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PRELIMINARY DESIGN STUDY OF
NEUTRON-SPECTRUM CONVERTER
FOR USE IN THE JUGGERNAUT REACTOR'S
EAST THERMAL COLUMN

by

Dale A. Herbst and James H. Talboy

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PRELIMINARY DESIGN STUDY OF
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ABSTRACT

A design study has been made for a neutron-spectrum converter for Juggernaut's east thermal column. Calculations were performed to determine: (1) the U^{235} and B^{10} burnup rates, (2) the neutron-flux distribution, and (3) the heat generation and resulting temperature profiles in the converter assembly. Results of calculations indicate that a practical design has been established.

I. INTRODUCTION

Calculations and measurements have been performed to determine the feasibility of designing a neutron-spectrum converter satisfying the following specifications:

1. A fast-neutron flux greater than 1×10^{11} n/cm²-sec must be available for sample irradiation.
2. A 2-in. diam, sample-irradiation cavity is needed.
3. The maximum permissible temperature in the sample cavity is 60°C.
4. A minimum flux gradient, both laterally and longitudinally, is desired in the sample cavity.
5. The thermal-neutron flux in the sample cavity must be negligible in comparison with the fast-neutron flux.

The object of this report is to relate the methods and results of calculations performed for a preliminary design of a neutron-spectrum converter meeting these specifications.

II. CONVERTER DESIGN

The preliminary converter design used for calculations (Fig. 1) consists of a closed cylindrical U^{235} -aluminum alloy converter section,

bounded externally by a concentric H_2O coolant channel and internally by a boron thermal-neutron shield. The internal surface of the boron shield is cooled by an H_2O coolant channel. The converter is 12 in. long and has an outside diameter of 4 in. The entire assembly is designed to be positioned horizontally in the Juggernaut reactor's east thermal column.

Fig. 1. Preliminary Converter Design

III. NEUTRON-FLUX CALCULATIONS

Before any calculations were attempted, a flux map was made of the east thermal column in the Juggernaut reactor.¹ The results of these flux measurements are shown graphically in Figs. 2 and 3. Previously, the thermal-neutron flux in Juggernaut's east thermal column had been calculated.² The results of the calculations are shown graphically in Fig. 4. Comparison of Figs. 3 and 4 indicates agreement between calculated and measured flux values.

Knowing the neutron flux in the east thermal column, a two-group diffusion-theory calculation was performed to determine the fast and thermal

Fig. 2. Measured Neutron Flux for Juggernaut East Thermal Column with Graphite Stringer in Place

Fig. 3. Measured Neutron Flux for Juggernaut East Thermal Column

neutron-flux distributions when different thicknesses of U^{235} were inserted. The uranium section of the converter was treated as being a slab of thickness Nw , of infinite extent in the y and z directions, and having a vacuum for its boundary at $x = Nw$. Thermal- and fast-neutron fluxes of $\phi_2(0)$ and $\phi_1(0)$ respectively, impinge on the uranium at $x = 0$. A diagram of this model is shown in Fig. 5.

Fig. 4. Calculated Thermal-neutron Flux for Juggernaut-East Thermal Column (Taken from Ref. 2)

Fig. 5. Model Used for Two-group Analysis of Flux Distribution

The following equations are applicable:

$$-D_1 \frac{d^2 \phi_1}{dx^2} + (\Sigma_{A1} + \Sigma_{R1}) \phi_1 = \sum_{i=1,2} (v_i \Sigma_{Fi} \phi_i), \quad (3.1)$$

and

$$-D_2 \frac{d^2 \phi_2}{dx^2} + \Sigma_{A2} \phi_2 = \Sigma_{R1} \phi_1, \quad (3.2)$$

where

D_2 = diffusion coefficient for thermal-neutron group, cm^{-1} ;

D_1 = diffusion coefficient for fast-neutron group, cm^{-1} ;

ϕ_2 = thermal-neutron flux, neutrons/ $\text{cm}^2\text{-sec}$;

ϕ_1 = fast-neutron flux, neutrons/ $\text{cm}^2\text{-sec}$;

v_i = fission-neutron yield, neutrons/fission;

Σ_{Fi} = fission cross section, cm^{-1} ;

Σ_{Ai} = absorption cross section, cm^{-1} ;

and

Σ_{R1} = fast-neutron-removal cross section, cm^{-1} .

Solution of Eqs. (3.1) and (3.2) using the conditions shown in Fig. 5 results in the following expressions for the neutron-flux distribution for different thicknesses of U^{235} :

$$\phi_2(x) = \phi_2(0) \frac{\cosh [\gamma(Nw - x)]}{\cosh (\gamma Nw)}, \quad (3.3)$$

and

$$\begin{aligned} \phi_1(x) = & \left[\phi_1(0) - \frac{\xi^2 \cosh (\gamma Nw)}{\alpha^2 - \gamma^2} \right] \frac{\cosh [\alpha(Nw - x)]}{\cosh (\alpha Nw)} \\ & + \frac{\xi^2}{\alpha^2 - \gamma^2} \cosh [\gamma(Nw - x)], \end{aligned} \quad (3.4)$$

where

$\phi_2(0)$ = thermal-neutron flux at $x = 0$, neutrons/ cm^2 -sec;

$\phi_1(0)$ = fast-neutron flux at $x = 0$, neutrons/ cm^2 -sec;

w = equivalent thickness of Juggernaut type fuel plate, cm;

$N = 1, 2, 3, \dots$;

$$\gamma^2 = \frac{\Sigma_{A2}}{D_2};$$

$$\alpha^2 = \frac{\Sigma_{A1} - v_1 \Sigma_{F1}}{D_1};$$

and

$$\xi^2 = \frac{v_2 \Sigma_{F2} \phi_2(0)}{D_1 \cosh (\gamma Nw)}.$$

Equations (3.3) and (3.4) are derived in Appendix A. The normalized results of Eqs. (3.3) and (3.4) for several values of N , are presented graphically in Fig. 6. The normalizing factors used are the fast- and thermal-neutron fluxes at $x = 0$. Figure 6 indicates that the use of a converter section of thickness $2w$ will result in an increase in fast-neutron flux of approximately 50%. Use of this thickness yields a fast-neutron flux of

1.35×10^{11} neutrons/cm²-sec at the inner face of the converter cylinder, which satisfies the first specification on p. 5. The fast and thermal neutron-flux distribution for a U²³⁵-aluminum alloy thickness of $2w$ is presented graphically in Fig. 7. On the basis of the results shown in Figs. 6 and 7, a U²³⁵ content of 400 g was chosen for the converter.

Fig. 6. Normalized Flux Distribution for Different Thicknesses of U²³⁵ in Juggernaut East Thermal Column

Fig. 7. Fast and Thermal Neutron-flux Distribution in U²³⁵-Aluminum Alloy (*W = Thickness of U²³⁵ Equivalent to Thickness of Juggernaut Fuel Plate)

IV. MASS AND BURNUP CALCULATIONS

The specifications for the converter design require a thermal-neutron shield that will essentially eliminate all thermal neutrons. Calculations indicate that a boral shield containing 258.4 g of B₄C or 50 g of B¹⁰ will approximate the desired result. This will, therefore, require a boral shield containing approximately 16 w/o B₄C and 84 w/o aluminum. Boral with a one-to-one B₄C-to-aluminum ratio by weight can be obtained. Therefore, construction of the desired boral shield should present no problem. The calculations performed to determine the B¹⁰ and B₄C masses are presented in Appendix B.

The reaction rate for an element in a neutron flux is given by the expression,

$$\text{Reaction Rate} = \frac{(Na)m\sigma\phi}{A}, \quad (4.1)$$

where

Na = Avogadro's number;

m = mass of element present, g;

σ = reaction cross section, cm^2 ;

ϕ = neutron flux causing reaction, neutrons/ cm^2 -sec;

and

A = atomic number of element.

The percent burnup of an element can be calculated from the expression,

$$\% \text{ Burnup} = \frac{\text{Reaction Rate} \times \text{Time in Flux} \times 10^2}{\text{Number of Nuclei of Element}} \% \quad (4.2)$$

From Eq. (4.2), the following expressions are obtained for the percent burnup of the U^{235} and B^{10} :

$$\% \text{ Burnup of } \text{U}^{235} = 1.5 \times 10^{-4} T, \quad (4.3)$$

and

$$\% \text{ Burnup of } \text{B}^{10} = 9.82 \times 10^{-3} T, \quad (4.4)$$

where

T = irradiation time, hr.

For example, we will assume that the converter is used 10 hr/day for a year. This gives an irradiation time of approximately 300 hr. Using this value of T in Eqs. (4.3) and (4.4), we obtain a burnup of U^{235} of 0.045% and a burnup of B^{10} of 2.95%. Therefore, the burnup of U^{235} and B^{10} should present no problems. Calculations performed in obtaining Eqs. (4.3) and (4.4) are contained in Appendix B.

The following expressions can be used to determine the fraction of incident flux transmitted through a material as a function of time and thickness:³

$$\frac{\phi(z,t)}{\phi_0} = \frac{\exp(\sigma\phi_0 t)}{\exp(\sigma\phi_0 t) + \exp(\sigma N_0 z) - 1}, \quad (4.5)$$

where

t = exposure time, sec;

σ = reaction cross section, cm^2 ;

ϕ_0 = incident flux, neutrons/ cm^2 -sec;

N_0 = number of nuclei per unit volume, nuclei/ cm^3 ;

and

z = material thickness, cm.

Results obtained from Eq. (4.5) are shown in Fig. 8, where it is apparent that an appreciable increase in thermal flux transmitted by the U^{235} and boral does not occur until after long irradiation periods.

Fig. 8. Increase in Transmitted Thermal Flux with Irradiation Time

V. HEAT CALCULATIONS

This section will deal with heat generation in, and its removal from, the U^{235} -aluminum alloy and boral converter components. An attempt will be made to determine the temperature profile across the U^{235} -aluminum alloy and neutron shield material

as well as the temperature distribution along the coolant channel and metal-coolant interface.

A. Maximum Heat Generation in U^{235} -Aluminum Alloy and Boral at 200 kW

Calculations show that a maximum of 15 kW of heat can be expected to be generated in the proposed converter design when it is placed in Juggernaut's east thermal column. Eighty percent, or 12 kW, of the total heat is produced by the fission of the U^{235} . The remaining 20%, or 3 kW, is produced by thermal-neutron absorption in the boral thermal-neutron shield. Calculations performed to obtain the above values are presented in Appendix C.

B. Temperature Profile across U^{235} -Aluminum Alloy and Boral

Knowing the heat-generation rates, one can determine the temperature profile across the U^{235} -aluminum alloy and boral shield. It is assumed

that a thermal flux ϕ_{th} , impinges on the U^{235} -aluminum alloy and boral. This gives a uniform heat-generation rate in each material. The assumption is

also made that the inner face of the thermal-neutron shield is at 60°C . Using the model shown in Fig. 9 and the following heat-conduction equation,

$$-K \frac{d^2\theta}{dx^2} = Q(x), \quad (5.1)$$

we obtain the expressions for the temperature profile in the U^{235} -aluminum alloy and boral thermal-neutron shield:

$$\theta^U(x) = \frac{\xi^U (x_0^U)^2}{2} \left(\frac{x}{x_0^U} \right)^2 + \theta^{12} - \theta^C - \frac{\xi^U (x_0^U)^2}{2} \frac{x}{x_0^U} + \theta^C, \quad (5.2)$$

Fig. 9. Model Used for Temperature-profile Analysis

$$\theta^B(x) = (x_0^B)^2 \frac{\xi^B}{2} \left[\left(\frac{x}{x_0^B} \right)^2 - 1 \right] + \left[\theta^{12} - \theta^C + \frac{\xi^U (x_0^U)^2}{2} \right] \frac{K^U x_0^B}{K^B x_0^U} \left[\left(\frac{x}{x_0^B} \right) - 1 \right] + 60, \quad (5.3)$$

where

$\theta^U(x)$ = temperature in U^{235} -aluminum alloy at distance x from outer surface, $^\circ\text{C}$;

$\theta^B(x)$ = temperature in boral at distance x from U^{235} -aluminum alloy and boral interface, $^\circ\text{C}$;

x_0^U = U^{235} -aluminum alloy thickness, cm;

x_0^B = boral thickness, cm;

θ^{12} = U^{235} -aluminum alloy--boral interface temperature, $^\circ\text{C}$;

θ^C = coolant temperature, $^\circ\text{C}$;

K^U = U^{235} -aluminum alloy thermal conductivity, cal/sec-cm- $^\circ\text{C}$;

K^B = boral thermal conductivity, cal/sec-cm- $^\circ\text{C}$;

ξ^U = $-Q^U(x)/K^U$, $^\circ\text{C}/\text{cm}^2$;

$$\xi^B = -Q^B(x)/K^B, \text{ } ^\circ\text{C}/\text{cm}^2;$$

$$Q^U(x) = \text{heat-generation rate in } U^{235}\text{-aluminum alloy, cal/cm}^3\text{-sec};$$

and

$$Q^B(x) = \text{heat-generation rate in boral, cal/cm}^3\text{-sec.}$$

Equations (5.2) and (5.3) are plotted in Fig. 10. Figure 10 shows that no excessive U^{235} -aluminum alloy or boral temperatures will result even if 80°C coolant water is used. Equations (5.2) and (5.3) are derived in Appendix C.

C. Cooling of U^{235} -Aluminum Alloy and Boral

The next problem considered is the transfer of heat from the converter walls to the coolant. Our objective will be to determine if any excessive wall temperature or coolant temperature exists in the converter. Coolant is passed through channels adjacent to the outer U^{235} -aluminum

alloy wall and the inner boral wall. The U^{235} -aluminum alloy and boral are treated as having all the heat generated by the material, deposited on the metal-coolant surface interface. This heat is removed by forced convection of the coolant. The following equations can be applied to the model shown in Fig. 11:^{4,5}

$$q(z) = \frac{2Q_T}{3z_0} \left(1 + \frac{z}{z_0} \right), \quad (5.4)$$

Fig. 10. Temperature Profile of U^{235} -Aluminum Alloy and Boral

Fig. 11. Model Used for Heat-transfer Analysis

$$q(z) = w_f c_f \frac{d\theta_f}{dz}, \quad (5.5)$$

$$q(z) = bh(\theta_m - \theta_f), \quad (5.6)$$

$$\frac{hD_e}{K} = 0.023 \left(\frac{D_e G}{\mu} \right)^{0.8} \left(\frac{C_p \mu}{K} \right)^{0.4}, \quad (5.7)$$

when

$$\frac{D_e G}{\mu} > 2100 \text{ in Eq. (5.7),}$$

where

$q(z)$ = heat flux at point z of converter, cal/sec-cm;

Q_T = total heat flux, cal/sec;

z_0 = converter length, cm;

w_f = mass flow rate of coolant, g/sec;

c_f = specific heat of coolant, cal/g-°C;

θ_f = coolant temperature, °C;

h = film coefficient, cal/sec-cm²-°C;

b = wetted perimeter of metal, cm;

θ_m = metal temperature, °C;

K = coolant thermal conductivity, cal/sec-cm-°C;

D_e = equivalent diameter of flow channel, cm;

G = mass flow rate per flow-channel unit area, g/sec-cm²;

σ = cross-section area of flow channel, cm²;

and

μ = coolant viscosity, g/cm-sec.

Equation (5.4) is derived in Appendix C. Applying the conditions of Fig. 11 to Eqs. (5.4-5.7), we obtain the following expressions for the temperature distribution in the coolant channel and on the metal surface:

$$\theta_f(z) = z_0 \xi \left[\frac{z}{z_0} + \frac{1}{2} \left(\frac{z}{z_0} \right)^2 \right] + \theta_{IN}, \quad (5.8)$$

and

$$\theta_m(z) = \frac{2Q_T}{3z_0bh} \left(1 + \frac{z}{z_0}\right) + z_0 \xi \left[\frac{z}{z_0} + \frac{1}{2} \left(\frac{z}{z_0}\right)^2 \right] + \theta_{IN} \quad (5.9)$$

where

$$\xi = \frac{2Q_T}{3z_0w_f c_f}, \text{ } ^\circ\text{C/cm},$$

and

$$\theta_{IN} = \text{coolant inlet temperature, } ^\circ\text{C}.$$

Using the appropriate constants for the U^{235} -aluminum alloy coolant channel and the boral coolant channel, with Eqs. (5.8) and (5.9), we obtain expressions for the coolant and metal temperature distributions in the respective coolant channels. The calculated coolant temperature distribution for the U^{235} -aluminum alloy and boral coolants are shown in Figs. 12 and 13, respectively. The calculated U^{235} -aluminum alloy metal surface-temperature distribution is shown in Fig. 14, and the calculated boral metal surface-temperature distribution is shown in Fig. 15. Equations (5.8) and (5.9) are derived in Appendix C.

Fig. 12. Temperature Distribution of U^{235} -Aluminum Alloy Coolant

Fig. 13. Temperature Distribution of Boral Coolant

Fig. 14. Surface-temperature Distribution of U^{235} -Aluminum Alloy

Fig. 15. Surface-temperature Distribution of Boral

Figures 12, 13, 14, and 15 indicate that no excessive heating of coolant or metal will occur within the converter assembly if flow rates of 10 and 5 gal/min, respectively, are used in the U^{235} -aluminum alloy coolant channel and the boral coolant channel, with a coolant inlet temperature of 30°C .

VI. SUMMARY

From the foregoing discussion, it appears feasible to design a neutron-spectrum converter that will meet the requirements set forth in Section I. The converter design would contain 400 g of U^{235} alloyed with aluminum. The fuel alloy would be 0.08 in. thick and in the shape of a cylinder which is closed at one end. A 0.25-in.-thick boral thermal-neutron shield would be placed inside the fuel section. This would contain approximately 16 w/o B_4C , the remaining 84 w/o being aluminum. H_2O coolant channels, 0.10 in. wide, would be used to cool the fuel and boral cylinders. The coolant inlet temperature would be 30°C , and the total flow rate would be 15 gal/min. Sixty-six percent of the coolant would be used to cool the outside of the fuel cylinder, and the remaining 34% would be used to cool the inside of the boral cylinder. With these design specifications, the temperature inside the sample cavity would be less than 50°C . The proposed converter design is shown in Fig. 16.

Fig. 16. Proposed Converter Design

This report is intended only as a preliminary design report analyzing several of the many physics aspects of the converter. The design of auxiliary equipment such as pumping facilities, heat exchangers, shielding, and removal coffins are obviously missing. Also, actual details of how to construct the assembly have not been considered. These points would have to be considered in an engineering analysis.

APPENDIX A

Flux-distribution Calculations

Consider a model as shown in Fig. A-1. (Figure A-1 is the same as Fig. 5.) The U^{235} is alloyed with aluminum, is in the form of a slab of thickness Nw , and is of infinite extent in the y and z directions. There is a fast flux, $\phi_1(0)$, and a thermal flux, $\phi_2(0)$, incident upon the U^{235} at $x = 0$. The slab is bounded at $x = Nw$ by a vacuum. Using two-group diffusion theory, we may write

$$-D_1 \frac{d^2 \phi_1}{dx^2} + (\Sigma_{A1} + \Sigma_{R1}) \phi_1 = \sum_{i=1,2} (v_i \Sigma_{Fi} \phi_i), \quad (A-1)$$

and

Fig. A-1. Model Used for Two-group Analysis of Flux Distribution

$$-D_2 \frac{d^2 \phi_2}{dx^2} + \Sigma_{A2} \phi_2 = \Sigma_{R1} \phi_1, \quad (A-2)$$

where,

D_2 = diffusion coefficient for thermal-neutron group, cm^{-1} ;

D_1 = diffusion coefficient for fast-neutron group, cm^{-1} ;

ϕ_2 = thermal-neutron flux, neutrons/ cm^2 -sec;

ϕ_1 = fast-neutron flux, neutrons/ cm^2 -sec;

v_i = fission-neutron yield, neutrons/fission;

Σ_{Fi} = fission cross section, cm^{-1} ;

Σ_{A2} = thermal-neutron-absorption cross section, cm^{-1} ;

Σ_{A1} = fast-neutron-absorption cross section, cm^{-1} ;

and

Σ_{R1} = fast-neutron-removal cross section, cm^{-1} .

Since U^{235} is a heavy element, it will be assumed that $\Sigma_{R1} = 0$. Letting $\Sigma_{A2}/D_2 = \gamma^2$, Eq. (A-2) then reduces to

$$(D^2 - \gamma^2) \phi_2 = 0. \quad (A-3)$$

The general solution of Eq. (A-3) is

$$\phi_2(x) = E \exp(-\gamma x) + F \exp(\gamma x). \quad (\text{A-4})$$

From Fig. A-1, the following conditions apply:

$$\text{a) } \phi_2(x) = \phi_2(0), \quad x = 0;$$

$$\text{b) } (d\phi_2/dx) = 0, \quad x = Nw.$$

Using these conditions with Eq. (A-4), we obtain the following expression for the thermal-neutron distribution in the U^{235} :

$$\phi_2(x) = \phi_2(0) \frac{\cosh [\gamma(Nw - x)]}{\cosh (\gamma Nw)}. \quad (\text{A-5})$$

Equation (A-5) is the same as Eq. (3.3). Combining Eqs. (A-1) and (A-5) and letting

$$\frac{\Sigma_{A1} - v_1 \Sigma_{F1}}{D_1} = \alpha^2,$$

and

$$\frac{v_2 \Sigma_{F2} \phi_2(0)}{D_1 \cosh (\gamma Nw)} = \xi^2,$$

we obtain

$$(D^2 - \alpha^2)\phi_1(x) = -\xi^2 \cosh [\gamma(Nw - x)]. \quad (\text{A-6})$$

The general solution of Eq. (A-6) is

$$\phi_1(x) = A \exp(-\alpha x) + B \exp(\alpha x). \quad (\text{A-7})$$

To obtain a particular solution of Eq. (A-6), let

$$\phi_1(x) = K \cosh [\gamma(Nw - x)] + J \sinh [\gamma(Nw - x)]. \quad (\text{A-8})$$

Therefore, we obtain the following complete solution of Eq. (A-6):

$$\phi_1(x) = A \exp(-\alpha x) + B \exp(\alpha x) + \frac{\xi^2}{\alpha^2 - \gamma^2} \cosh [\gamma(Nw - x)]. \quad (\text{A-9})$$

From the situation shown in Fig. (A-1), we obtain the following conditions:

- a) $\phi_1(x) = \phi_1(0), \quad x = 0;$
 b) $(d\phi_1/dx) = 0, \quad x = Nw.$

Using these conditions with Eq. (A-9), we obtain the following expression for the fast neutron-flux distribution in the U^{235} :

$$\begin{aligned} \phi_1(x) = & \phi_1(0) - \frac{\xi^2 \cosh(\gamma Nw)}{\alpha^2 - \gamma^2} \frac{\cosh[\alpha(Nw - x)]}{\cosh(\alpha Nw)} \\ & + \frac{\xi^2}{\alpha^2 - \gamma^2} \cosh[\gamma(Nw - x)]. \end{aligned} \quad (A-10)$$

Equation (A-10) is the same as Eq. (3.4). Several assumptions must be made to render Eqs. (A-5) and (A-10) usable. We have

$$D = \frac{1}{(1 - \mu) \Sigma_S + \Sigma_A},$$

where

$$\Sigma_S \ll \Sigma_A,$$

and

$$1 - \mu \approx 1.$$

Therefore, we assume that

$$D = \frac{1}{\Sigma_A}.$$

Equations (A-5) and (A-10) give the neutron-flux distribution in a slab of U^{235} of thickness Nw . The quantity w will be considered as being the equivalent thickness of U^{235} in a Juggernaut-type fuel plate. The U^{235} -aluminum alloy in a Juggernaut-type fuel plate is 0.04 in. thick. Since the converter is 12 in. long and 4 in. in diameter, a converter fuel surface area of 150.8 in.² and an active fuel volume of 6.032 in.³ results. The converter would contain 160 g of U^{235} . This gives a U^{235} concentration of 26.54 g U^{235} per in.³, or 1.62 g of U^{235} per cm³. The volume of a fuel section of 1 sq cm surface area and 0.04-in. thickness is equal to 0.102 cm³. Therefore,

$$\frac{\text{Mass of } U^{235}}{\text{Total Fuel Volume}} \times \text{Volume of 1 cm}^2 \text{ Section} = 0.165 \text{ g}$$

of U^{235} in a fuel section of 1 sq cm surface area and of 0.04-in. thickness.

Replacing the U^{235} -aluminum alloy with an equivalent thickness of pure U^{235} , we have

$$1 \text{ cm}^2 \times w \text{ cm} \times \rho_{U^{235}} \text{ g/cm}^3 = 0.165 \text{ g},$$

where $\rho_{U^{235}}$ = density of U^{235} , g/cm³. Therefore, w , the equivalent thickness of U^{235} for a Juggernaut-type fuel plate, is 0.9×10^{-2} cm.

We therefore use the following constants when evaluating Eqs. (A-5) and (A-10):

$$w = 0.9 \times 10^{-2} \text{ cm},$$

$$\Sigma_{A2} = 32.98 \text{ cm}^{-1},$$

$$\Sigma_{A1} = 4.32 \text{ cm}^{-1},$$

$$\Sigma_{F2} = 27.84 \text{ cm}^{-1},$$

$$\Sigma_{F1} = 2.88 \text{ cm}^{-1},$$

$$v_1 = v_2 = 2.5 \text{ neutrons/fission},$$

$$\phi_2(0) = 4.5 \times 10^{11} \text{ neutrons/cm}^2\text{-sec},$$

$$\phi_1(0) = 9.0 \times 10^{10} \text{ neutrons/cm}^2\text{-sec},$$

and

$$N = 1, 2, 3, \dots$$

Solutions of Eqs. (A-5) and (A-10) normalized to $\phi_1(0)$ and $\phi_2(0)$ are already given in Fig. 6. The neutron-flux distribution in a U^{235} thickness of $2w$ is shown in Fig. 7.

APPENDIX B

Mass and Burnup Calculations for U^{235} and B^{10}

One specification placed on the converter design was the essentially complete elimination of the thermal-neutron flux in the sample irradiation cavity. Therefore, some type of material must be used to absorb the thermal neutrons present in the flux converter. A mixture of B_4C and aluminum, commonly called boral, was chosen to act as the thermal-neutron shield. The major portion of thermal-neutron absorption is caused by the B^{10} in the boral shield. Therefore, we will determine the B^{10} thickness needed to essentially eliminate the thermal-neutron flux. This will be the equivalent thickness of B^{10} for the boral. We may write

$$\phi = \phi_0 \exp(-\Sigma x), \quad (B-1)$$

where

ϕ = flux transmitted, neutrons/cm²-sec;

ϕ_0 = incident flux, neutrons/cm²-sec;

x = material thickness, cm;

and

Σ = absorption cross section, cm⁻¹.

Normalizing Eq. (B-1) to the incident flux and solving for x , we obtain

$$x = -\frac{1}{\Sigma} \ln \left(\frac{\phi}{\phi_0} \right), \quad (B-2)$$

where

$$\frac{1}{\Sigma} = 1.882 \times 10^{-3} \text{ cm}$$

for B^{10} . Therefore,

$$x = -1.882 \times 10^{-3} \ln \left(\frac{\phi}{\phi_0} \right). \quad (B-3)$$

To obtain 99% attenuation of incident flux, we need 8.667×10^{-3} cm of B^{10} . The total surface area of the thermal-neutron shield is approximately 973 cm². Therefore, we have 8.433 cm³ of B^{10} , or approximately 20 g of B^{10} . To allow for burnup, the B^{10} content is increased by a factor of two. We therefore specify that 40 g of B^{10} is needed in the thermal-neutron shield. This gives an attenuation greater than 99.9%.

We can now determine the B_4C mass content needed for the shield.
We have

$$B^{10} \text{ mass} = 40 \text{ g,}$$

$$B^{10} \text{ abundance} = 19.78\% \text{ of natural boron,}$$

$$\text{AMU of boron} = 10.811,$$

$$\text{AMU of carbon} = 12.0115,$$

and

$$\rho_B = 2.3 \text{ g/cm}^3.$$

We determine that the percentage of boron in B_4C is 78.26 w/o, and the percentage of B^{10} in B_4C is 15.47 w/o. Thus, 258.4 g or 103.36 cm^3 of B_4C is required in the thermal-neutron shield. The shield is assumed to be 0.25 in. thick. This gives a shield volume of 600 cm^3 . Therefore, we have $600 - 100 = 500 \text{ cm}^3$, or 1350 g of aluminum. This indicates that the B_4C content in the thermal-neutron shield is approximately 16 w/o.

The reaction rate for a mass (m) of material in a flux, ϕ , is given by

$$\text{Reaction Rate} = \frac{(\text{Na})m\sigma\phi}{A} \quad (\text{B-4})$$

where

Na = Avogadro's number, nuclei/g-atom;

m = mass of element present, g;

σ = reaction cross section, cm^2 ;

ϕ = flux causing reaction, neutrons/ cm^2 -sec;

and

A = atomic number of element, g/g-atom.

The number of nuclei of an element of mass, m , is given by

$$\text{No. of Nuclei} = \frac{(\text{Na})m}{A} \quad (\text{B-5})$$

The percent burnup of an element is the ratio of the number of nuclei consumed in the reaction, to the total number of nuclei originally present. Therefore,

$$\% \text{ Burnup} = \frac{\text{Reaction Rate} \times \text{Time} \times 10^2}{\text{No. of Nuclei}} \quad (\text{B-6})$$

Equation (B-6) gives an expression for determining the percent burnup of the U^{235} and B^{10} in the flux converter. The following constants apply to this situation:

$$m(U^{235}) = 400 \text{ g,}$$

$$m(B^{10}) = 40 \text{ g,}$$

$$N_a = 6.023 \times 10^{23} \text{ nuclei/g-atom,}$$

$$\sigma(U^{235}) = 700 \text{ barns,}$$

$$\sigma(B^{10}) = 4 \times 10^3 \text{ barns,}$$

$$\phi = 6 \times 10^{11} \text{ neutrons/cm}^2\text{-sec,}$$

$$A(U^{235}) = 235 \text{ g/g-atom,}$$

and

$$A(B^{10}) = 10.013 \text{ g/g-atom.}$$

Using the above constants, we obtain the following expressions for the percent burnup of the U^{235} and B^{10} :

$$\% \text{ Burnup of } U^{235} = 1.5 \times 10^{-4} T, \quad (B-7)$$

$$\% \text{ Burnup of } B^{10} = 9.82 \times 10^{-3} T, \quad (B-8)$$

where

T = irradiation time, hr.

Equations (B-7) and (B-8) are the same as Eqs. (4.3) and (4.4), respectively.

APPENDIX C

Heat-generation and Temperature-distribution Calculations

The heat-generation rate for the flux converter can be calculated from the following expression:

$$\text{Heat-generation Rate} = \text{Reaction Rate} \times \frac{\text{Energy Release}}{\text{Reaction}}, \quad (\text{C-1})$$

where

$$\text{Reaction Rate} = \frac{(Na)m\sigma\phi}{A},$$

Na = Avogadro's number, nuclei/g-atom;

m = mass of element present, g;

σ = cross section for reaction, cm^2 ;

ϕ = flux causing reaction, neutrons/ cm^2 -sec;

A = atomic number of element, g/g-atom;

and

$$\frac{\text{Energy Release}}{\text{Reaction}} = W/\text{reaction}.$$

For the U^{235} we have the following constants:

$$\text{Na} = 6.023 \times 10^{23} \text{ nuclei/g-atom},$$

$$m(\text{U}^{235}) = 400 \text{ g},$$

$$\sigma(\text{U}^{235}) = 600 \text{ barns},$$

$$A(\text{U}^{235}) = 235 \text{ g/g-atom},$$

and

$$\phi = 6 \times 10^{11} \text{ neutrons/cm}^2\text{-sec}.$$

For the U^{235} fission, we have 1 W released for 3.1×10^{10} fission/sec. Therefore, the heat generation for U^{235} is

$$H(\text{U}^{235}) = \frac{3.69 \times 10^{14} \text{ fissions/sec}}{3.1 \times 10^{10} \text{ fissions/sec/W}} = 12 \text{ kW}. \quad (\text{C-2})$$

The B^{10} reaction is

Using the following atomic mass numbers, we obtain an atomic mass defect of 0.0033 AMU:

$${}_5\text{B}^{10} = 10.01924 \text{ AMU},$$

$${}_0\text{n}^1 = 1.00897 \text{ AMU},$$

$${}_3\text{Li}^7 = 7.01601 \text{ AMU},$$

and

$${}_2\text{He}^4 = 4.00260 \text{ AMU}.$$

Using the following conversion factors:

$$1 \text{ AMU} = 931 \text{ MeV},$$

and

$$1 \text{ MeV} = 1.603 \times 10^{-13} \text{ W},$$

we obtain a heat generation/reaction of $4.921 \times 10^{-13} \text{ W/reaction}$. The following constants are used for the B^{10} reaction:

$$m(\text{B}^{10}) = 40 \text{ g},$$

$$\sigma(\text{B}^{10}) = 4000 \text{ barns},$$

$$A(\text{B}^{10}) = 10.013 \text{ AMU},$$

and

$$\phi = 6 \times 10^{11} \text{ neutrons/cm}^2\text{-sec}.$$

Therefore, we obtain the following value for heat generation for the B^{10} :

$$\begin{aligned} H(\text{B}^{10}) &= 5.77 \times 10^{15} \text{ reactions} \times 4.921 \\ &\times 10^{-13} \text{ W/reaction} = 3 \text{ kW}. \end{aligned} \quad (\text{C-3})$$

The total heat generation in the converter is therefore 15 kW.

Knowing the heat-generation rates, we can determine the temperature profile across the U^{235} -aluminum alloy and boral shield. A model of the existing conditions is shown in Fig. C-1. (Figure C-1 is the same as Fig. 9.) We may apply the following heat-conduction equation to each layer of material

$$-K \frac{d^2\theta}{dx^2} = Q(x), \quad (\text{C-4})$$

Fig. C-1. Model Used for Temperature-profile Analysis

where

K = thermal conductivity of material, cal/sec-cm-°C;

and

$Q(x)$ = heat generation rate, cal/sec-cm³.

We will consider $Q(x)$ as a constant quantity, for each material, equal to (the reaction rate) $\times$ (the energy released per reaction); thus,

$$Q(x) = \frac{(Na)\rho\sigma\phi E}{A},$$

where

ρ = density of the material, g/cm³;

and

E = energy released/reaction, cal/reaction.

Letting

$$-\frac{Q(x)}{K} = \xi,$$

Eq. (C-4) reduces to the following differential equation:

$$D_x^2 \theta = \xi. \quad (C-5)$$

The solution of Eq. (C-5) yields

$$\theta = \frac{\xi x^2}{2} + C_1 x + C_2. \quad (C-6)$$

Equation (C-6) can be applied to both the U^{235} material and the boron material.

For the U^{235} in Fig. C-1 we have the following conditions:

$$a) \theta^U = \theta^C, \quad x = 0;$$

$$b) \theta^U = \theta^{12}, \quad x = x_0^U.$$

Using these conditions with Eq. (C-6), we obtain the following expression for the temperature distribution in the U^{235} material:

$$\theta^U(x) = \frac{\xi^U}{2} (x_0^U)^2 \left(\frac{x}{x_0^U} \right)^2 + \theta^{12} - \theta^C - \frac{\xi^U}{2} (x_0^U)^2 \frac{x}{x_0^U} + \theta^C, \quad (C-7)$$

where

$\theta^U(x)$ = U^{235} -aluminum alloy temperature, °C;

x_0^U = U^{235} -aluminum alloy thickness, cm;

θ^{12} = U^{235} -aluminum alloy B¹⁰ interface temperature, °C;

θ^C = U^{235} -aluminum alloy coolant temperature, °C;

ξ^U = $-Q(x)U/K^U$, °C/cm²;

$Q(x)U$ = U^{235} -aluminum alloy heat generation rate, cal/sec-cm³;

and

K^U = U^{235} -aluminum alloy thermal conductivity, cal/sec-cm-°C.

Equation (C-7) is the same as Eq. (5.2).

We have the following conditions for the boral material in Fig. C-1:

$$a) \quad -k^B \frac{d\theta^B}{dx} \bigg|_{x=0} = -K^U \frac{d\theta^U}{dx} \bigg|_{x=x_0^U}$$

$$b) \quad \theta^B = 60^\circ\text{C}, \quad x = x_0^B.$$

Using these conditions with Eqs. (C-6) and (C-7), we obtain the following expression for the temperature profile in the boral material:

$$\begin{aligned} \theta^B(x) = & (x_0^B)^2 \frac{\xi^B}{2} \left[\left(\frac{x}{x_0^B} \right)^2 - 1 \right] \\ & + \left[\theta^{12} - \theta^C + \frac{\xi^U (x_0^U)^2}{2} \right] \frac{K^U x_0^B}{K^B x_0^U} \left[\left(\frac{x}{x_0^B} \right) - 1 \right] + 60, \end{aligned} \quad (C-8)$$

where

$\theta^B(x)$ = boral temperature, °C;

x_0^B = boral thickness, cm;

ξ^B = $-Q^B(x)/K^B$, °C/cm²;

$$Q^B(x) = \text{boral heat-generation rate, cal/sec-cm}^3;$$

and

$$K^B = \text{boral thermal conductivity, cal/sec-cm-}^\circ\text{C.}$$

Equation (C-8) is the same as Eq. (5.3). The following constants were used in obtaining the graphical representation of Eqs. (C-7) and (C-8) shown in Fig. 10:

$$N_a = 6.023 \times 10^{23} \text{ nuclei/g-atom,}$$

$$\rho^B = 2.3 \text{ g/cm}^3,$$

$$\rho^U = 18.7 \text{ g/cm}^3,$$

$$\sigma^B = 3.8 \times 10^3 \text{ barns,}$$

$$\sigma^U = 6.9 \times 10^2 \text{ barns,}$$

$$\phi_{th} = 6 \times 10^{11} \text{ neutrons/cm}^2\text{-sec,}$$

$$E^B = 1.175 \times 10^{-13} \text{ cal,}$$

$$E^U = 7.65 \times 10^{-12} \text{ cal,}$$

$$K^B = 0.08 \text{ cal/sec-cm-}^\circ\text{C,}$$

$$K^U = 0.40 \text{ cal/sec-cm-}^\circ\text{C,}$$

$$A^B = 10.013 \text{ g/g-atom,}$$

$$A^U = 235 \text{ g/g-atom,}$$

$$x_0^B = 0.635 \text{ cm,}$$

and

$$x_0^U = 0.102 \text{ cm.}$$

Now we will attempt to determine the temperature distribution on the metal coolant interface, and the coolant temperature distribution along the coolant channel. A model of the existing conditions is shown in Fig. C-2. (Figure C-2 is the same as Fig. 11.) Before trying to determine the temperature distributions, we must obtain an expression for the heat flux in the converter as a function of position along the converter. To do this, we assume that the heat flux is directly proportional to the neutron flux. We already have the neutron-flux distribution from our flux-map data. This is shown in Figs. 2 and 3. Figure 3 indicates that a linear flux distribution can be assumed for the first 12 in. of the beam hole. Therefore, we may write the following expression for the neutron-flux distribution in Juggernaut's east thermal column:

$$\phi(z) = \phi_0(1 + z/z_0), \quad (\text{C-9})$$

Fig. C-2. Model Used for Heat-transfer Analysis

where

$\phi(z)$ = flux at distance z from end of converter, neutrons/cm²-sec;

z_0 = converter length, cm;

and

ϕ_0 = flux at end of converter, neutrons/cm²-sec.

Subsequently, we write the following expression for the heat-flux distribution in the flux converter:

$$q(z) = q_0(1 + z/z_0) \quad (C-10)$$

where

$q(z)$ = heat flux at point z in converter, cal/sec-cm;

q_0 = heat flux at end of converter, cal/sec-cm;

and

z_0 = converter length, cm.

The total heat flux in the converter is given by

$$Q_T = \int_0^{z_0} q(z) dz. \quad (C-11)$$

From Eqs. (C-10) and (C-11) we determine that $q_0 = 2Q_T/3z_0$. Therefore,

$$q(z) = \frac{2Q_T}{3z_0} \left(1 + \frac{z}{z_0}\right). \quad (C-12)$$

Equation (C-12) is the same as Eq. (5.4). We can write two more equations involving $q(z)$:⁴

$$q(z) = w_f c_f \frac{d\theta_f}{dz}, \quad (C-13)$$

and

$$q(z) = bh(\theta_m - \theta_f). \quad (C-14)$$

We may also write the following expression to define the film coefficient h :⁵

$$\frac{hD_e}{K} = 0.023 \left(\frac{D_e G}{\mu} \right)^{0.8} \left(\frac{C_p \mu}{K} \right)^{0.4} \quad (C-15)$$

when

$$\frac{D_e G}{\mu} > 2100.$$

The following definitions hold for Eqs. (C-12), (C-13), (C-14), and (C-15):

$q(z)$ = heat flux at point z of converter, cal/sec-cm;

Q_T = total heat flux, cal/sec;

z_0 = converter length, cm;

w_f = mass flow rate of coolant, g/sec;

c_f = specific heat of coolant, cal/g-°C;

θ_f = coolant temperature, °C;

h = film coefficient, cal/sec-cm²-°C;

b = wetted perimeter of metal, cm;

θ_m = metal temperature, °C;

K = coolant thermal conductivity, cal/sec-cm-°C;

D_e = equivalent diameter of flow channel, cm;

G = mass flow rate per flow-channel unit area, g/sec-cm²;

σ = cross-section area of flow channel, cm²;

and

μ = coolant viscosity, g/sec-cm.

Combining Eqs. (C-12) and (C-13), and letting $2Q_T/3z_0w_fc_f = \xi$, we obtain the following differential equation:

$$\frac{d\theta_f}{dz} = \xi \left[1 + \left(\frac{z}{z_0} \right) \right] \quad (C-16)$$

Using the condition that

$$\theta_f = \theta_{IN}$$

$$z = 0$$

with Eq. (C-16), we obtain the following expression for the coolant temperature distribution:

$$\theta_f(z) = z_0 \xi \left[\frac{z}{z_0} + \frac{1}{2} \left(\frac{z}{z_0} \right)^2 \right] + \theta_{IN}, \quad (C-17)$$

where

$\theta_f(z)$ = coolant temperature at distance z from entrance to coolant channel, °C;

z_0 = coolant-channel length, cm;

and

θ_{IN} = coolant inlet temperature, °C.

Equation (C-17) is the same as Eq. (5.8). Combining Eqs. (C-12), (C-14), and (C-17), we obtain the following expression for the temperature distribution along the metal-coolant interface:

$$\theta_m(z) = \frac{2Q_T}{3z_0bh} \left(1 + \frac{z}{z_0} + z_0 \xi \left[\frac{z}{z_0} + \frac{1}{2} \left(\frac{z}{z_0} \right)^2 \right] \right) + \theta_{IN}. \quad (C-18)$$

Equation (C-18) is the same as Eq. (5.9). The following constants were used in determining h values for the coolant channels:

$$c_f = 0.9976 \text{ cal/g-}^\circ\text{C},$$

$$z_0 = 30.48 \text{ cm},$$

$$b = 31.92 \text{ cm},$$

$$D_e = 1.016 \text{ cm},$$

$$\sigma = 8.1073 \text{ cm}^2,$$

$$K = 1.5 \times 10^{-3} \text{ cal/sec-cm-}^\circ\text{C},$$

and

$$\mu = 0.656 \times 10^{-2} \text{ g/cm-sec.}$$

The h values were calculated for flow rates of 5 and 10 gal/min. These h values and other pertinent constants are listed in Table I.

TABLE I. Constants Used in Temperature-profile Calculations

Flow rate, gal/min	$\left(\frac{C_p \mu}{K} \right)^{0.4}$	$\frac{D_e G}{\mu}$	$\left(\frac{D_e G}{\mu} \right)^{0.8}$	$h, \frac{\text{cal}}{\text{sec-cm}^2-^\circ\text{C}}$
10	1.8026	1.205×10^4	1.8399×10^3	0.11214
5	1.8026	6.025×10^3	1.0567×10^3	0.06441

The resulting temperature distribution for the U^{235} -aluminum alloy and boral coolant channels are shown graphically in Figs. 12-15.

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