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SHIELDING ANALYSIS FOR THE FAST FLUX TEST FACILITY

By

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1.0 SUMMARY

The shielding requirements for the Fast Flux Test Facility were investigated and an initial specification of required types and thicknesses of shield materials was obtained.

The radial shield specification was 3.5 ft. of iron plus 5 ft. of graphite. The axial shield was specified to consist of 3 ft. of iron above the 30' sodium pool. A number of alternate materials for the shield were also investigated. The best of these alternate materials was found to consist of a 5' layer of borated Fe_2O_3 in place of the 5' radial graphite shield. The borated Fe_2O_3 produced a secondary sodium activation about a factor of 8 less than the graphite shield. Furthermore, the use of borated Fe_2O_3 could very possibly result in a cost saving of approximately one million dollars by the elimination of the graphite and its associated inert gas systems.

Complete neutron flux traverse calculations for the axial shield and the radial shield are presented both for the reference design and for a number of alternate materials. The results of parametric studies on a number of alternate shield materials are also presented. The results of variations in shield thickness for the reference design are also given in the report.

2.0 INTRODUCTION

2.1 Objective

The objective of this work was to perform a conceptual design study on the shielding requirements for the fast flux test facility (FFTF). The output of such a study is the required types of materials and required thicknesses of materials to provide adequate radiation shielding.

2.2 Background

A fast reactor of the power levels contemplated constitutes a large source of high energy neutrons. The shielding of these high energy neutrons is the fundamental shielding engineering problem in a fast reactor as the core gammas and low energy neutrons are of little importance. The shielding problem is made more difficult due to limitations of temperature levels and safety considerations since the use of hydrogenous shielding materials (which are the most desirable for the shielding of high energy neutrons) is somewhat limited, and one is forced to use materials such as carbon and iron. As is the case of shielding of thermal reactors there is no fundamental feasibility problem in arriving at the allowable radiation levels; the problem is one of economics. One obviously wants to construct a shield which performs the necessary functions at the lowest possible cost. In addition, in the early stages of a project such as the present phase of the FFTF, it is necessary to determine the approximate dimensions of the shielding materials in order that a correct layout of the equipment surrounding the reactor core can be made.

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The reactor core description used for the shielding calculations was taken from BNWL-71 entitled "Progress Report Fast Flux Test Facility Studies". The PuO_2 volume percent was taken as 7%, the core volume was 400 liters and the L/D was 1. The core power level for the shielding calculations was taken as 500 MW.

At present the most useful methods of calculation for a fast reactor shield are diffusion theory, transport theory and Monte Carlo methods. Each method is appropriate at various stages in the shield design. Clearly in the early stages of the shield design one is interested in carrying out many survey calculations in order to establish the effect of using the different materials and the consequences of changing dimensions, etc. As the design progresses and becomes more fixed interest changes to obtaining more precise numerical values of the design.

Obviously at this stage in the design one would want to use the fastest methods in order that the necessary survey calculations be carried out. In this case, the fastest method would be diffusion theory with the use of existing codes. The most obvious deficiency of diffusion theory calculations for shielding purposes is that when one considers the problem of neutron attenuation in thick shields it is quickly determined that the neutrons which emerge from a shield are derived primarily from the very high energy neutrons initially incident on the inner part of the shield. In treating these neutrons diffusion theory considers all scattering processes to be isotropic whereas in fact the scattering of very high energy

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neutrons is quite anisotropic. Hence, diffusion theory is optimistic inasmuch as it ignores the preferential forward scattering of high energy neutrons. However, in spite of this deficiency, by the judicious choice of cross sections and by the proper normalization of diffusion theory results with more accurate analytical procedures and with experiments, it is possible to obtain very useful results with diffusion theory. In our particular case, we are using the AIM-6 code with the particular set of cross sections that were used to calculate the neutron shield on the Enrico Fermi Fast Breeder Reactor. In this case, there are experimental results now available for this shield so that it is possible to quite accurately normalize the results from these particular calculations. The Fermi data indicate that the particular cross section set used in AIM-6 outwardly results in calculated doses that are somewhat higher than those measured. This is a desirable situation for the present phase inasmuch as it means that the results we are using are in fact slightly conservative and this is the appropriate bias one would want to introduce in the early stages of a design. On the other hand, as far as comparative calculations in parametric surveys the results provided by the present diffusion theory calculations are quite good and provide correct conclusions at this stage.

2.3 Scope

Our attention has been primarily directed to the primary shield which is the shield immediately around the reactor. This shield has several functions. They are as follows:

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- (1) Reduce the neutron flux impinging on the sodium vessel so that neutron embrittlement of the reactor vessel is within allowable limits.
- (2) Reduce the neutron flux impinging on the sodium heat exchangers so that activation of the secondary sodium is kept within allowable limits.
- (3) Reduce radiation levels in inhabited regions to below biological tolerances.

The assumed allowable flux levels for the above items are as follows:

- (1) The total integrated neutron flux above 0.1 mev during a 20 year lifetime is conservatively taken to be no greater than 10^{21} nvt on the steel sodium vessel.
- (2) The activation of the sodium in the secondary cooling circuit is to be less than 2×10^{-4} micro curies/cc.
- (3) The radiation levels in any occupied regions must be less than .25 mr/hr. This radiation limit provides for a safety factor of 10 over that specified by AEC regulations and is an appropriate design goal.

3.0 DISCUSSION

3.1 General

The present phase of the shielding study can be divided into two areas. The first is an investigation of the shield radially outward from the core and the second is the investigation of the shield axially outward above the top of the core. These two shield regions are shown schematically depicted in Figure 1. Section AA shows the radial shield and Section BB shows the axial shield.

The present conceptual design study has concerned itself primarily with the so-called "ideal" shield. That is, shields which ignore the many penetrations, ducts, crevices, etc., which actually exist in a final fabricated shield. The ideal shield is treated because obviously this is the starting point in a conceptual design study and it does establish the fundamental dimensions and materials of the shield. Nevertheless, one finds that the bulk of the shielding effort eventually goes into treating all these various discontinuities in the shield. The study of these discontinuities is not expected to produce any major changes in the dimensions of materials of the ideal shield but could very well result in specific local design changes for the proper handling of each individual discontinuity. One particularly significant area that requires considerable investigation are the sodium ducts through the primary shield. These ducts are important because of their very large size (approximately 30" in diameter). In such a large size sodium duct there is the capability of considerable transmission of neutrons.

3.2 Radial Shield

As shown in Figure 1 the shielding materials in Section AA are as follows: Starting from the core the first material is the Inconel reflector. The purpose of the Inconel is to reduce the critical mass of the reactor and not primarily that of shielding. Nevertheless, it does form an integral part of the shield. This is particularly true because of the high scattering cross section of nickel for intermediate energy neutrons. Following the Inconel there is a sodium region which is then followed by a steel region. The function of this latter steel region is to reduce the neutron flux level at the primary sodium vessel to tolerable levels. Its thickness is therefore predicated on this function. This steel reflector is within the primary sodium vessel and is cooled by the primary sodium. The steel neutron shield is followed by a graphite shield outside the primary sodium vessel. The thickness of this graphite region is primarily determined by the allowable activation levels in the secondary sodium in the intermediate heat exchangers. The secondary system is essentially non-radioactive and the activation of the secondary sodium is to be kept down to tolerable levels.

The graphite as presently contemplated is contained in iron pipes with inert gas cooling provided. If necessary, boron can be added to a part of the graphite shield if further low energy neutron absorption is needed. If this boron is to be provided for the aforementioned function, namely, that of reducing activation of the secondary sodium it need only be used in the

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outer part of the graphite shield. However, there is another function that boron provides and that is one of reducing the heat production in the graphite shield due to the energy release deriving from neutron absorption within the shield. If the steel pipes are the primary neutron absorbers within a graphite shield, one has an energy release of approximately 10 mev per neutron absorbed in the steel. Whereas in the case of boron the energy release is only 2.8 mev per neutron absorbed in the boron. Furthermore, the presence of the boron can be used to change the spatial distribution of the heat generation. For example, in the Fermi reactor an inner layer of borated graphite was used in the graphite shield in order that the heat production be concentrated here. This region was within an insulated barrier which was used to provide heat flow into the primary sodium vessel. The remainder of the heat generated in the graphite reflector had to be removed by conduction through several feet of graphite. Therefore it was desirable to have the heat production in the outer layers of graphite as low as possible.

Another part of the radial shield will be a steel facing on the concrete outside the intermediate heat exchangers. This will probably be necessary in order to keep heating within the concrete down to permissible levels. This steel layer will reduce considerably the gamma radiation incident on the concrete. It will also reduce the neutron flux impinging on the concrete.

In analyzing the radial shield, attention was given to various alternate materials that might be used in place of the graphite neutron shield. The following materials were examined:

graphite, borated graphite, iron, iron oxide, borated iron and borated iron oxide. A convenient method of comparing the effectiveness of each of these shield materials is to examine the sodium activation produced in each case. (See Table 1, Section 4.0 for further discussion).

It is interesting to note that the borated iron oxide provides the lowest sodium activation. Therefore this material may be a possible alternate to the graphite. The primary reason for considering the alternate would be that of economics (See Section 5.0). However, there may be a preference for the graphite as this is presently used as a fast neutron shield and has been very well explored.

3.3 Axial Shield

The basic design problem in the axial shield is one of getting the radiation levels down to biological tolerance at the operating floor above the reactor. Starting at the core the first shield region is the sodium pool. (The sodium is not there primarily for shield purposes but 30 feet of sodium provides a very significant amount of attenuation of the neutron flux). In addition to the sodium in this region there is also the steel which is derived from the extensions of the tubes in the reactor. The steel fraction is continually decreasing as one goes up through the sodium because of the skewed design of the core. Above the sodium there is the final neutron and gamma shield. The present reference design depicts this as a steel region. Alternate materials may be considered for this region. A study similar to that of the radial shield was performed. Again borated iron oxide was found to produce the lowest neutron dose rate for a given shield thickness. (See Table 2). In the case of the

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axial shield the important parameter is the neutron dose rate (rather than the secondary sodium activation as in the case of the radial shield). It is significant to note that in all cases the neutron dose is well below the design goal of .025 mr/hr. Thus, it can be concluded that with a 30' sodium pool little additional neutron shielding is required. Of course, gamma shielding is still needed for the sodium decay gammas and for capture gammas. Further, as will be pointed out below, if the thickness of the sodium pool is later decreased, the neutron dose rate goes up rapidly. For these reasons, at this stage in the design, it is prudent to indicate something in the order of a three foot shield above the sodium pool.

A comparison of shields similar to that above was also made for a 10' sodium pool. (See Table 3). The most striking result is that the dose rate for all materials goes up very much (in the order of 10^{10}). Therefore, particular attention should be paid to the effect on shield requirements of any proposed reduction in the thickness of the sodium pool.

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4.0 RESULTS

The radiation limits for the shield design are derived from various sources as mentioned in Section 2.3 and are described in a general form in Table 4 below under the heading, Radiation Limits (General). However, the radiation limits as given in their general form are not in the same form as the output data obtained in standard computer calculations. Therefore, these radiation limits were put into a more usable form as indicated in the column, Radiation Limits (As used in the calculations).

TABLE 4
RADIATION LIMITS

<u>Location</u>	<u>Radiation Limits (General)</u>	<u>Radiation Limits (As used in Calculations)</u>
Sodium vessel	Fast nvt of 10^{21} in 20 years	$0 - 1.6 \times 10^{12}$ ($E > .1$ mev)
Secondary Sodium HX	Specific activity 2×10^{-4} /cc	Total Secondary Sodium activity - 0.1 curie
Operation flux above reactor	Total dose - .25 mr/hr	Neutron dose - .025 mr/hr

It should be noted that the total dose is composed of two components, the neutron dose and the secondary gamma dose. The neutron dose was calculated directly and in cases like this is about a factor of ten less than the total dose. Therefore the criterion used was that the neutron dose was to be no greater than .025 mr/hr.

The required shield thickness to obtain the above radiation limits are given in Table 5. This table includes both the as-calculated values and the suggested design values. The radial iron shield thickness was obtained from Figure 1. The radial graphite shield thickness was obtained from Figure 3. The top axial shield thickness was obtained from Table 2. In the earlier stages of a design it is desirable to have the shield thicknesses somewhat on the conservative side as it is easier to decrease the required shield thickness at a later date than to have to increase it. Clearly an increase in shield thickness can cause a perturbation in the entire plant layout. Based on previous experience with other fast reactor shield designs it is reasonable to add approximately 25% to the as-calculated thickness of these parts of the shield at this earlier stage in the design. This also provides allowance for ducts, crevices, and other penetrations. In case of the top axial iron shield, as mentioned previously, calculations indicate that with a 30' sodium pool no further neutron shielding is required (See Table 2). However, because of gamma shielding requirements and possible changes in the sodium pool thickness a three foot shield is recommended at this stage in the design.

TABLE 5
REQUIRED SHIELD THICKNESSES

	<u>As-Calculated</u>	<u>Suggested Design Value</u>
Radial iron shield	2.8 feet	3.5 feet
Radial graphite shield	4 feet	5 feet
Top axial iron shield	3 feet	3 feet

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Table 6 presents a description of the zoned core case used in the multigroup calculation of the radial shield. This zoned core was used in the earlier calculations and is the one described in the report, "Fast Fuel Test Reactor Study" by Albaugh and Holsted dated April 1, 1964. The table contains the dimensions and the atomic compositions of all of the regions in the reactor and shield.

Table 7 contains the same material for the zoned core case used in the multigroup calculations of the axial shield.

Table 8 contains a description of the single region core and the associated radial shield. This is the core described in BNWL-71. The particular core chosen in this study was the one which had a core composition of 7 volume per cent PuO_2 , a core volume of 400 liters and an L/D of 1.

A comparison of various shield materials which can be used in the radial shield is presented in Table 1, previously mentioned. The basis of comparison is the neutron activation in the secondary sodium outside the neutron shield. In all cases, the total neutron shield thickness was kept fixed at six feet. The different shield materials considered were as follows:

- (1) A combination of graphite and borated graphite.
- (2) Iron
- (3) Magnetite (represented as Fe_2O_3)
- (4) Iron plus 1% boron
- (5) Fe_2O_3 plus 1.9% boron
- (6) Fe_2O_3 plus 10% boron

As the results in Table 1 indicate Fe_2O_3 with 10% boron produces a lower sodium activity than the graphite and

borated graphite shield. The present reference design indicates the use of graphite in the neutron shield. The results in Table 1 however indicate a definite advantage to the use of borated Fe_2O_3 . The Fe_2O_3 produces a secondary sodium activation a factor of 8 less than the graphite. As early indicated borated Fe_2O_3 is considerably less expensive than graphite and has fewer problems with respect to radiation damage and the requirement of an inert gas atmosphere. The neutron shield as presently described in the reference design contains approximately 2 million lbs. of graphite. Graphite has a cost of approximately 50¢/lb., whereas borated Fe_2O_3 would have a cost of 2 to 3¢/lb. This together with the elimination of a need for an inert gas system could result in a saving of approximately one million dollars for the shield.

As mentioned previously, Table 2 presents a comparison of various shield materials for use in the axial shield. In this case, it was assumed that the sodium pooled thickness was 30'. The results here are very similar to those obtained in Table 1 and again indicate the advantage of the borated Fe_2O_3 .

Likewise Table 3 presents a comparison of various shield materials for the axial shield when a 10' sodium pool is used. Again, the advantage of borated Fe_2O_3 with respect to iron is indicated.

Table 9 shows the effect of variations in the shield design in the reference case as described in BNWL drawing SK-3-12058 and Figure 1. The table shows the effect of varying the thickness of the iron shield and the thickness of the graphite shield. In both cases the effect on the neutron flux at the sodium vessel and the secondary sodium activation is indicated. These results are also plotted in Figure 3 and from

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these results the required shield thicknesses for the allowable radiation limits earlier mentioned were obtained. Table 10 shows the same information when borated graphite is used in the neutron shield instead of ordinary graphite.

Figure 1 shows the effect of graphite thickness on secondary sodium activity in the radial shield. In this case unborated graphite is assumed. The results show that in order to obtain an activity in the secondary sodium of no greater than .1 curies a 4' thick graphite shield is required.

Figure 2 shows the effect of the iron shield thickness on the secondary sodium activity using unborated graphite. This figure indicates that for the entire range of thicknesses between 1' and 4' the sodium activity is well below the specified value of .1 curies.

Figure 3 shows the effect of the iron shield thickness on the neutron flux at the sodium vessel. This figure shows that the required iron shield thickness to obtain a flux of 1.6×10^{12} n/cm² sec is 2' of iron.

Figure 4 shows the effect of graphite thickness on sodium activity using borated graphite in the radial shield. The results here indicate that a thickness of less than 2' of borated graphite would be required.

Figure 5 shows the effect of iron shield thickness on secondary sodium activity when using borated graphite in the radial shield. In this case, the results indicate that for all iron shield thicknesses between 1' and 4' the sodium activity is well below the required value of .1 curies.

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Figure 6 is a plot of the total neutron flux versus distance in the radial shield when using 6' of ordinary graphite.

Figure 7 is a plot of the total neutron flux versus distance in the radial shield when using 6' of borated Fe_2O_3 .

Figure 8 is a plot of the total neutron flux versus distance in the axial shield when using 3' of iron as the top neutron shield in the 30' sodium pool case.

5.0 RECOMMENDATIONS

(1) It is strongly recommended that attention be given to the use of alternate materials for the presently contemplated graphite in the radial shield. The reasons are:

- a. Graphite is a relatively expensive material particularly if borated graphite be used.
- b. Graphite must be contained in an inert atmosphere for the entire life of the reactor.
- c. Graphite undergoes radiation damage and particular attention must be paid to the swelling phenomena that might occur under irradiation.

As a possible alternate, one might consider the use of Fe_2O_3 (magnetite) with the addition of a borating agent such as colemanite. These materials are both quite inexpensive, the magnetite costs approximately 1¢ per lb., and the colemanite costs approximately 5¢ per lb, as compared to graphite which is in the order of 50¢ per lb. Furthermore, both of these materials (magnetite-colemanite) can exist in a normal air atmosphere so no special gas system is required. Finally, these materials are not subject to radiation damage.

As there are about two million pounds of graphite in the present radial shield, it is possible that savings of over one million dollars may be made by the use of an alternate material such as magnetite plus colemanite.

(2) It is recommended that the diffusion theory calculations be complemented by the use of more sophisticated methods such as Monte Carlo calculations for two purposes.

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- a. To reduce the degree of conservatism in the calculations and thereby taking advantage of savings and capital costs which can be affected by reduction in the amounts of shield materials.
- b. To increase the degree of confidence in the results by spot checking of a few points with more sophisticated methods.

(3) As mentioned earlier the calculations done to date have concerned themselves with the ideal shield. That is, the effects of ducts, crevices, etc., have not been taken into account. As these effects are important the next stage of design should address itself to this problem. A number of Monte Carlo codes exist for the treatment of the problems of ducts and gaps and should be used. It is significant to point out, however, that these codes in general are quite expensive to run. For this reason attention should be given to the use of the new AEC computer, the CDC 6600, inasmuch as this computer is about a factor of 10 - 20 faster than an IBM 7094 and probably a factor of 5 to 10 cheaper in terms of dollars per problem. Because of this very large decrease in the cost of calculation it will be reasonable to apply the present methods to a degree previously not possible.

TABLE 1

COMPARISON OF VARIOUS SHIELD MATERIALS

FOR RADIAL SHIELD

<u>Shield Material</u>	<u>Relative Secondary Sodium Activity</u>
Graphite and borated graphite	.524
Iron	452
Fe_2O_3	1070
Iron + 1% boron	30.9
Fe_2O_3 + 1.9% boron	8.62
Fe_2O_3 + 10% boron	.0627

TABLE 2

COMPARISON OF VARIOUS SHIELD MATERIALS
FOR AXIAL SHIELD WITH 30 FOOT SODIUM POOL

<u>Shield Material</u>	<u>Neutron Dose Rate Outside Shield - mr/hr</u>
Graphite and borated graphite	1.23 x 10 ⁻⁵
Iron	213 x 10 ⁻⁵
Fe ₂ O ₃	277 x 10 ⁻⁵
Iron + 1% boron	52.3 x 10 ⁻⁵
Fe ₂ O ₃ + 1.9% boron	9.86 x 10 ⁻⁵
Fe ₂ O ₃ + 10% boron	0.643 x 10 ⁻⁵

← Point at which dose is calculated

Shield material being varied

TABLE 3

COMPARISON OF VARIOUS SHIELD MATERIALS
FOR AXIAL SHIELD WITH 10 FOOT SODIUM POOL

<u>Shield Material</u>	<u>Neutron Dose Rate Outside Shield - r/hr</u>
Graphite and borated graphite	67.3
Iron	27100
Fe_2O_3	10100
Iron + 1% boron	7070
Fe_2O_3 + 1.9% boron	936
Fe_2O_3 + 10% boron	134

← Point at which dose is calculated
 Shield material being varied

TABLE 4
RADIATION LIMITS

<u>Location</u>	<u>Radiation Limits (General)</u>	<u>Radiation Limits (As used in calculations)</u>
Sodium vessel	Fast nvt of 10^{21} in 20 years	$0 - 1.6 \times 10^{12}$ ($E > .1$ mev)
Secondary Sodium HX	Specific activity - 2×10^{-4} μ /cc	Total Secondary Sodium activity - 0.1 curie
Operation flux above reactor	Total dose - .25 mr/hr	Neutron dose - .025 mr/hr

TABLE 5
REQUIRED SHIELD THICKNESSES

	<u>As-Calculated</u>	<u>Suggested Design Value</u>
Radial iron shield	2.8 feet	3.5 feet
Radial graphite shield	4 feet	5 feet
Top axial iron shield	< 3 feet	3 feet

TABLE 6

DESCRIPTION OF ZONED CORE CASE USED IN
MULTIGROUP CALCULATIONS - RADIAL CASE

Region	Name	Outer Radius cm	Composition	
			Material	Atom Density
1	Core zone-1	15.5	Pu	0.000903
			O	0.00196
			Fe	0.0265
			Na	0.0165
2	Core zone-2	25.6	Pu	0.000990
			O	0.00212
			Fe	0.0262
			Na	0.0165
3	Core zone-3	35.7	Pu	0.00113
			O	0.00244
			Fe	0.0257
			Na	0.0165
4	Core zone-4	45.9	Pu	0.00131
			O	0.00282
			Fe	0.0252
			Na	0.0165
5	Inconel Reflector	71.3	Ni	0.0920
6	Neutron Shield	86.5	B	0.0067
			C	0.0795
7	Neutron Shield	147.5	C	0.0837
8	Neutron Shield	162.7	B	0.0067
			C	0.0795
9	Secondary Sodium	172.7	Na	0.0254
10	Secondary Sodium	182.7	Na	0.0254

TABLE 7

DESCRIPTION OF ZONED CORE CASE USED IN
MULTIGROUP CALCULATIONS - AXIAL CASE

Region	Name	Outer Radius In cm	Composition	
			Material	Atom Density
1	Core zone-1	15.5	Pu O Fe Na	0.000903 0.00196 0.0265 0.0165
2	Core zone-2	25.6	Pu O Fe Na	0.000990 0.00212 0.0262 0.0165
3	Core zone-3	35.7	Pu O Fe Na	0.00113 0.00244 0.0257 0.0165
4	Core zone-4	45.9	Pu O Fe Na	0.00131 0.00282 0.0252 0.0165
5	Inconel Reflector	84.0	Ni Na	0.0460 0.0127
6	Inconel Reflector	389	Na	0.0254
7	Neutron Shield	404	B C	0.0067 0.0795
8	Neutron Shield	465	C	0.0837
9	Neutron Shield	481	B C	0.0067 0.0795
10	Receptor Region	491	Na	0.0254
11	Receptor Region	501	Na	0.0254

TABLE 8

DESCRIPTION OF SINGLE REGION, CORE CASE USED IN
MULTIGROUP CALCULATIONS - RADIAL CASE

<u>Region</u>	<u>Name</u>	<u>Outer Radius In cm</u>	<u>Composition</u>	
			<u>Material</u>	<u>Atom Density</u>
1	Core	40.0	Pu	0.00178
			O	0.00356
			SS	0.0362
			Na	0.0114
2	Inconel	70.4	Na	0.00284
			Ni	0.0669
			Fe	0.00518
3	Iron Shield	101	Na	0.00228
			Fe	0.0764
4	Sodium	275	Na	0.0255
5	Neutron Shield	458	Fe	0.00593
			C	0.0567
6	Secondary Sodium	478	Na	0.0255

TABLE 9

EFFECT OF VARIATIONS IN SHIELD DESIGN OF
REFERENCE CASE - NORMAL GRAPHITE

Fe	Shield Thickness - Feet Graphite	Flux at Sodium Vessel $n/cm^2 \text{ sec}$	Secondary Sodium Activation Curies
1	6	$2.69 \times 10^{+14}$	0.914×10^{-3}
4	6	$1.01 \times 10^{+12}$	0.639×10^{-4}
2	6	$4.54 \times 10^{+13}$	0.388×10^{-3}
2	11	$4.54 \times 10^{+13}$	0.298×10^{-9}
2	1	$4.54 \times 10^{+13}$	$0.330 \times 10^{+3}$

TABLE 10

EFFECT OF VARIATIONS IN SHIELD DESIGN OF
REFERENCE CASE - BORATED GRAPHITE

Shield Thickness - Feet Fe Borated Graphite		Flux at Sodium Vessel $n/cm^2 \text{ sec}$	Secondary Sodium Activation Curies
1	6	$2.69 \times 10^{+14}$	0.473×10^{-8}
4	6	$9.83 \times 10^{+11}$	0.478×10^{-9}
2	6	$4.54 \times 10^{+13}$	0.198×10^{-8}
2	11	$4.54 \times 10^{+13}$	0.267×10^{-15}
2	1	$4.54 \times 10^{+13}$	0.308

NUCLEAR TECHNOLOGY CORPORATION

APPENDIX

The Appendix contains the energy and lethargy group structures and the cross sections that were used in the AIM-6 code diffusion theory calculation. The elements described are Pu-239, nickel, iron, sodium, oxygen, carbon and boron.

TABLE 11
 AIM-6
 Energy and Lethargy
Group Structures

<u>Group</u>	<u>E</u>	<u>Eav</u>	<u>Lethargy</u>	<u>Δu</u>
1	10 Mev - 3 Mev	5.4 Mev	0 - 1.204	1.204
2	3 - 1.4	2.05	1.204 - 1.966	0.762
3	1.4 - 0.9	1.13	1.966 - 2.408	0.442
4	0.9 - 0.4	0.60	2.408 - 3.219	0.811
5	0.4 - 0.1	0.20	3.219 - 4.605	1.386
6	0.1 Mev-17 Kev	42.0 Kev	4.605 - 6.377	1.772
7	17 - 3.353	7.6	6.377 - 8.0	1.623
8	3.353 - 0.454	1.23	8.0 - 10.0	2.0
9	0.454 Kev -61.44 ev	0.166	10.0 - 12.0	2.0
10	61.44 - 22.6	37 ev	12.0 - 13.0	1.0
11	22.6 - 8.315	13.6	13.0 - 14.0	1.0
12	8.315 - 3.059	5.0	14.0 - 15.0	1.0
13	3.059 - 1.1256	1.84	15.0 - 16.0	1.0
14	1.1256 - 0.4141	0.68	16.0 - 17.0	1.0
15	0.4141 - 0.1523	0.25	17.0 - 18.0	1.0
16	0.1523 - 0.0561	0.093	18.0 - 19.0	1.0
17	0.0561 - 0.0252	0.037	19.0 - 19.8	0.8
18	0.0252	0.015	19.8 - -	-

TABLE 12.

AIM-6 EIGHTEEN GROUP CROSS SECTIONSPu-239

Gr	σ_a	$\nu\sigma_f$	σ_{tr}	$\sigma_{1 \rightarrow 1+1}$	$\sigma_{1 \rightarrow 1+2}$	$\sigma_{1 \rightarrow 1+3}$	$\sigma_{1 \rightarrow 1+4}$	$\sigma_{1 \rightarrow 1+5}$
1	2.07	6.612	4.25	.2	.27	.31	.31	.04
2	2.15	6.026	4.5	.18	.35	.35	.05	
3	2.05	5.472	4.8	.3	.3	.06		
4	1.81	4.981	5.7	.29	.05			
5	1.84	4.81	8.4	.05				
6	2.67	5.291	12.17	.05				
7	3.65	6.778	13.15	.05				
8	8.71	14.671	18.21	.05				
9	37.6	60.26	47.1	.05				
10	81.1	128.7	90.6	.05				
11	109.84	171.6	119.34	.05				
12	39.476	65.78	48.976	.05				
13	39.11	65.2	48.61	.05				
14	163.3	343.2	172.8	.05				
15	1901.89	3231.8	1911.39	.05				
16	722.6	1430.0	732.1	.05				
17	834.1	1716.0	843.6	.05				
18	1042.6	2145.0	1052.1					

TABLE 13

AIM-6 EIGHTEEN GROUP CROSS SECTIONS

Nickel

Gr	σ_a	$\sqrt{\sigma_f}$	$\sqrt{\sigma_{tr}}$	$\sigma_{1 \rightarrow 1+1}$	$\sigma_{1 \rightarrow 1+2}$	$\sigma_{1 \rightarrow 1+3}$	$\sigma_{1 \rightarrow 1+4}$	$\sigma_{1 \rightarrow 1+5}$
1	.00167		2.16	.851	.31	.354	.15	
2	.00372		2.56	.302	.28	.158		
3	.00718		2.95	.3				
4	.00691		3.35	.216				
5	.00951		5.32	.118				
6	.0170		11.5	.598				
7	.0195		24.0	.236				
8	.032		17.6	.162				
9	.051		16.7	.144				
10	.11		17.0	.338				
11	.18		17.1	.414				
12	.31		17.1	.379				
13	.52		17.1	.307				
14	.875		17.3	.255				
15	1.42		17.3	.097				
16	2.4		17.1	.081				
17	4.0		16.5	.1				
18	4.26		16.0					

TABLE 14

AIM-6 EIGHTEEN GROUP CROSS SECTIONS

Iron

Gr	σ_a	$\sqrt{\sigma_f}$	σ_{tr}	$\sigma_{1 \rightarrow 1+1}$	$\sigma_{1 \rightarrow 1+2}$	$\sigma_{1 \rightarrow 1+3}$	$\sigma_{1 \rightarrow 1+4}$	$\sigma_{1 \rightarrow 1+5}$
1	.005		1.713	1.21				
2	.005		2.322	.477	.322	.023		
3	.005		2.61	.335	.346	.03		
4	.005		2.764	.123				
5	.00708		2.816	.071				
6	.018		4.377	.089				
7	.012		5.939	.151				
8	.019		7.11	.3				
9	.03		11.23	.2				
10	.07		11.47	.2				
11	.11		11.51	.4				
12	.17		11.57	.4				
13	.3		11.7	.4				
14	.49		11.89	.4				
15	.8		12.2	.4				
16	1.34		12.74	.4				
17	2.13		13.53	.4				
18	2.14		12.94					

TABLE 15

AIM-6 EIGHTEEN GROUP CROSS SECTIONS

Sodium

Gr	σ_a	$\nu \sigma_f$	σ_{tr}	$\sigma_{1 \rightarrow 1+1}$	$\sigma_{1 \rightarrow 1+2}$	$\sigma_{1 \rightarrow 1+3}$	$\sigma_{1 \rightarrow 1+4}$	$\sigma_{1 \rightarrow 1+5}$
1	.03		1.607	.44				
2			2.26	.556				
3	.0002		2.9902	1.191				
4	.0005		4.0905	.7081		.036	.008	
5	.0009		3.6898	.2317				
6	.001		4.272	.207				
7	.001		6.089	.301				
8	.001		4.545	.234				
9	.011		3.01	.216				
10	.018		3.03	.236				
11	.032		3.069	.219				
12	.076		3.099	.219				
13	.057		3.099	.238				
14	.1		3.21	.294				
15	.094		3.35	.294				
16	.094		3.35	.294				
17	.19		3.35	.198				
18	.447		3.747					

TABLE 16

AIM-6 EIGHTEEN GROUP CROSS SECTIONS

Oxygen

Gr	σ_a	$\nu \sigma_f$	σ_{tr}	$\sigma_{j \rightarrow j+1}$	$\sigma_{j \rightarrow j+2}$	$\sigma_{j \rightarrow j+3}$	$\sigma_{j \rightarrow j+4}$	$\sigma_{j \rightarrow j+5}$
1	.04		1.33	.424				
2			1.18	.191				
3			3.23	.902				
4			3.63	.556				
5			3.71	.337				
6			3.26	.231				
7			3.55	.255				
8			3.64	.23				
9			3.64	.23				
10			3.64	.46				
11			3.64	.46				
12			3.64	.46				
13			3.64	.46				
14			3.64	.46				
15			3.64	.46				
16			3.64	.46				
17			3.64	.46				
18	.0002		3.6412					

TABLE 17

AIM-6 EIGHTEEN GROUP CROSS SECTIONS

Carbon

Gr	σ_a	$\gamma \sigma_f$	σ_{tr}	$\sigma_{1 \rightarrow 1+1}$	$\sigma_{1 \rightarrow 1+2}$	$\sigma_{1 \rightarrow 1+3}$	$\sigma_{1 \rightarrow 1+4}$	$\sigma_{1 \rightarrow 1+5}$
1			1.23	.515				
2	.000001		1.42	.314				
3			2.26	.856				
4			2.93	.604				
5			3.59	.433				
6			4.25	.401				
7			4.44	.43				
8			4.34	.37				
9			4.34	.37				
10			4.34	.73				
11			4.34	.73				
12			4.34	.73				
13			4.34	.73				
14			4.34	.73				
15			4.44	.75				
16			4.44	.75				
17	.001		4.441	.75				
18	.0027		4.5327					

TABLE 18

AIM-6 EIGHTEEN GROUP CROSS SECTIONS

Boron (Natural)

Gr	$\bar{\sigma}_a$	$\lambda \bar{\sigma}_f$	$\bar{\sigma}_{tr}$	$\bar{\sigma}_{j \rightarrow j+1}$	$\bar{\sigma}_{j \rightarrow j+2}$	$\bar{\sigma}_{j \rightarrow j+3}$	$\bar{\sigma}_{j \rightarrow j+4}$	$\bar{\sigma}_{j \rightarrow j+5}$
1	.04		1.52	.67				
2	.06		1.79	.42				
3	.04		2.14	.88				
4	.08		2.27	.5				
5	.27		3.16	.38				
6	.61		4.08	.36				
7	1.5		4.97	.37				
8	3.5		7.0	.3				
9	9.5		13.0	.3				
10	20.0		23.5	.6				
11	33.0		36.5	.6				
12	51.0		54.5	.6				
13	90.0		93.5	.6				
14	146.0		149.5	.6				
15	239.0		242.5	.6				
16	395.0		398.5	.6				
17	634.2		637.7	.6				
18	638.0		641.5					

K&E 20 X 20 TO THE INCH 46 1242
7 X 10 INCHES KEUFFEL & ESSER CO
MADE IN U.S.A.

Figure 1

Effect of graphite thickness on sodium activity
in radial shield - (unborated graphite)
Reactor power = 500 MW

Curries of
Secondary Sodium Activity

core 1 ft 2 ft
Inconel Fo t secondary sodium

Design Limit

Graphite Thickness, t

feet

Figure 2

Effect of iron shield thickness on Sodium activity in the radial shield - (unborated graphite)

Reactor power = 500 MW

Secondary Sodium Activity

Core $\square$ 1 ft $\leftarrow t \rightarrow$ 6 ft

Inconel

Fe Carbon

Secondary Sodium

-3

10

9

8

7

6

5

4

3

2

1

0

-4

10

9

8

7

6

5

4

3

2

1

0

-5

0

1

2

3

4

feet

Iron thickness, t

-39-

Curles
Secondary Sodium Activity

Figure 4

Effect of graphite thickness on secondary sodium activity in the radial shield - (borated graphite)

Reactor power = 500 MW

Figure 6

Total neutron flux vs distance in the radial shield -

6' graphite neutron shield

Reactor power = 500 MW

TOTAL FLUX
 $n/cm^2 \cdot sec$

10^{+16}
 10^{+15}
 10^{+14}
 10^{+13}
 10^{+12}
 10^{+11}
 10^{+10}
 10^{+9}
 10^{+8}
 10^{+7}
 10^{+6}
 10^{+5}
 10^{+4}
 10^{+3}
 10^{+2}
 10^{+1}

1 PuO2

2 Inconel

3 Fe

4 Na

5 C

6 Na

0 100 200 300 400 500 CM

Radial Distance

Figure 7

K&E 20 X 20 TO THE INCH 46 1242
7 X 10 INCHES
KELUFFEL & ESSER CO.
MADE IN U.S.A.

Figure 8

Total neutron flux vs distance in the axial shield

3' steel + 30' sodium pool

Reactor power = 500 MW

TOTAL FLUX
 $n/cm^2 \cdot sec$

10^{+12}

10^{+10}

10^{+8}

10^{+6}

10^{+4}

10^{+2}

10^0

10^{-2}

10^{-4}

1 - 4 PuO_2
Fe
Na
5 Ni
Na
6 Na
7 - 9 Fe
10 - 11 Na

Radial Distance

20 X 20 TO THE INCH 46 1242
7 X 10 INCHES
MADE IN U.S.A.
KEFEL & ESSER CO