner surface of the graphite which receives the highest flux as shown in Figure 4. The flux at that point is equivalent to 4.3×10^{21} n/cm² per 1000 full power days of operation. It is obvious, therefore, that a lifetime of only 8 to 10 years, assuming a 60% operating efficiency, could be obtained from a graphite similar to the rather typical nuclear graphite for which data were presented in reference 1. Better graphites can be obtained, however, and we have chosen grade 9640 (See Table XII) from among a large number of graphites irradiated at this laboratory^(15,16,17) as being representative of the better grade graphites and one for which radiation data are available for the temperatures and neutron fluences of interest. This graphite also has the advantage of being fairly isotropic and its dimensional changes, which are shown in Figure 6, are reasonably temperature independent over the range 1000-1200°C. The cost of producing large quantities of graphite similar to 9640 is expected to be in the range of 1 to 2 dollars per pound in the unmachined condition⁽¹⁸⁾ which is more expensive than extruded needle-coke graphites but modest compared to some of the graphites that might be considered.

Using the most pessimistic dimensional-change curve in Figure 6 (i.e., the 1000-1100°C curve for the direction parallel to the molding direction), the dimensional-changes in the circumferential direction as a function of radial position were plotted in Figure 7 after various full power days (FPD) of operation. It can be seen that the lifetime of this graphite in this reactor would be limited to 2500-3000 FPD if one does not wish to let the graphite expand beyond its original dimensions. It should be noted that

TABLE XII

PROPERTIES OF GRADE 9640 GRAPHITE

MANUFACTURER: Airco, Speer

COKE: Needle

FORMING METHOD: Molded

GRAPHITIZATION TEMPERATURE: 2800°C

DENSITY (g/cm³): 1.6

COEFFICIENT OF THERMAL EXPANSION
(25-425°C) x 10⁶: Parallel* = 3.6
Perpendicular* = 3.8

YOUNG'S MODULUS, psi x 10⁻⁶: Parallel* = 1.24

*Directions are relative to the molding direction.

the total dimensional change in the radial direction is given approximately (this assumes that the graphite is isotropic which is not quite true) by the integral of the curves in Figure 7 for any given time.

Radiation-Induced Stresses

In addition to radiation-induced dimensional changes, one will have to be concerned with stresses induced in the graphite due to thermal gradients and gradients in the radiation-induced dimensional changes. The latter result from different dimensional-change rates at the different temperatures, and also due to flux gradients. Because the temperature gradients are small in the present case, we will concern ourselves only with stresses due to the flux gradient which is shown in Figure 5.

We have simplified the calculation by assuming that the graphite is in the shape of a parallelepiped which does not distort and calculated stresses in the Y (circumferential) direction due to a flux gradient in the X(radial) direction. These are given by⁽¹⁹⁾

$$\sigma_Y = \int_0^1 \frac{r(\phi_X)}{K(\phi_X)} \left[1 - e^{-K(\phi_X) E(\phi_X) \phi_X/2} \right] dX - \frac{r(\phi_X)}{K(\phi_X)} \left[1 - e^{-K(\phi_X) E(\phi_X) \phi_X/2} \right] \quad (4)$$

where $r(\phi_X)$ is the unstressed dimensional change rate

$K(\phi_X)$ is the steady-state radiation creep coefficient

$E(\phi_X)$ is the elastic modulus

ϕ_X is the equivalent fission fluence

The parameters $r(\phi_X)$, $K(\phi_X)$, and $E(\phi_X)$ are fluence dependent and the fluence, in turn, depends upon position in the X direction. The above equation has been normalized to a length in the X direction of 1 and it also assumes that the transient creep in graphite takes place instantly and then saturates. The latter is a good assumption for fluences above about $1 \times 10^{21} \text{ n/cm}^2$.

The creep coefficient, $K(\phi_X)$, is known to be about the same in tension and compression and it is also known to be independent of stress and fluence at least up to about $7 \times 10^{21} \text{ n/cm}^2$.^(3,19) We have, therefore, assumed that

it is fluence independent even for fluences above 7×10^{21} n/cm² although we will have more to say about that later. The elastic modulus increases fairly quickly due to radiation but then remains fairly constant. We have therefore assumed that it is fluence independent and used its saturation value after irradiation. We have also assumed the same value in tension and compression. With these assumptions, equation (4) can be rewritten

$$\sigma_y = \frac{1}{K} \int_0^1 r(\phi_x) [1 - e^{-KE\phi_x/2}] dx - \frac{r(\phi_x)}{K} [1 - e^{-KE\phi_x/2}]. \quad (5)$$

Now ϕ_x as shown in Figure 4 is given by

$$\phi_x = \phi_0 e^{-ax}$$

Changing variables and dropping the x subscript:

$$\sigma_y = \frac{-1}{aK} \int_{\phi_0}^{\phi_0 e^{-a}} r(\phi) [1 - e^{-KE\phi/2}] \frac{d\phi}{\phi} - \frac{r(\phi)}{K} [1 - e^{-KE\phi/2}]. \quad (6)$$

Finally, the dimensional change curve in Figure 6 can be fit very well with a polynomial of degree 3 such that

$$\frac{\Delta L}{L} = A + B\phi + C\phi^2 + D\phi^3 \quad (7)$$

then:

$$r(\phi) = \frac{d\left(\frac{\Delta L}{L}\right)}{d\phi} = B + 2C\phi + 3D\phi^2 \quad (8)$$

Therefore:

$$\sigma_y = \frac{-1}{aK} \int_{\phi_0}^{\phi_0 e^{-a}} (B + 2C\phi + 3D\phi^2) (1 - e^{-KE\phi/2}) \frac{d\phi}{\phi} - \frac{(B + 2C\phi + 3D\phi^2)}{K} (1 - e^{-KE\phi/2}) \quad (9)$$

Using a numerical technique to evaluate the integral in equation (9), the stresses as a function of radial position shown in Figure 8 were derived after various full power days of operation. Negative stresses are compressive and positive stresses are tensile. The stresses shown are less than the fracture stress in either tension or compression even after 3500 operating days. However, several assumptions render these stresses only approximations.

The most serious assumptions made were uniaxial stress and no distortion of the graphite block. Neither of these conditions apply in the real case. Also, recent radiation creep data⁽³⁾ indicate that the steady-state creep coefficient may decrease drastically at fluences higher than about $7 \times 10^{21} \text{ c/cm}^2$. If this is confirmed, the graphite stresses at the higher fluences would be much higher than those shown in Figure 8.

SUMMARY AND CONCLUSIONS

We have estimated the graphite performance in Version III of the ORNL conceptual fusion reactor described by Fraas.⁽²⁾ The neutron spectra in the graphite for this design were provided by Kriese⁽⁴⁾ and we have related this to the "equivalent fission fluence for damage in graphite" (Φ_G) using group average displacement cross sections provided by Jenkins⁽⁹⁾ who used, in turn, the Thompson and Wright form of the atomic displacement weighting function. Graphite temperatures were calculated using heat generation rates supplied by Steiner. The temperature and fluence experienced by the graphite determine the dimensional changes (and changes in other physical properties as well) for a given type of graphite. Using these two parameters, we have determined the dimensional changes for the ORNL reactor using data for type 9640 graphite which is representative of graphite that could probably be obtained in large quantities for one to two dollars per pound.

In the course of this study we have made some observations and assumptions that should be noted and qualified as follows:

- 1) The heat generation rate in the graphite is quite low resulting in a small temperature gradient thus reducing the tendency for internal stresses.
- 2) Graphite temperatures range from 1000 to 1200°C which is a worst temperature for most graphites in the sense that graphite will shrink and then expand back to and beyond its original dimensions sooner than at either lower or higher temperatures. One should be cautioned, however, against immediately concluding that higher graphite temperatures are desirable because corrosion problems will become increasingly severe at higher temperatures. Also, while experimental graphite radiation data are available to ~1500°C, there is presently no actual fission-reactor operating experience with graphite moderator temperatures above 1200°C.

- 3) The graphite in the ORNL fusion reactor concept is open to the plasma vacuum. This may present outgassing problems.
- 4) Better graphite radiation damage data and models are needed at the higher neutron energies as described in more detail elsewhere.⁽¹⁾
- 5) We have assumed that the graphite is isotropic. This is not quite true, and other graphites that could be selected may be even more anisotropic. Thus, the dimensional changes will depend somewhat on orientation whereas we have assumed they are not. It should be noted here, however, that anisotropy is not necessarily detrimental; it can be used to advantage.
- 6) Graphite stresses were calculated with the following simplifying assumptions:
 - a) uniform temperature
 - b) isotropic graphite
 - c) elastic modulus is independent of fluence
 - d) transient creep occurs instantly and is equal to the elastic strain
 - e) steady-state creep coefficient is the same in tension and compression
 - f) uniaxial stress
 - g) no distortion of the graphite prism
 - h) steady-state creep coefficient is independent of fluence.

The first five assumptions are reasonable but the last three will have to be replaced with more realistic approximations as reactor designs become more nearly finalized. In particular, recent data indicate that the steady-state creep coefficient may decrease drastically at fluences above about $7 \times 10^{21} \text{ n/cm}^2$.⁽³⁾ While only a very limited amount of data are available in this region, they emphasize the need for more experimental data above $7 \times 10^{21} \text{ n/cm}^2$ and under a wider variety of conditions and types of graphite before meaningful stress calculations can be performed at high fluences.

Within the limitations of the above assumptions, the following conclusions can be drawn:

- 1) Type 9640 graphite will, as will most graphites, shrink initially but will eventually expand back to and beyond its original dimensions. If one makes the reasonable assumption that the graphite should not be allowed to expand beyond its original dimensions, then type 9640 graphite will have a lifetime in the ORNL reactor of 2500 to 3000 full power days of operation or about 13 years at 60% operating efficiency. This is the time beyond which the inside portion of the graphite will be expanding beyond its original dimensions.
- 2) Graphite was shown to be about twice as good for fast neutron attenuation as lithium and one might therefore conclude that the lithium blanket should be reduced to the minimum thickness required for tritium breeding and that graphite be used for neutron moderation.
- 3) Calculated stresses due to the flux gradient were shown to be appreciable but less than the fracture stress throughout the lifetime of the graphite. However, as pointed out above, the stresses will have to be calculated under conditions that more closely approximate the real case when reactor designs become more nearly finalized.

TOROIDAL FUSION REACTOR (1000 MWt)

FIGURE 1. OVERALL VIEW OF ORNL CONCEPTUAL FUSION REACTOR

FIGURE 2. CROSS SECTIONAL VIEW THROUGH ONE LITHIUM/GRAPHITE SEGMENT

FIGURE 3. NEUTRON FLUXES VS BLANKET POSITION FOR 1 MW/m² WALL LOADING

FIGURE 4. NEUTRON FLUX AND HEAT GENERATION RATE IN GRAPHITE AT 0.43 MW/m^2 WALL LOADING

FIGURE 5. GRAPHITE TEMPERATURES

FIGURE 6. LENGTH CHANGES OF 9640 TYPE GRAPHITE

FIGURE 7. LENGTH CHANGES OF TYPE 9640 GRAPHITE AFTER VARIOUS FULL POWER DAYS OF OPERATION

FIGURE 8. STRESS IN GRAPHITE AFTER VARIOUS FULL POWER DAYS OF OPERATION

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