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APAE MEMO 124

INTEGRAL FLUX SUPPRESSOR
FOR APPR-1, CORE II

Contract No. AT(30-3)-278

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ABSTRACT

An internal flux suppressor will be used for the APPR-1 Core II instead of the external suppressors that are employed in Core I. The reference design of the integral flux suppressor for Core II is as follows:

Material - Eu_2O_3 in stainless steel matrix
Composition - 23.2 w/o Eu_2O_3
Dimensions - 0.02" x 2.281" x 0.875"

The survey on the effectiveness of this reference design has shown that the peaking of the flux in the control rod fuel element tip has been eliminated throughout the lifetime and therefore eliminating the possibility of local boiling in this region.

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I. Introduction

During the APPR-1 Zero Power Experiments conducted at the Alco Critical Facility in Schenectady, New York (1), it was discovered that the flux peaks sharply in the water region just above the control rod fuel element. Axial foil measurements taken within the centerline control rod fuel element showed this high thermal flux peak. The tail of the peak extended down into the control rod fuel element thereby causing an increased fission rate and a possibility of local boiling of the water at the tips. The investigations and results are reported in (1), (2), and (3).

Due to the flux peaking, the necessity of suppressing the flux was obvious. For Core II, it was decided to use an integral suppressor containing 23.2 w/o Eu_2O_3 in a stainless steel matrix (4). See Figure 1 for layout of control rod fuel element and absorber.

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Figure 1
A. Control Rod Fuel Element (Not to Scale)

B. Division of Control Rod Fuel Element
into Regions For Valprod Calculation (Not to Scale)

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II. Calculation Model

A one-dimensional two-group multiregion model was set up to get the local flux and power distribution along the vertical centerline of the core. The following set of equations are used to solve for the distributions in the control rod fuel element and suppressor.

$$-D_{fj} \nabla^2 \phi_f(Z) + \sum r_j \phi_f(Z) = S(Z) \quad (1)$$

$$-D_{sj} \nabla^2 \phi_s(Z) + \sum a_s \phi_s(Z) = p_j \sum r_j \phi_f(Z) \quad (2)$$

where,

D_f = Fast diffusion coefficient, cm

D_s = Slow diffusion coefficient, cm

p = Resonance escape probability

ϕ_f = Fast flux, neutrons/cm²-sec

ϕ_s = Slow flux, neutrons/cm²-sec

$\sum_r$ = Fast group removal cross-section, cm⁻¹

$\sum_{as}$ = Slow group absorption cross-section, cm⁻¹

S = Source term

Subscript j = Denotes region (Table I gives the materials of each region while Figure 1 gives the division of the center-line element into regions.)

As the control rod absorber section is a hollow box filled with

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Table I

Region Materials Used in Calculation Model

Region	Atom Densities* (atom/cc) x 10 ⁻²⁴					
	H	O	SS	U ²³⁵	B	Eu
1	.056001	.028001	-----	-----	-----	-----
2	.056001	.028001	-----	-----	-----	-----
3	.056001	.028001	-----	-----	-----	-----
4	.045705	.022852	.015646	-----	-----	-----
5	.045705	.023987	.013246	-----	-----	.000756
6	.045705	.023987	.013246	-----	-----	.000756
7	.045705	.023987	.013246	-----	-----	.000756
8	.045705	.023964	.012421	.000498	.000107	-----
9	.045705	.023964	.012421	.000498	.000107	-----
10	.045705	.023964	.012421	.000498	.000107	-----
11	.045705	.023964	.012421	.000498	.000107	-----
12	.045705	.022852	.015646	-----	-----	-----
13	.044801	.022400	.017022	-----	-----	-----

* The atom densities are based on one fuel plate and one water gap without the side edges and the side plates.

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water, the primary effect is that the water slows down the fast neutrons which penetrates from outside the absorber section. In order to obtain a more realistic picture of the source distribution in the rod, a fast neutron source distribution was obtained from a Windowshade calculation(5) for the reactor as a whole. Therefore, $S(Z)$ in equation 1 becomes,

$$S(Z) = \frac{1}{K} \left[\nu_f \sum_{fj}^f \phi_f(Z) + \nu_s \sum_{sj}^f \phi_s(Z) \right] \quad (3)$$

where,

K = conversion factor, $= 3.24 \times 10^{10}$ fission/watt-sec

$\sum_{fj}^f$ = Fast group fission cross-section, cm^{-1}

$\sum_{sj}^f$ = Slow group fission cross-section, cm^{-1}

ν_f = Fast neutrons produced per fast fission

ν_s = Fast neutrons produced per thermal fission

The solution of equations 1 and 2 are solved by the Valprod code(6). This is not a criticality calculation but a computation of the local flux and power distribution using the axial source, $S(Z)$, from a Windowshade calculation.

From the way the model is set up, the suppressor calculation is dependent on the reactor as a whole only through the source term, $S(Z)$. The source is of course dependent upon the properties of the core and upon

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the bank position. However, the source distribution in the suppressor region does not change much in shape but essentially is only displaced from one position to another as the bank is moved. Therefore, the effectiveness of the suppressor is a local phenomena and not a strong function of the reactor as a whole. Even though the effect of the suppressor was not calculated for the deepest insertion at operating temperature, (approximately midlife), it can be assumed that it would not be much different from the calculated case.

This model has two limitations but previous calculations have not shown them to be very serious. The first is that diffusion theory is not strictly applicable at the interfaces of highly dissimilar regions; and second that for given regions, region properties were homogenized.

In as much as self shielding factors are not accurately known and fast cross-sections for all europium isotopes are not known, it was decided to run two parallel cases as limits. Case I used available fast cross-sections for Eu-151 and Eu-153 (7) and assumed cross-sections for Eu-152, Eu-154 and Eu-155 based on the ratio of fast to thermal cross-section for Eu-153 (8) and no self shielding factors. Case II used available fast cross-sections for Eu-151 and Eu-153 and self shielding factors (see Appendix I). Essentially this means that we have cases of maximum absorption and minimum absorption for the limits. The actual case will be somewhere in this range.

Both cases were calculated with and without an integral suppressor at start of life and at 440⁰ F. with equilibrium Xe. Regions 5, 6, and 7 were also considered to be fuel for the calculation without the suppressor. Table II gives the group constants for each region with flux suppressors (see Appendix II).

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Table II
Group Constants (at Startup) for Calculation Model

Case I

<u>Region</u>	<u>D_s</u>	<u>D_f</u>	<u>Σ_s</u>	<u>τ</u>	<u>p</u>	<u>K_f</u>	<u>K_{th}</u>
1, 2, 3	0.26500	1.8937	0.01105	47.5034	0.98848	0	0
4, 12	0.29423	1.6331	0.03660	38.9325	0.77202	0	0
5, 6, 7	0.13499	1.5120	1.38085	6.6938	0.13848	0	0
8, 9, 10, 11	0.24494	1.7027	0.27121	29.7900	0.66993	1.21272	1.47075
13	0.29711	1.6206	0.03885	38.4667	0.75343	0	0

Case II

<u>Region</u>	<u>D_s</u>	<u>D_f</u>	<u>Σ_s</u>	<u>τ</u>	<u>p</u>	<u>K_f</u>	<u>K_{th}</u>
1, 2, 3	0.26500	1.8937	0.01105	47.5034	0.98848	0	0
4, 12	0.29423	1.6331	0.03660	38.9325	0.77202	0	0
5, 6, 7	0.18628	1.6144	1.70090	19.9185	0.40127	0	0
8, 9, 10, 11	0.25143	1.7027	0.23794	29.7900	0.66993	1.21272	1.45343
13	0.29711	1.6206	0.03885	38.4667	0.75343	0	0

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III. Suppressor Burnout

The previous calculation gives the flux and power distribution in the suppressor regions at startup and at 440° F, and with equilibrium Xenon. Assuming that the suppressor sufficiently depresses the flux initially, the Eu_2O_3 will burn out over the reactor lifetime depending on the flux to which it is subjected and may not give the necessary suppression after startup.

A one step burnout calculation was performed. It was assumed that the source term was constant over the lifetime of the core. Since the flux distribution in the flux suppressor is a local effect, this is a reasonable assumption.

The first step is the calculation of the burnup in each region assuming that only europium burns up in regions 5 through 7 and that the burnup in regions 8 through 11 is based on U-235 with the burnup of B-10 dependent on the U-235 burnup. In order to determine the burnups, it is necessary to know the absolute fast and thermal fluxes in all regions.

The relative fluxes from the output of the axial Valprod calculation can be converted to absolute values by using the power density as a coupling quantity. A region from the Windowshade calculation identical to a region from the Valprod calculation was chosen where no perturbation was felt due to the suppressor. This point is inches from the tip of the fuel meat. The derivation is as follows:

From the Windowshade, the fraction of power produced in that

region is obtained. Assuming that 10 MW is the average total power produced, $10 X_i$ is the average power produced in that region, when X_i is the fraction of power produced in region i. Equation 4 gives the relationship used to calculate the absolute thermal flux.

$$\frac{\overline{P}}{V} = \phi_t \left[K_1 + K_2 \frac{\phi_f}{\phi_t} \right], \quad (4)$$

where,

$$\frac{\overline{P}}{V} = \text{average power density in region i.}$$

$$\phi_t = \text{absolute thermal flux at point where the relative power from the Valprod calculation has the average value in region i.}$$

$$\frac{\phi_f}{\phi_t} = \text{ratio of relative fast flux to relative thermal flux at point where the relative power from the Valprod calculation has the average value in region i.}$$

$$K_1 = \frac{\sum_f^t}{K'}$$

$$K_2 = \frac{\sum_f^f}{K'}$$

$$\sum_f^t = \text{macroscopic thermal fission cross-section in region i.}$$

$$\sum_f^f = \text{macroscopic fast fission cross-section in region i.}$$

$$K' = 3.24 \times 10^{10} \text{ fissions/watts-sec.}$$

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Since the Windowshade and axial Valprod gives the axial flux distribution independent of the radial distribution, the absolute flux obtained from Equation 4 must be multiplied by the ratio of maximum radial thermal flux to the average radial flux in the core. Once the absolute flux is normalized to the relative flux distribution obtained from the Valprod calculation, the absolute average thermal and fast fluxes are determined for regions 5 through 11. The values of the fluxes calculated are given in Table III.

Table III
Regional Absolute Average Fluxes in Calculation Model

Region	<u>Case I</u>		<u>Case II</u>	
	Absolute ϕ_t	ϕ_f / ϕ_t	Absolute ϕ_t	ϕ_f / ϕ_t
5	1.5982×10^{13}	5.64354	3.1724×10^{13}	4.00328
6	0.3453×10^{13}	26.22860	1.1788×10^{13}	10.96774
7	0.4304×10^{13}	22.37285	1.0837×10^{13}	12.28070
8	1.8993×10^{13}	7.41341	2.3917×10^{13}	6.51638
9	2.2883×10^{13}	7.12605	2.6427×10^{13}	6.25007
10	1.6045×10^{13}	7.24321	1.8742×10^{13}	6.18523
11	0.1979×10^{13}	8.21775	0.2667×10^{13}	6.18138

Now that the absolute fluxes are known, the burnups in all regions can be calculated. The calculation of the burnups in regions 5 through 7 is a trial and error process. The computation uses the analytic curves of the ratio of the

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cross-section of europium at any burnup to the original cross-section as a function of the burnup, number of absorptions per original atom of europium

(8). Equation 5 was used to obtain the burnups in the suppressor.

$$B_i = \frac{(A)}{(No)_i} = \left[\sigma_{to} \left(\frac{\bar{\Sigma}}{\bar{\Sigma}_o} \right)_t + \gamma \sigma_{fo} \left(\frac{\bar{\Sigma}}{\bar{\Sigma}_o} \right)_f \right] \phi_t \Delta t \quad (5)$$

where

B_i = burnup in region i

σ_{to} = original thermal microscopic absorption cross-section,
 cm^2/atom

σ_{fo} = original fast microscopic absorption cross-section, cm^2/atom

γ = ϕ_f / ϕ_t

ϕ_t = average absolute thermal flux in region i , neutrons/ $\text{cm}^2\text{-sec}$

Δt = time of reactor operation, sec

$\left(\frac{\bar{\Sigma}}{\bar{\Sigma}_o} \right)_t$ = average thermal cross-section ration in the interval ΔB .

$\left(\frac{\bar{\Sigma}}{\bar{\Sigma}_o} \right)_f$ = average fast cross-section in the interval ΔB

A burnup, B , was assumed and the average cross-section ratios were taken from the graphs in APAE Note 97 (8). By using Equation 2, the burnup was calculated. When the calculated burnup was equal to the assumed burnup, the average burnup in region i is then known.

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In the case of the burnup in the fuel regions, 8 through 11, an analytic expression is known between the cross-section ratio and the burnup.

For U-235,

$$\frac{\overline{(N)}}{\overline{(N_0)}} = \left(1 - \frac{B}{2}\right) \quad (6)$$

where

$$\frac{\overline{(N)}}{\overline{(N_0)}} = \text{average number density ratio in the interval } \Delta B$$

or

$$\left(\frac{\overline{\Sigma}}{\overline{\Sigma_0}}\right)_t = \left(\frac{\overline{\Sigma}}{\overline{\Sigma_0}}\right)_{\phi_t} = \left(1 - \frac{B}{2}\right) \quad (7)$$

substituting equation (7) into equation (5), one obtains

$$B = (\sigma_{to} + \gamma \sigma_{fo}) \phi_t \Delta t \left(1 - \frac{B}{2}\right) \quad (8)$$

Solving explicitly for B

$$B = \frac{2 (\sigma_{to} + \gamma \sigma_{fo}) \phi_t \Delta t}{2 + (\sigma_{to} + \gamma \sigma_{fo})} \quad (9)$$

The burnups, B, in each fuel region can be calculated using Equation 9. The ratio of the cross-section at any burnup to the original cross-section is then determined by using Equation 10.

$$\left(\frac{\overline{\Sigma}}{\overline{\Sigma_0}}\right)_{U-235} = (1 - B) \quad (10)$$

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The same ratios for the suppressor regions $\left(\frac{\Sigma}{\Sigma_0}\right)_t^{eu}$ and $\left(\frac{\Sigma}{\Sigma_0}\right)_f^{eu}$,

were obtained from APAE Note 97 (8) for the burnups that were calculated.

Table IV lists the burnups in each region and the cross-section ratios in each region for Cases I and II,

Table IV

Burnup And Cross-Section Ratios in Each Region

Region	B	<u>Case I</u>		B	<u>Case II</u>	
		$\left(\frac{\Sigma}{\Sigma_0}\right)_t$	$\left(\frac{\Sigma}{\Sigma_0}\right)_f$		$\left(\frac{\Sigma}{\Sigma_0}\right)_t$	$\left(\frac{\Sigma}{\Sigma_0}\right)_f$
5	1.300	0.188	0.280	0.956	0.298	0.103
6	0.915	0.278	0.401	0.569	0.510	9.220
7	0.976	0.250	0.366	0.552	0.520	0.235
8	0.382	0.618	0.618	0.411	0.589	0.589
9	0.440	0.560	0.560	0.442	0.559	0.559
10	0.331	0.669	0.669	0.334	0.666	0.666
11	0.050	0.951	0.951	0.055	0.945	0.945

In the first regions, the burnup of B-10, the equivalent amount of B-10 due to fission products, and the amount of equilibrium Xenon was coupled to the

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U-235 burnup. Since the burnups and the cross-section ratios are now known in each region, the two group constants can be calculated for all regions at an average core burnup of 30 percent. The group constants at 30 percent burnup are given in Table V. Regions 1, 2, 3, 4, 12 and 13 are not included as they are assumed to suffer no burnup.

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Table V

Group Constants at 30% Burnup for Calculation Model

Region	Case I						
	D_s	D_f	Σ_s	τ	ρ	K_f	K_{th}
5	0.24341	1.6154	0.28589	17.4866	0.35045	0	0
6	0.22352	1.5957	0.40725	13.8957	0.28200	0	0
7	0.22935	1.6016	0.36944	14.7818	0.29888	0	0
8	0.24494	1.7027	0.15850	36.6898	0.70813	1.03480	1.55425
9	0.24494	1.7027	0.14625	37.3268	0.71587	0.98057	1.52733
10	0.24494	1.7027	0.16970	36.0963	0.70117	1.07647	1.57763
11	0.24994	1.7027	0.25268	31.1399	0.67602	1.21687	1.50050

Region	Case II						
	D_s	D_f	Σ_s	τ	ρ	K_f	K_{th}
5	0.22772	1.6579	0.34004	32.3815	0.71173	0	0
6	0.20549	1.6500	0.51636	27.6767	0.60885	0	0
7	0.20335	1.6465	0.52381	27.1731	0.59905	0	0
8	0.25143	1.7027	0.14882	36.9847	0.71196	1.00798	1.46824
9	0.25143	1.7027	0.14381	37.3072	0.71626	0.97873	1.45615
10	0.25143	1.7027	0.16323	36.1308	0.70146	1.07429	1.49709
11	0.25143	1.7027	0.22509	31.3232	0.67398	1.20961	1.45637

With the use of the constants in Table V, the flux and power distributions in the control rod fuel element and the suppressor were calculated at end of life.

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IV. Results

A. Relative Distributions

The results from the Valprod calculations are plotted in the following figures. Figure 2 shows the relative axial thermal flux at start-up with and without suppressors and at 30 percent core burnup with suppressor for Case I. Figures 3 and 4 give the relative axial fast flux and relative axial power respectively under the same conditions for Case I. Figures 5, 6 and 7 show the relative thermal flux, relative fast flux and the relative power respectively for Case II. All curves labeled 1 are at startup with no suppressor, while curves labeled 2 are at startup with suppressors. As can be seen from these curves, all distributions are lowered by the use of suppressors. The curves labeled 3 are at 30 percent core burnup with suppressors. The fluxes and power are greater than at startup but they all show suppression due to the Eu_2O_3 . From these end of life graphs, it can be seen that the suppressors will adequately keep the flux from peaking and therefore eliminate the possibility of local boiling as the maximum values occur away from the control rod fuel element tips.

B. Absolute Power Density Distribution

To be able to determine whether local boiling will occur, an absolute power density distribution is needed along the axial centerline of the control rod fuel element and absorber box. In order to get the absolute power densities, the axial power distribution from the Windowshade calculation and a

FIG.2 RELATIVE THERMAL FLUX VS. AXIAL POSITION
WITH INTEGRAL Eu_2O_3 -SS FLUX SUPPRESSORS
(ALONG ϕ RODS)
CASE I

FIG.3 RELATIVE FAST FLUX VS. AXIAL POSITION
WITH INTEGRAL Eu_2O_3 -SS FLUX SUPPRESSORS
(ALONG $\downarrow$ RODS)
CASE I

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FIG. 4 RELATIVE POWER VS. AXIAL POSITION
WITH INTEGRAL EU_2O_3 FLUX SUPPRESSORS
(ALONG 4 RODS)
CASE I

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270 024

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FIG. 5 RELATIVE THERMAL FLUX VS. AXIAL POSITION
WITH INTEGRAL Eu_2O_3 SS FLUX SUPPRESSORS
(ALONG C RODS)
CASE II

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RELATIVE THERMAL FLUX

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FIG. 6 RELATIVE FAST FLUX VS. AXIAL POSITION
WITH INTEGRAL Eu_2O_3 -SS FLUX SUPPRESSORS
(ALONG 1 RODS)
CASE II

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FIG.7 RELATIVE POWER VS. AXIAL POSITION
WITH INTEGRAL EuO_2 -SS FLUX SUPPRESSOR
(ALONG $\downarrow$ RODS)
CASE II

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power distribution from a radial Valprod calculation must be normalized.

The absolute power density distribution along the axial centerline of the core can now be determined for fixed fuel elements. At the centerline, $r = 0$. Therefore,

$$P(0, z) = UQ P(r)_{r=0} \quad (11)$$

Table VI gives the appropriate values for Equation 11 at startup.

Table VI

Normalization Factors For Absolute Power Density

<u>Quantity</u>	
Total power , P	10^7 watts
Reactor volume, V	$1.396 \times 10^5 \text{ cm}^3$
U	$71.6302 \text{ watts/cm}^3$
Q	48
$P(r)_{r=0}$	1.489154
$UQ P(r)_{r=0}$	5.1201×10^3

One curve in Figure 8 gives the absolute power density distribution along the axial centerline for fixed fuel elements at startup, 440° F , and with equilibrium Xenon. The other curves in Figure 8 are for Case I and Case II with a control rod fuel element replacing the fixed fuel element with and without flux

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suppressors.

At startup, the use of the suppressor brings the power density below that of the distribution for fixed fuel elements. From this figure, it is obvious that there is a need for flux suppressors. It also seems that the specified suppressor will keep the flux at the fuel element tips below the maximum in the control rod fuel element but a detailed thermal analysis would be needed to determine if there is local boiling.

At the time that the most adverse power distribution occurs, presumably near midlife, axial and radial power distributions can be determined and normalized to get the absolute axial power density at that time. From the results shown in Figures 4 and 7, the local distribution can be superimposed upon the gross axial distribution by normalizing the local curve to the gross curve at a point where no perturbation will be felt due to the suppressor.

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APPENDIX I - SELF SHIELDING FACTORS

A. Thermal Factors (9)

Let F_0 = number of absorption/incident neutron

$\bar{\Sigma}_a \phi_0 V$ = number of absorption

$(S \phi_0)/2$ = incident number of neutrons

$$\therefore F_0 = \frac{\bar{\Sigma}_a \phi_0 V}{(S \phi_0)/2} = 2 \bar{\Sigma}_a t$$

$$\text{or } \bar{\Sigma}_a = F_0/2t$$

The thermal self-shielding factor may be defined as follows:

$$f = \frac{\bar{\Sigma}_a}{\Sigma_a} = \frac{F_0}{2 \Sigma_a t}$$

where,

V = volume of slab-cm³

S = surface of slab-cm²

t = thickness of slab-cm

$t_{1/2}$ = half thickness of slab-cm

$\bar{\Sigma}_a$ = shielded or apparent macroscopic absorption cross-section cm⁻¹

Σ_a = unshielded or calculated macroscopic absorption cross-section-cm⁻¹

In WAPD-149, (9), F_0 is given as a function of $\Sigma_t t_{1/2}$ for different ratios of scattering cross-sections to total cross-section.

B. Fast Factors (10)

Assuming (E) is a constant, the definition of the self-shielding factor is,

$$f = \frac{\int_{E_1}^{E_2} \phi(E) \left[\frac{1}{2} - E_3(\Sigma_a t) \right] \frac{dE}{E}}{\int_{E_1}^{E_2} \phi(E) \Sigma_a t \frac{dE}{E}} = \frac{\int_{u_1}^{u_2} \left[\frac{1}{2} - E_3(\Sigma_a t) \right] du}{\int_{u_1}^{u_2} \Sigma_a t du}$$

$$\Sigma_a t = \frac{N \sigma_0 \Gamma_\gamma t}{\Gamma} \left[\frac{1}{1 + \left(\frac{E - E_0}{\Gamma/2} \right)^2} \right]$$

where,

σ_0 = peak total cross-section of resonance - cm^2/atom

Γ_γ = partial level width for neutron capture - ev

Γ = total level width - ev

N = number density

t = thickness of slab-cm

E_0 = energy at which resonance occurs

$E_3(\Sigma_a t)$ = third exponential integral of $\Sigma_a t$

The equation for the self-shielding factor, f , was solved by numerical integration.

Even though different methods were used to calculate the thermal and fast factors, they both can be used for either factor. They will give

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approximately the same results. However, it was done in this manner to give the author familiarity with both methods.

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APPENDIX II - GROUP CONSTANTS

A. Thermal Constants

$$\bar{\Sigma}_{at}^j = \sum_{i=1}^n \Sigma_{at}^{ij} = \sum_{i=1}^n N^{ij} \sigma_{at}^i$$

$$D_t^j = \frac{1}{3 (\bar{\Sigma}_{at}^j)_t}$$

$$(\bar{\Sigma}_{at}^j)_t = \sum_{i=1}^n (\Sigma_{at}^{ij})_t = \sum_{i=1}^n N^{ij} (\sigma_{at}^i)_t$$

$$K_t^j = \frac{\nu_t (\bar{\Sigma}_{ft}^j)_{U-235}}{\bar{\Sigma}_{at}^j} = \frac{\nu_t N_{U-235}^j (\sigma_{ft}^j)_{U-235}}{\bar{\Sigma}_{at}^j}$$

where,

superscript i denotes i^{th} material

superscript j denotes j^{th} region

B. Fast Constants

$$(\bar{\Sigma}_a^j)_{\text{fast}} = \sum_{i=1}^n (\Sigma_a^{ij})_{\text{fast}} = \sum_{i=1}^n N^{ij} (\bar{\sigma}_a^i)_{\text{fast}}$$

where,

$$(\bar{\sigma}_a^i)_{\text{fast}} = \frac{\int_0^{u_t} \sigma_a^i(u) \phi(u) du}{\int_0^{u_t} \phi(u) du}$$

$$D_{Fast}^j = \frac{\int_0^{u_t} \frac{\phi(u) du}{3 \sum_n^j(u)}}{\int_0^{u_t} \phi(u) du}$$

$$\sum_n^j(u) = \sum_{l=1}^n \sum_n^{ilj} = \sum_{i=1}^n N^{ij} \sigma_n^{il}$$

$$\sum_n^j = \sum_{l=1}^n \sum_n^{ilj}$$

$$\gamma^j = \frac{D_{Fast}^j}{\sum_n^j}$$

$$\rho^j = 1 - \frac{(\sum_a^j)_{Fast}}{\sum_n^j}$$

$$K_{Fast}^j = \frac{\nu_{Fast} (\sum_p^j)_{Fast}^{U-235}}{(\sum_a^j)_{Fast}} = \frac{\nu_{Fast} N_{U-235}^j (\sigma_F^j)_{Fast}^{U-235}}{(\sum_a^j)_{Fast}}$$

where,

superscript l denotes l^{th} lethargy group

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