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DECAY HEATING IN THE ETR CORE WITH THE FEFPL LOOP, AND IN THE ETR CANAL STORAGE

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AEROJET NUCLEAR COMPANY

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ABSTRACT

This report accomplishes two purposes. First, a detailed description of the computer oriented methods used at Aerojet Nuclear Company to calculate decay heating is given. This can be used as a guide for future calculations of decay heating. Second, the methods are applied to calculate the decay heating in the ETR core at various times after shutdown following 56.7 days operation at 175 MW. The methods are sufficiently versatile to provide for estimating the distribution of heat generation within fuel elements, and in adjacent regions.

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1.0 PURPOSE

The data reported here were requested by the engineers preparing the ETR Technical Specifications. Calculated values are given for the heat generation in the ETR core subsequent to shutdown after 45.7 days operation at 175 MW with the FEFPL loop installed. Results are given, also, for the heat generation in the ETR canal storage assuming the cooling water has been accidentally lost. These data provide bases for determining cooling requirements, temperature induced stresses and other properties which must be known in establishing technical specifications for the ETR.

The calculational methods used are described in considerable detail. This may help others who are confronted for the first time with the task of calculating the heating arising from release of gamma rays, beta particles and, possibly, delayed neutrons from fission products in the ETR or other cores.

2.0 SUMMARY OF RESULTS

1. Seventy-five percent of the gamma energy release following shutdown is retained in the core. This fraction was calculated for the sources prevailing immediately after shutdown. Some increase in this fraction is to be expected at later times because the fission product gamma ray spectrum degrades to lower energy as time goes on. However, 75% is a good gross number for use in this analysis.
2. Gammas from neutron activation of core materials make a negligible contribution to the decay heating.
3. The hot spot for the heat generation after reaching xenon equilibrium was determined to be in fuel position 7I as shown in Figure 1.
4. The distribution of gamma heating around the hot spot is displayed in Figures 2-7 for various times after shutdown. Heating rates in aluminum, water and fuel "meat" are displayed. The maximum heat rate is in the fuel meat at the center of position 7I and goes from 50 watts/cc just after shutdown to 9 w/cc one hour after shutdown.
5. The distribution of beta heating in the fuel "meat" of the "hot" element is displayed in Figures 8, 9. It is a maximum at the

outer corners of the fuel element and at this point goes from 700 w/cc just after shutdown to 100 w/cc one hour after shutdown. These heating rates are about 10 times higher than the gamma heating rates because beta particles release all their energy in the meat.

6. The fission heat has the peak in the same location as that for the beta heating. It goes from 700 w/cc just after shutdown to 36 w/cc one minute after shutdown. Since the longest lived emitter has a 56 second half life, the fission heat quickly becomes negligible after one minute. The distribution in the hot element is shown in Figures 10, 11. The time decay is shown in Figure 21. This gives the ratio of the fission heat at time t after shutdown to that immediately following shutdown after all prompt neutrons from full power operation are gone.
7. Tables I and II summarize the heat contribution in the fuel meat and give the totals for several times after shutdown. These totals are presented in Figures 12, 13. The fission and beta heat are comparable just after shutdown, but the fission heat declines rapidly in relative importance.
8. Table III gives the heat generation in cadmium and in the aluminum after the fuel is stored in the dry canal rack. The times are from 50 to 200 days after shutdown. Table IV gives the peak heat production in the fuel meat arising from beta decay at these times.
9. Section 7 presents a method for adjusting beta and fission heat to take account of alteration in power distribution. Some discussion of corrections to gamma heating is given, also. However, for large changes in power distribution and core composition, the gamma heating will require recalculation. The average source densities in Tables VII and VIII used in the present calculations could be adapted for use with altered power distributions. Data compiled in Table XII giving absorption cross sections for gamma rays in aluminum, water and fuel meat would be useful in repeating gamma heating calculations.

3.0 OUTLINE OF COMPUTATIONAL PROCEDURE

3.1 Calculation of Sources of Gamma Photons and Beta Particles

The largest sources are from decay of fission products. These products are distributed like the power so the first problem is to determine the power distribution. In this work, ETRC measurements designed to reveal the power distribution after attaining xenon equilibrium were mainly used.⁽¹⁾ Some computational results^(2,3) were employed, also, to arrive at a final version of the local power distribution within fuel elements and control rod fuel sections.

To determine photon and beta particle sources for any time after shutdown, the SINBAD⁽⁴⁾ and ISOSHLD⁽⁵⁾ codes were used. Only sources from decay of fission products were calculated because it was shown that radioactive nuclides produced by neutron activation were a negligible source compared with the fission products.

The ISOSHLD and SINBAD codes contain libraries of fission yields, decay chains, gamma and beta energies and capture cross sections. If the average power density during operation, and the time of operation, are known, these codes can give the average photons/cc sec in each photon energy group at any time during operation or after shutdown. In addition, the SINBAD code gives a convenient calculation of the average energy density at these times due to beta particle release.

3.2 Calculation of the Fraction of the Gamma Energy Release Which is Deposited in the Core

A special application of the point kernel method for calculating photon fluxes was used. In essence, the rate of the energy flux at an exterior point calculated with and without attenuating material in the core was taken to be a measure of the fraction of photon energy which escapes the core. The QAD code⁽⁶⁾ was used in this application.

3.3 Calculation of the Gamma Heating Near the "Hot Spot"

First the "hot spot" location must be identified. Then the flux of gamma rays in each energy group must be calculated for selected points at

and near the hot spot. The heat generation rate at any point in some specified material (aluminum, water, fuel "meat") is the product of the photon flux in each energy group multiplied by the group energy and the macroscopic energy absorption cross section for the material, summed over all energy groups. The QAD point kernel code does all of these operations using as input data for the average source density, the source distribution, the attenuation properties of the medium and the group wise energy absorption coefficients.

3.4 Calculation of the Beta Heating

This is easily accomplished because beta particles release their energy near the point of origin. The distribution of power from this source is exactly like that for the fission power in the operating reactor. The SINBAD output gives the necessary information about the average beta power.

3.5 Calculation of the Heat Production from Fissioning Induced by Delayed Neutrons

A special method is presented for this in Section 5.6. It is based on an approximate equation for decay of power after shutdown. The distribution of fission heating in the hot fuel element is assumed to be the same as for the power prior to shutdown. This is a conservative approach.

3.6 Calculation of Decay Heating in the Dry Canal Storage Rack

This calculation is simple. The fission product sources of gamma photons and beta particles are calculated from SINBAD for any desired shutdown time. The calculation of gamma fluxes and gamma heating parallels that for the gamma heating calculation in the core. The beta heating at the hottest spot is readily determined using the same methods as applied in the core.

3.7 Application of These Results to Different Core Loadings with Altered Power Distributions

This is easily accomplished for the beta and delayed neutron fission heating provided the operating time remains at 45.7 days. The calculation

of the gamma energy production can be expedited using sources obtained from ISOSHLD or SINBAD for this work. However, considerable effort will be required for significant changes in power distribution and core composition.

4.0 ETR CORE CONFIGURATION AND COMPOSITIONAL DATA

4.1 Geometry and Power Distribution

The ETRC was used to determine the core loading suited to giving FEFPL the best possible distribution of neutrons from the core. Figure 1 shows the loading pattern recommended on the basis of these studies⁽¹⁾. The fraction of the total power carried by each fuel element or control rod fuel section at operation conditions after xenon equilibrium is shown.

The power distribution in Figure 1 was used to determine that the greatest post shutdown heat generation rate would be in and near the 7I position.

In addition to the horizontal plane measurements shown in Figure 1, vertical (axial) power distributions were measured for all positions. These distributions were quite similar throughout the core and in this work it is assumed that they are all the same as for the measurements in the hot element in position 7I. Figure 14 gives this distribution normalized so that

$$\int_0^{\hat{Z}} z(z) dz = \hat{Z}; \hat{Z} = \text{full height of core (91.44 cm)} .$$

See Section 5.1 for the basis of this normalization.

Details of the local power distribution around the hot spot were inferred from several sources of data. These included ETRC measurements and computational results^(2,3). They will be discussed in detail in the relevant sections.

4.2 Composition

The atom densities needed to calculate the gamma ray attenuation and other properties were obtained from several sources⁽⁷⁾. Table V summarizes the data. Table VI gives a special homogenization which was used in calculating the gamma ray leakage.

4.3 The FEFPL Loop

Figure 15 displays a horizontal cross section of the FEFPL loop. It is included for information purposes. For the present calculation it was deemed adequate to homogenize all materials inside the 6" x 6" area containing the loop. The power generated in the loop was assumed to be generated within this volume and to have the same vertical distribution as that which applies to the entire core. These gross simplifications should be adequate because the gamma rays from the loop make a very minor contribution to the heat generation around the hot spot.

5.0 COMPUTATIONS

5.1 Power Distributions

5.1.1 Representation of the power distribution

All calculations are based on rectangular geometry. The power distribution used to define the photon and beta distribution in any region, local or extended, is assumed to be separable in the form

$$p(x,y,z) = \bar{p}X(x)Y(y)Z(z) \quad (1)$$

$$p(x,y,z) = \text{watts/cc at } x, y, z$$

$$\bar{p} = \text{average power density in the volume under question}$$

$$\bar{p} = \frac{P}{V}$$

$$P = \text{total power in the volume (watts)}$$

$$V = \text{volume size (cc)}$$

The functions $X(x)$, $Y(y)$ and $Z(z)$ tell how p is distributed along x , y and z respectively within the region under consideration.

The normalization is determined by the requirement:

$$\int_0^{\hat{Z}} \int_0^{\hat{Y}} \int_0^{\hat{X}} p(x,y,z) dx dy dz = \bar{p}V = \bar{p} \int_0^{\hat{Z}} Z(z) dz \int_0^{\hat{Y}} Y(y) dy \int_0^{\hat{X}} X(x) dx \quad (2)$$

In Eq. (2), $\hat{X}$, $\hat{Y}$ and $\hat{Z}$ are the dimensions of V. This leads, using $V = \hat{X}\hat{Y}\hat{Z}$ to the requirement

$$\int_0^{\hat{Z}} Z(z) dz = \hat{Z} ; \int_0^{\hat{Y}} Y(y) dy = \hat{Y} ; \int_0^{\hat{X}} X(x) dx = \hat{X} . \quad (3)$$

5.1.2 Condition of validity of Equation (1)

The separation in Eq. (1) cannot be valid over large volumes in the ETR. The ETRC data do indicate that one axial (z) component, $Z(z)$, should apply reasonably well in all fuel locations. Equation (1) should be reasonable, also, for expressing the planar x,y distribution within a single fuel element such as that in the "hot" position 7I or within a single control rod fuel section such as 7H or 7K.

In this work, Eq. (1) was used as follows:

- a. For calculating the gamma leakage, an artificial model was assumed which permitted using Eq. (1) for the entire core. Justification for this application will be given in the appropriate section.
- b. For calculating the heat release due to beta particles and fissioning by delayed neutrons, Eq. (1) was applied to a single fuel element or control rod fuel section. The $X(x)$ and $Y(y)$ were inferred from ETRC measurements and from calculational results.
- c. For calculating the photon sources to be used in the QAD code, Eq. (1) was used to express the distribution in a column of fuel elements, control rod fuel sections and filler pieces in rows 5 through 9. It is true that the

x,y distribution along the entire column cannot be expressed in separable form as in Eq. (1). However, it is only in the source volumes close to where the heat rates are desired that it is important to know the distribution. Therefore, X(x) and Y(y) were chosen to be appropriate to the fuel along row 7 where the heat rates were needed. The inadequacy of this distribution in the fuel in other rows should not cause serious errors. The method used takes account of the relative powers in the fuel bearing locations, and the fact that no power is generated in the filler pieces.

5.1.3 Average ETR power density

All of the calculations in this work use sources of gamma photons, beta particles and fission heat which are based on an average $\bar{p}$ for the ETR operating at 175 MW. In calculating the fraction of the gamma energy deposited in the core, a model was used in which the 175 MW was assumed to be released in 74 fuel positions. This comprises all the core volume excluding the aluminum fillers on the core boundary and the FEFPL loop. Then

$$\bar{p} = \frac{175 \times 10^6}{74 \times V_{\text{fuel}}} = 435 \text{ W/cc}$$

$$\begin{aligned} V_{\text{fuel}} &= \text{volume of one fuel element position} \\ &= 36 \times (3.034)^2 = 331 \text{ in}^3 \\ &= 5430 \text{ cm}^3 \end{aligned}$$

In the other applications, the 175 MW was dissipated in 52 fuel elements. In this case

$$\bar{p} = 621 \text{ W/cc}$$

The manner in which $\bar{p}$ is adjusted for generating the sources used in the gamma ray leakage, the gamma heating within the core and the beta and fission heating will be presented when these topics are considered.

5.2 Calculation of Gamma Ray Fluxes at a Detector Location

5.2.1 The photon sources

To calculate the flux of gamma rays of any energy at a particular core location, it is first necessary to know the photon source distribution. Within the core the photons produced by decay of fission products exceed by a large amount those which arise from neutron activation of structural material. The fission product emitters are produced directly from fission, or are descendants of isotopes produced from fission. In either case, the source is distributed like the power. This means Eq. (1) can be applied. In the application made here the average power density, $\bar{p}$ must be replaced by quantities $\bar{p}_j$, which are the average photons/cc sec in energy group j produced in the core by fission product decay.

$$p_j(x,y,z) = \bar{p}_j X(x)Y(y)Z(z) \quad (4)$$

p_j = photons/cc sec for photons in energy group j at location x,y,z

$\bar{p}_j$ = average photons/cc sec for photons in energy group j in the entire volume under consideration.

The decay of primary fission products, and the build-up and decay of daughter products throughout the operating history of the core is a complex problem - particularly if neutron capture by the fission products is accounted for. In this work, it is assumed that $X(x)$, $Y(y)$ and $Z(z)$ remain unchanged during the operation and the total power is maintained at 175 MW.

The calculation of $\bar{p}_j$ for any time during operation, and subsequent to shutdown, was performed by the computer codes ISOSHL⁽⁵⁾ and SINBAD⁽⁴⁾. These codes contain libraries of fission products and fission yields, decay chains, decay constants, capture cross sections and photon and beta energy releases. For any value of the average power $\bar{p}$ it is possible with these codes to calculate $\bar{p}_j$ at shutdown, and at any time subsequent to shutdown. Table VII gives data calculated from ISOSHL for 25 neutron groups assuming

operation of $\bar{p} = 435$ W/cc for 45.7 days prior to shutdown. Table VIII gives results from SINBAD for the same operating period but using $\bar{p} = 621$ W/cc. Table VIII also gives data in β decay.

The SINBAD results are in rather good accord with the ISOSHL D results corrected to $\bar{p} = 621$ W/cc for times greater than about one minute. The SINBAD results are higher near $t = 0$ because the SINBAD library has been adjusted by including "dummy" short lived isotopes which give the correct value for the total energy release immediately following shutdown.* The ISOSHL D code is deficient at these times because the ISOSHL D library is based on experimental data and these are lacking for the short lived isotopes. SINBAD has isotopes emitting up to 5.5 MeV and ISOSHL D up to 3 MeV. Hence, the SINBAD sources may better represent high energy photon production.

The data in Tables VII and VIII are used as input to the QAD code which calculates fluxes from these sources, and energy absorption in materials exposed to these fluxes.

5.2.2 The point kernel method for calculating gamma ray fluxes

Once the source distribution of photons in each energy group is known, it is possible to calculate the flux of gammas in each energy range at any desired location within or external to the core. The most accurate method is to use a Monte Carlo calculation which follows the histories of photons which contribute to the flux at any point. Another method is to solve the transport equation. The Monte Carlo method would be very expensive in a core with geometry as complex as the ETR. A transport calculation would have to be used in at least two dimensions, and probably all three, and also would be very expensive. In addition, a complicated analysis of the energy transfer mechanisms accompanying photon-nucleon interaction would be required to obtain correct transfer cross sections.

* G. C. Putnam achieved this by adding five isotopes each with half lives of 5 seconds which yield gamma rays of 0.3, 0.7, 1.3, 2.0 and 4.5 MeV. The fission yields are 2.07, 1.66, 0.408, 0.581 and 0.0357, respectively.

Since these methods are unrealistic for dealing with most shielding and gamma heating calculations, they are generally circumvented by an approximation called the "point kernel" method. The procedure assumes that the photons do not transfer in energy from group to group during scattering collisions. The undirected flux at a point $\underline{r}$ due to photons produced at $\underline{r}'$ is calculated by multiplying the uncollided flux by a "build-up" factor which depends on the scattering properties along the flight path from $\underline{r}'$ to $\underline{r}$. These ideas can be explained best using the fundamental equation

$$\phi(E, \underline{r}) = \int \frac{S(E, \underline{r}') B[\mu(E, |\underline{r} - \underline{r}'|)] e^{-\mu(E, |\underline{r} - \underline{r}'|)}}{4\pi |\underline{r} - \underline{r}'|^2} d\underline{r}' \quad (5)$$

In Eq. (5)

$\phi(E, \underline{r})$ = photons/cm²-sec of energy E at position $\underline{r}$.

This is an undirected flux.

$S(E, \underline{r}')$ = photons/cm²-sec of energy E at position $\underline{r}'$.

$$\mu(E, |\underline{r} - \underline{r}'|) = \int_{\underline{r}}^{\underline{r}'} \mu(E, \underline{r}'') d\underline{r}'' ; \text{ (along the direct flight path from } \underline{r}' \text{ to } \underline{r})$$

$\mu(E, \underline{r}'')$ = total attenuation coefficient for photons of energy E associated with the material composition at $\underline{r}''$. Units are cm⁻¹.

If the only photons which reached $\underline{r}$ from $\underline{r}'$ were those which had suffered no collisions along the direct flight path, $\phi(E, \underline{r})$ would be calculated by putting $B = 1$. However, an important contribution to the flux at $\underline{r}$ arises from photons which, after being deflected by a primary scattering event, are rescattered back to $\underline{r}$ by secondary or higher order scattering events. For a pure material the build-up factor $B[\mu(E, |\underline{r} - \underline{r}'|)]$ can be developed from an analysis of experimental data so that it increases the uncollided flux by the right amount. The build-up factor for the pure material is a function of the number of relaxation lengths between $\underline{r}'$ and $\underline{r}$.

When a mixture of materials is present, the number of relaxation lengths from $\underline{r}'$ to $\underline{r}$ in the mixture is calculated and the build-up factor for some relevant pure material experiencing the same number of relaxation lengths is used. For example, in the present calculations the computer code calculates the number of relaxation lengths in each region between the source and detector. Then it uses for the build-up factor the number appropriate to pure aluminum, or pure water, for this number of relaxation lengths. Most calculations in this work use the build-up factor for aluminum.

For a number of reasons the point kernel method cannot be highly accurate for a system with regions of different composition. It takes account of the scattering properties (number of relaxation lengths) only along the direct flight path. If rays traversing other directions not much different from this travel through media with markedly different attenuation properties, the build-up factor may not take adequate account of the changes. Again, since the build-up factor is determined by finding the number of relaxation lengths in the actual medium, and then transposing them to a pure material to permit using the build-up factor for the material, errors can occur if a poor choice is made for the pure material. Finally, the point kernel method takes no account of scattering processes which can cause a photon to alter in energy from one energy group to a lower energy group.

Nevertheless, the point kernel method is a basic tool for shielding design and for calculating gamma heating. Once the flux at $\underline{r}$ is calculated, it is possible to obtain the rate of heat generation in any material at $\underline{r}$. For the calculations in this work, it is assumed that local depressions in the gamma flux within lumped absorbers of high density are unimportant. The heating rate H_i in material i is calculated from

$$H_i = \sum_j \mu_i(E_j) \phi(E_j, \underline{r}) E_j \times 1.603 \times 10^{-13} \quad (6)$$

(watts/cc)

$\mu_i(E_j)$ = energy absorption cross section of material i for
photons of energy E_j . (cm^{-1})

$1.603 \times 10^{-13} = \text{joules/MeV}$

E_j = photon energy, (MeV)

$\phi(E_j, \vec{r})$ = flux of photons of energy E_j at position $\vec{r}$.
(photons/cm² sec)

5.2.3 The QAD point kernel code

The QAD code calculates Eq. (5) using the relation

$$\phi(E_j, \vec{r}) = \sum_m \frac{S_m(E_j)}{4\pi R_m^2} \left[\sum_k B_{jk} \exp\{-(\mu_j t)_k\} \right] \quad (7)$$

$S_m(E_j)$ = photons/sec of energy E_j produced in a volume element m . (In QAD the entire source associated with a volume element is placed at the geometrical center of the element.)

R_m = direct flight distance from the center of volume element m to the point $\vec{r}$.

$(\mu_j t)_k$ = relaxation length for photons of energy E_j experiencing a flight path of length t in region k which is characterized by a specific composition.

B_{jk} = build-up factor for region k for photons of energy E_j . The B_{jk} is computed in the code from an equation involving E_j and $(\mu_j t)_k$, the number of relaxation lengths.

The QAD code also provides for inputting $\mu_k(E_j)$ and, from this, calculating the sum in Eq. (7).

In using QAD it is necessary to define regions with different material compositions, and volume elements for the source subdivision. The two are unrelated. Each region has a unique composition. In the present work, rectangular geometry is involved and the regions are defined for the compositions in Table V. These are for fuel elements, control rod fuel sections, SFR filler pieces (four kinds) and the FEFPL loop.

The source volumes are defined separately from the regions. Data are supplied from which the code calculates $X(x)$, $Y(y)$ and $Z(z)$ in Eq. 4. The size of a source volume and the value of $X(x)$, $Y(y)$, $Z(z)$ at the center of the volume will determine the value of $S_m(E_j)$ in Eq. (7). It is necessary to have the source volumes small in the vicinity of the detector points. For sources far from a detector the volumes can be larger. This is because the $1/R_m^2$ term changes rapidly with distance when R_m is small.

5.3 Calculation of Gamma Ray Leakage

5.3.1 The source distribution for the leakage calculation

The QAD code requires the source distribution to be expressible using Eq. (4). This is not valid for the entire core because the x, y distribution is not at all separable either for fission product or activation gammas. However, if the core is divided into suitable small volumes, Eq. (4) can be used for the local regions. To do a complete calculation of gamma heating separate calculations must be run for each of the local source regions and the results must be added by hand. (The option in QAD which would accomplish this is not yet operable on our 360/75.)

This is not a very practical procedure to follow in calculating gamma ray leakage. Not only is it cumbersome to set up this multi-source problem, but the added accuracy over simpler formulations may be small or even insignificant. This is partly because of the point kernel approximation which does not give a true directional photon flux. As a result, the method of inferring leakage is itself an approximation beyond some of those already implicit in the point kernel method.

To produce a simple computational method capable of yielding a realistic result, an artificial model was assumed for which Eq. (4) applies over the entire core volume. The QAD code can then treat the sources over the entire core volume in one calculation. To accomplish this, all core materials, including the FEFPL loop but excluding the 22 aluminum filler pieces which are along the core boundary, was homogenized. Furthermore, the volume was depicted as a cuboid 36" high of square cross section equal to that of 78 core positions (718 in²). The 178 MW power was expressed by Eq. (1) with $X(x) = Y(y)$ and $Z(z)$ represented as chopped cosine functions. These were chosen such that the values at the boundary were reduced to $1/\sqrt{2} = .71$ of the values at the central maximum. Equation (4) for the photon sources follows the same distribution. The functions are

$$\begin{aligned} X(x) &= \frac{\pi}{2\sqrt{2}} \cos\left(\frac{\pi}{2} \frac{x}{\hat{X}}\right) \\ Y(y) &= \frac{\pi}{2\sqrt{2}} \cos\left(\frac{\pi}{2} \frac{y}{\hat{Y}}\right) \\ Z(z) &= \frac{\pi}{2\sqrt{2}} \cos\left(\frac{\pi}{2} \frac{z}{\hat{Z}}\right) \end{aligned} \tag{8}$$

$$\begin{aligned} \hat{X} &= \hat{Y} = 68.06 \text{ cm} \\ \hat{Z} &= 91.44 \text{ cm} \end{aligned}$$

In Eq. (8) the origin is taken at the center and the normalizing factors assures that Eqs. (3) apply when adapted to this origin.

The QAD code carries out these calculations internally using simple geometric input with parameters chosen so the required chopped cosine function is calculated.

5.3.3 Method for calculating gamma leakage

There are at least two ways to calculate the fraction of the total gamma ray energy which leaks out of the core or, conversely, is deposited within the core. Neither is free of approximations which are in addition to those inherent in the point kernel method.

One procedure is to divide the core into volume elements of uniform composition and to calculate the heat generation rate within each element. The sum over the core then gives the total heat generation rate. The ratio of this to the total rate of energy release in gamma photon production from fission product decay would yield an estimate of the fraction of the gamma energy deposited in the core. However, for this result to be accurate the point kernel fluxes must be reasonably accurate, the energy absorption cross sections must be correct and rapid spatial variations in gamma flux must be treated adequately.

A preferable procedure is to discard the attempt to calculate the energy dissipation inside the core in favor of estimating the fraction of photons in each energy group which escape. A ratio method is used in which the photon energy flux in each energy group is calculated at some external point, first assuming the actual homogenized core materials attenuate the flux and, second, calculating the flux when no attenuation is present. The assumption is that the ratio of the first to the second gives the fraction of photon energy in each energy group which escapes. From this the fraction of photon energy which stays in the core can be computed.

This method would be correct if the calculated fluxes were correct, and could be interpreted as being directed outward from the core. This is true for the case where no attenuation is present. When attenuation is present it is subject to the approximations of the point kernel method. However, the method does not involve the further approximation of calculating the heat generation which is required in the method described first, and it should be less sensitive to rapid changes in source distribution within the core.

5.3.4 Gamma ray leakage results

The flux ratio computed in this way depends somewhat on the location of the external point where the fluxes are evaluated. Therefore, in this work, a number of external points were chosen and an average of the ratios has been used.

The core was divided into 1500 source volumes of equal size. There were 10 units along x and y of 6.8 cm width and 15 units along z of 6.10 cm width. The chopped cosine distribution expressed in Eq. (8) was evaluated within the QAD code.

Table IX gives the fluxes with and without attenuation, and the ratio, for selected points external to the core. The data are referred to axes at the core center. These results use the build-up factor for aluminum and the attenuation properties for the fully homogenized core (Table VI). Some calculations were made, also, with the H₂O build-up factor and gave ratios very close to those in Table IX.

The average ratio in Table IX is 0.25. Actually, calculations were made for 25 points and gave an average of 0.24. It is reasonable to infer that 25 percent of the fission gamma energy produced just after shutdown escapes the core and 75 percent is deposited in the core.

This procedure uses a simple model for the core in which all materials are homogenized and the fission source is distributed in a very simple manner. It is unlikely that an elaborate point kernel calculation which takes account of the actual core structure and power distribution would be much more meaningful. The same approximate procedure would be required. Again, as a photon traverses the core to its final external target point, the variation in relaxation properties it experiences would tend to average out to be something like that for the homogenized material. Similarly, the use of the actual power distribution might not gain much in accuracy because a detector located externally won't be too sensitive to the exact spot in which the photons originated within a source volume element.

The calculations were made for fission gammas produced just after shutdown. For longer time after shutdown the fission gamma spectrum becomes somewhat softer and the fraction of energy released within the core might exceed the 75% calculated just after shutdown. However, for methods as qualitative as these, it seems appropriate to adopt 75% core deposition as reasonable for all times after shutdown,

5.4 Calculation of Gamma Heating Near the Hot Spot

5.4.1 Location of the hot spot

The fuel element in position 7I was identified as sustaining the highest heating rates from gamma rays, beta particles and fissions induced by delayed neutrons. This is because the measurements in Figure 1 show that during the 45.7 days operating time, this position carries the highest fraction of the total power (3.08%) of any fuel position and the powers carried by the fuel elements with faces or edges touching 7I add up to more than for the powers which surround any other element in the core.

Since 7I has the highest power it must experience the highest beta heating because the betas are retained in the fuel. Since it is surrounded by more power than any other element it will experience a higher gamma ray flux, and it will tend to be exposed to more delayed neutrons which can induce fissions within it. Therefore, it is reasonable to assume that the distribution of heat generation within and near 7I will show higher values than for any other core location.

5.4.2 The source distribution for calculating gamma heating in and near the hot fuel element

Since position 7I will experience the highest gamma fluxes, the spatial distribution of gamma heat generation was computed at selected points along the mid lines of row 7, and of column I, in the vertical (axial) plane of highest source density. The gamma flux at any point on these traverses will be sensitive to local variations in the source density near the point but insensitive to local variations some distance away. For this reason, it is necessary to describe the source variation relatively accurately only in limited regions near to the places where the heating is desired. Since attenuation reduces contributions to the gamma heating from photons originating far from 7I and its environs, the sources can be restricted to less than the full core. Figure 16 shows the region comprising columns G through K and rows 5 through 9 which was deemed adequate for defining the sources needed in calculating the gamma heating around 7I.

Section 5.1.2, part c indicates how a column of fuel can be treated as a gamma source with a distribution expressed by Eq. (4). The row distribution, $X(x)$, and the column distribution, $Y(y)$, need to be realistic only in the fuel close to the points where the gamma heating rates are desired.

In this work, $X(x)$ and $Y(y)$ were chosen to be realistic in the fuel in 7H, 7I and 7J. For other rows in columns H, I and J the $X(x)$ used in row 7 was employed but $Y(y)$ was taken to be flat. Both $X(x)$ and $Y(y)$ were flat for fuel in all rows of columns G and K.

The mathematical form for Eq. (4) is expressed as follows. Let

$\bar{p}_j$ = average photons/cc sec for group j in the entire core.
(52 fuel elements in this case.)

F_{nI} = fraction of the total power carried by the fuel in row n and column I .

$\bar{p}_{nI,j}$ = average photons/cc sec in group j at core location nI .

$$\bar{p}_{nI,j} = \bar{p}_j \frac{V}{V_E} F_{nI}$$

V = volume of core (cm^3)

V_E = volume of a fuel location (cm^3)

$$[p(xyz)]_{nI,j} = \bar{p}_{nI,j} X_n(x) Y_I(y) Z(z) \quad (9)$$

$$\int_0^{X_E} X_n(x) dx = X_E \quad (10a)$$

$$\int_0^{5X_E} Y_I(y) dy = 5X_E \quad (10b)$$

$$\int_0^{\hat{Z}} z(z) dz = \hat{Z} \quad (10c)$$

In the above

X_E = distance across one fuel position (7.706 cm)

$5X_E$ = length of a column of fuel five core positions long
(38.53 cm)

$\hat{Z}$ = axial length of a fuel element (91.44 cm)

In this gamma heat calculation the same distribution was used for $X(x)$ and $Y(y)$ in fuel positions 7H, 7I and 7J. It was obtained by constructing a "symmetrized" distribution based on examining ETRC fission wire activation data which traversed fuel element 9J in two directions⁽¹⁾. The result is shown in Figure 17.

This is an adequate, although inexact, distribution for the control rod fuel section 7H and 7J. In Section 5.5.1 more refined distributions are developed for calculating the beta and fission heat distributions.

Along each column the over-all $Y(y)$ was determined in accordance with Eq. (10b). The distribution is proportional to the power in each fuel location and varies locally in row 7 according to Figure 17. The $Y(y)$ is flat in the fuel in other rows.

The distribution for each column G through K are shown in Table X.

5.4.3 Source volumes for calculating gamma heating around the hot spot

In Section 5.4.2 the spatial distribution of the gamma sources was discussed. Equation (7) requires integrating these spatially distributed sources over volume elements. These volume elements must be fine grained where the source distribution changes rapidly with position if advantage is to be taken of the detailed source density variation in space.

The source volumes used in the QAD calculations were different for each column. In every case, however, the vertical (z) direction was divided into 10 equal slices 9.144 cm long. (In retrospect, it would have been an improvement to take smaller segments near the plane of highest source density).

In columns G and K the spacing along the row direction, x, was one fuel element width wide. Along the column direction y there were five segments equal in width to one fuel element.

In columns H, I and J the width in the row direction was divided into six segments as shown in Table XI. Along the column direction the widths in rows 5, 6, 8 and 9 was one fuel element distance (7.706 cm) but across row 7 the spacing was divided into six parts in the same way as along x. Thus the source volumes close to the points where heat rates are to be calculated are quite small. These are the regions, also, where $\chi(x)$ and $Y(y)$ vary rapidly. In columns G and K, the volume units are much larger. This information is displayed in Table XI which is based on the distributions in Table X.

5.4.4 Gamma heat results

Table XII lists the energy absorption coefficients μ_{ij} (cm^{-1}) for aluminum, H_2O fuel "meat" and cadmium for each of the 22 energy groups used in the calculation. The cadmium is included for use in the canal storage problem. The conversion factors which are used in QAD to calculate the energy fluxes are also given.

The gamma heat generation rate in watts/cc in the fuel "meat" is shown in Tables I and II along with data on beta and fission product heating to be discussed later. The data show the heating at selected points along the centerline of row 7 and column I in the plane of highest source density. Plots of the gamma heating rates are presented in Figures 2-7. These include heating in water and aluminum.

5.5 Calculation of Beta Heat Generation Rate

5.5.1 The beta source distribution

The beta particles have very short ranges and a conservative assumption is that they release all their energy in the fuel "meat". Therefore, the heat production due to beta particles is distributed like the power.

A more elaborate analysis was used in defining the power distributions used for determining the beta sources than was used in the gamma heating calculations. This reflects improvements which evolved during the calculations. The same ETRC measurements and computational results were used as for the source distributions used in calculating the gamma heating. However, the re-analysis yielded new distributions $X(x)$ and $Y(y)$ within 7I, and distributions in the control fuel sections which conform more realistically to the fuel distributions in these positions. These refinements are more important for the beta heating than for the gamma heating.

The row distribution $X(x)$ deduced for 7I may be more realistic than for the column direction, $Y(y)$. This is because the row-wise environment of 9J is more like that for 7I than for the column-wise direction. However, the measurements in 9J are consistent with using a relatively flat distribution in the y direction than for x .

The distributions in the control fuel sections were chosen to conform to the computational results of Rushton⁽³⁾. They take better account of the distribution of fuel within the control fuel sections.

These distributions should give realistic ratios for the power densities at the peak and valley positions within 7I and the control fuel sections. Beyond this, inaccuracies remain because the data for 9J are for an environment somewhat different from 7I. In addition, the computational results used to support the 7I analysis, and to define the distributions in 7H and 7J, are not for the same environment. Also, the methods have not been tested fully against experiment. Because of this the distributions $X(x)$ and $Y(y)$ used in these positions have been "symmetrized" so they are symmetrical for reflection in the mid-lines of the fuel and control rod fuel sections.

The row-wise distribution function for position 7I is shown in Figure 18 and the column-wise distribution in Figure 19. The distribution used for both row and column directions in 7H and 7J is shown in Figure 20.

5.5.2 The beta heat generation

The SINBAD code provides $\bar{p}_\beta(t)$, the average power density due to beta release in the entire core as a function of time after shutdown. This must be corrected by the ratio of fuel element volume to fuel meat volume (or control fuel section volume to meat volume in the control fuel section) to take account of the localization of this energy within the fuel meat. This ratio is 9.87 for a fuel element and 14.7 for a control rod fuel section.

Figures 8 and 9 show the power distributions due to beta energy release in fuel across the element in 7I in the axial (z) plane of highest source density. The traverses are along peaks at the edges of the fuel element where the power density is highest (due to power peaking at the edges of the assembly).

The equations for the beta heating distribution through 7I in the axial plane of highest source density are

$$p_{7I}(x,t) = 9.87 \bar{p}_\beta(t) F_{7I} \frac{V}{V_E} Y_{\max}(y) Z_{\max}(z) X(x) \quad (11a)$$

$$p_{7I}(y,t) = 9.87 \bar{p}_\beta(t) F_{7I} \frac{V}{V_E} X_{\max}(x) Z_{\max}(z) Y(y) \quad (11b)$$

9.87 = concentration factor = volume of a fuel element divided
by the "meat" volume

using

$$F_{7I} = .0308$$

$$\frac{V}{V_E} = 52$$

$$X_{\max}(x) = 1.25$$

$$Y_{\max}(y) = 1.13$$

$$Z_{\max}(z) = 1.23$$

$$p_{7I}(x,t) = 22.0 \bar{p}_\beta(t)X(x) \quad (12a)$$

$$p_{7I}(y,t) = 24.3 \bar{p}_\beta(t)Y(y) \quad (12b)$$

$X(x)$ from Figure 18. $Y(y)$ from Figure 19.

The plots have not been given for the power distribution in the control rod fuel section. They can be constructed using

$$.0149 = F_{7H}, \quad .0173 = F_{7K}$$

$$\begin{matrix} X(x) & = & Y(y) & = & 1.05 \\ \text{max} & & \text{max} & & \end{matrix}$$

The result is

$$p_{7H}(x,t) = p_{7H}(y,t) = 14.7 \bar{p}_\beta(t)X(x) \quad (13a)$$

$$p_{7K}(x,t) = p_{7K}(y,t) = 17.1 \bar{p}_\beta(t)X(x) \quad (13b)$$

$$X(x) = Y(y) \text{ from Figure 20 .}$$

5.6 Calculation of Fission Heat Due to Delayed Neutrons

5.6.1 The delayed neutron source

When the control rods are suddenly inserted in an operating reactor the power drops almost instantly to a level determined by the effective multiplication of the system. Fissioning continues because delayed neutrons continue to feed into the system and act as a source in a multiplying medium.

In this work a simple one neutron group analysis was made to obtain an equation for the fission rate subsequent to shutdown. The method reduces to the prompt drop equation for one delayed neutron group at $t = 0$. A 16% scram worth is assumed in the calculations.

Assume that the effect of the control rods can be simulated by distributing a poison in the core such that

$$\sum_p(r) = \gamma \sum_a(o,r)$$

$$\sum_p(r) = \text{poison added at } r \text{ (cm}^{-1}\text{)}$$

$$\sum_a(o,r) = \text{macroscopic absorption cross section at shutdown} \\ (t=0) \text{ at position } r.$$

Before shutdown equilibrium conditions prevail and

$$0 \simeq v \int_c \Sigma_f \varphi_o dr - \int_c \Sigma_a \varphi_o dr - f \int_c v \Sigma_f \varphi_o dr \quad (14)$$

$\varphi_o(r)$ = flux at r just before shutdown,

f = fraction of neutron production which leaks out.

The integral is over the core volume.

After shutdown, assuming f is unchanged,

$$\begin{aligned} \frac{1}{v} \dot{\int_c \varphi(r,t) dr} &= (1-f)(1-\beta) v \int_c \Sigma_f(r,v) \varphi(r,t) dr \\ &\quad - (1+\gamma) \int_c \Sigma_a(r,v) \varphi(r,t) dr \\ &\quad - (1-f) v \int_c \Sigma_f(r,o) \varphi(r,o) dr \left\{ \Sigma \beta_i \left| c^{-\lambda_i t} \right. \right\} \end{aligned} \quad (15)$$

The first term on the right side is the production minus the leakage of prompt neutrons. This is where the $1-\beta$ factor comes in, β being the fraction of the neutrons produced which come out delayed. The second term is the absorption rate and the last term is the production minus leakage of neutrons which are emitted from the delayed neutron emitters which were present prior to the rod drop. A second order term arising from emission of delayed emitters produced after $t=0$ is omitted.

If it is assumed that, independent of r ,

$$\varphi(r,t) \simeq K(t) \varphi(r,o)$$

the above equation can be simplified to the form

$$\begin{aligned} \int \frac{\dot{\varphi}}{v}(\underline{r}, t) d\underline{r} = & -K(t) \left\{ (1-f) v \beta \int \Sigma_f(\underline{r}, 0) \varphi(\underline{r}, 0) d\underline{r} \right. \\ & \left. + \gamma \int \Sigma_a(\underline{r}, 0) \varphi(\underline{r}, 0) d\underline{r} \right\} \\ & + (1-f) v \int \Sigma_f(\underline{r}, 0) \varphi(\underline{r}, 0) d\underline{r} \sum \beta_i e^{-\lambda_i t} \end{aligned} \quad (16)$$

Suppose that at any time t the delayed emitters continue to yield neutrons, but do so at a steady rate. In other words, assume $\dot{\varphi}(\underline{r}, t) = 0$. Then

$$K(t) = \frac{\varphi(\underline{r}, t)}{\varphi(\underline{r}, 0)} = \frac{p_F(t)}{p(0)} = \frac{\sum \beta_i e^{-\lambda_i t}}{\beta + \gamma / (1-f)} \quad (17)$$

This equation is similar to Eq. 5.30 of Glasstone and Sesonke (Ref. 8) under the restrictions

$$\rho = -\gamma$$

$$|\rho| \gg \beta$$

$$f = 0$$

An effective one group decay constant λ is used in Glasstone-Sesonke.

A more elaborate analysis can be made by using

$$\dot{\varphi}(\underline{r}, t) = \dot{K} \varphi(\underline{r}, 0) \quad .$$

This yields

$$\dot{K} = -K(t)B_1 + B_2 \Sigma \beta_i e^{-\lambda_i t}$$

$$B_1 = v \left[(1-f) v \beta \bar{\Sigma}_f + \gamma \bar{\Sigma}_a \right]$$

$$B_2 = v(1-f) v \bar{\Sigma}_f$$

This yields Eq.(17) under the restriction

$$B_1 \gg \lambda_i$$

5.6.2 Fission heating results

In these calculations results are given for $f = 0$ since this gives a conservative (large) prediction of shutdown power. In addition, the power distributions in 7I were assumed to be the same as given by Eqs. (11, 12) with $\bar{p}_\beta(t)$ replaced by the average fission power density after shutdown, $\bar{p}_F(t)$

$$\bar{p}_F(t) = K(t)\bar{p}(o)$$

Here $\bar{p}(o)$ is the average power density in the operating reactor. Figure 21 shows the function $\bar{p}_F(t)/\bar{p}_F(o)$. A value of $\gamma = 0.16$ was used in Eq. (17) since this corresponds with typical shutdown reactivities for ETR⁽¹⁰⁾.

Complete results are given in Tables I, II and Figures 10 and 11. These results are conservative because they assume $f=0$ and take no account of perturbations in the flux in 7I arising because the adjacent control rods have been dropped. Some account could be taken of the effect of adjacent control poison but a rather elaborate extension of this analysis would be needed. It is assumed that the present, conservative, procedure is adapted to safety analysis purposes.

5.7 Basis for Neglect of Activation Gammas

To assess whether or not gammas resulting from decay of isotopes activated by neutrons are a significant source relative to the fission gammas,

an analysis was made assuming an average thermal flux of 2.5×10^{14} n/cm² sec. This flux is appropriate for an average power of 621 W/cc. It was estimated assuming the fuel in the 38 fuel elements and 14 control fuel sections was homogenized into a volume of 52 assemblies. A g factor of 0.95 for the microscopic fission cross section was taken from Figure 2.16 of Reference 8. A cross section of 486b based on a U²³⁵/H atom ratio of .005 was inferred from Figure 2.17 of Reference 8.

The most potent set of activated isotopes were selected by examining the data in Reference 9. In doing this, the alloy composition of the stainless steels and aluminum were considered. The equation used for estimating the activation sources is

$$\lambda_i N_i(t) = \sigma_{i-1} N_{i-1} \phi \left[1 - e^{-\lambda_i t_{op}} \right] e^{-\lambda_i t_s} \quad (18)$$

ϕ = flux (2.5×10^{14} n/cm² sec)

N_i = atoms/cc of species i

λ_i = decay constant for species i (sec⁻¹)

t_{op} = operating time (sec)

t_s = shutdown time (sec)

σ_{i-1} = calculation cross section for species i-1 whose capture leads to species i. (barns)

The rate of energy productions is

$\lambda_i N_i E_i$ where E_i = energy of photon released in decay of species i.

Table XIII summarizes the results. The energy production rate from this source is small enough compared to the fission product source that it can be neglected.

6.0 GAMMA AND BETA HEATING IN THE DRY CANAL STORAGE RACK

6.1 Gamma Heating Rate

The gamma heating rate in the cadmium was particularly important because loss of cadmium on melting would create a hazard when the dry canal was reflooded.

The gamma heating at the center of a 7 x 7 array in the plane of highest source density was calculated. The source was assumed to be flat in the horizontal (x,y) plane but to have the same vertical variation $Z(z)$ as shown in Figure 14. The $\bar{p}_j(t)$ values were appropriate to the average power $\bar{p}$ at 175 MW in 52 fuel positions. The structural materials in the rack were homogenized with the fuel materials. Figure 22 displays the simple 7 x 7 configuration considered and Figure 23 gives the cross sectional detail of one cell.

Figure 24 gives the highest heating rate in cadmium and aluminum due to gamma ray absorption for times from 50 to 200 days after shutdown following 45.7 days operation at 175 MW.

6.2 Beta Heating Rates

The maximum beta heating rates will occur in the fuel element which carried the highest power during operation. This is the element in position 71. Figure 25 shows the variation in the maximum heating due to beta emission in this element. It is computed from adapting Eq. (12a) to the form

$$p_{\beta\max} = 27.5 \bar{p}_{\beta}(t) \quad . \quad (19)$$

In Eq. (19) $\bar{p}_{\beta}$ was obtained from SINBAD for times varying from 50 to 200 days after shutdown.

7.0 RECALCULATION FOR ALTERED POWER DISTRIBUTIONS

The analysis reported here does not have the benefit of the final ETRC measurements. These will give much more detail regarding power distributions around the hot spot. In addition, the possibility exists that experiments in addition to FEFPL may be inserted in the core and this could alter the power distribution significantly.

The question arises as to whether the results from this analysis could be used to revise shutdown heat production when more accurate power distributions are available. In addition, the more general question of how to recalculate when large changes in power distribution are introduced needs to be examined.

For calculating beta and fission heating rates, the answer is that recalculation is easy. It is simply necessary to use new experimental data to construct $X(x)$, $Y(y)$ and $Z(z)$ in any local region of interest. This, together with appropriate values of the F_{nI} is all that is needed, for the $\bar{p}_\beta(t)$ can be obtained from Table VIII with proper adjustment for any changes in average power density. If decay heats are needed at times other than those in Table VIII, a curve can be drawn.

For the fission heat the same power distributions are used and Eq. (17) is employed.

New gamma heating distribution cannot be calculated so easily. If the power distribution changes are small and the core configuration is not changed significantly, the present results will apply (possibly adjusted to take account of changes in $\bar{p}$). If large changes in power distribution and/or core configurations are made, new calculations will be needed. However, the data in Tables VIII and XII will be very helpful. In practice, Table VIII with proper adjustments for changes in $\bar{p}$, provides the input to the QAD code. A considerable effort in setting up regions, defining sources, etc., would still be required.

The above statements assume 45.7 d operation. If the operating time is changed, new β and γ sources must be computed - Table XV no longer applies.

8.0 ACCURACY

The SINBAD code from which the source terms were developed, generally used the same basic data as the ANS.5 Standard. However, for engineering purposes, the calculated decay heats should be increased by 20% to allow for uncertainty in the that basic data, and for errors introduced by approximations in the calculations.

The fission product data are very complex. There are many fission products and accurate calculations of gamma and beta sources requires very detailed data on yields and decay chains. The decay processes are often complicated by branching, by complex decay processes associated with gamma emissions and by emissions of beta particles which are generally, but not always, in continuous energy distributions. The relevant data are not always known with high accuracy and even the most elaborate computer codes may not deal correctly with every process which occurs.

After the sources are determined the gamma ray heating involves additional approximations associated with the point kernel method and with assumptions regarding the gamma ray energy group structures. It depends, furthermore, on the having accurate data on energy absorption coefficients.

These are the main reasons for adding 20% to the calculated energy releases.

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TABLE I
Heating Rates in Fuel Meat Along Row 7 in Column I (a,b)

	0				6 sec				1 min.				10 min.				1 hr			
f(c)	γ	β	F	Total	γ	β	F	Total	γ	β	F	Total	γ	β	F	Total	γ	β	F	Total
0.0																				
.1	44	705	695	1444	29	523	264	816	16	344	36	396	12	186		198	8	103		111
.2	47	580	575	1202	30	430	220	680	17	285	30	332	12	153		165	8	85		93
.3	49	542	535	1126	31	402	203	636	17	265	28	310	13	143		156	9	80		89
.4	50	532	532	1120	32	400	202	634	18	263	28	309	13	142		155	9	79		88
.5	50	538	532	1120	32	400	202	634	18	263	28	309	13	142		155	9	79		88
.6	50	538	532	1120	32	400	202	634	18	263	28	309	13	142		155	9	79		88
.7	49	542	535	1126	31	402	203	636	17	265	28	310	13	143		156	9	80		89
.8	48	580	575	1203	30	430	220	680	17	285	30	332	13	153		166	9	85		94
.9	46	705	695	1446	29	523	264	816	16	344	36	396	12	186		198	8	103		111
1.0																				

(a) All rates watts/cc.

(b) Beta and fission heating rates are calculated for the fuel plate at the edge of the assembly. This plate has the highest column-wise peak to average source density. The gamma heating is along the centerline of row 7. Transposing to the edge plate makes minor errors.

(c) Fraction of distance across column I.

TABLE II

Heating Rates in Fuel Meat Along Column I in Row 7^(a,b)

	0				6 sec				1 min.				10 min.				1 hr			
f ^(c)	γ	β	F	Total	γ	β	F	Total	γ	β	F	Total	γ	β	F	Total	γ	β	F	Total
0.0	38	705	698	1441	24	524	264	812	14	345	36	395	10	187		197	7	103		110
.1	44	663	650	1357	28	493	247	768	15	321	34	370	11	175		186	8	96		104
.2	47	630	620	1297	30	469	235	734	16	306	32	354	12	166		178	8	92		100
.3	49	608	600	1257	31	450	228	709	17	295	31	343	13	160		173	9	88		97
.4	50	593	585	1228	32	440	224	696	17	290	30	337	13	156		169	9	86		95
.5	50	586	578	1214	32	435	223	690	18	286	30	334	14	155		169	9	85		94
.6	50	593	585	1228	32	440	224	696	17	290	30	337	13	156		169	9	86		95
.7	49	608	600	1257	31	450	228	709	17	295	31	343	13	160		173	9	88		97
.8	47	630	620	1297	30	469	235	734	16	306	32	354	12	166		178	8	92		100
.9	44	663	650	1357	28	493	247	768	15	321	34	370	11	175		186	8	96		104
1.0	38	705	698	1441	24	524	264	812	14	345	36	395	10	187		197	7	103		110

(a) All rates in watts/cc.

(b) Beta and fission heating are calculated along the tips of the fuel plates where the row-wise peak to average source density is the highest. The gamma heating is along the center line of column I. Translation to the edge makes minor errors.

(c) Fraction of distance up from bottom of row 7.

TABLE III

Gamma Heating Rate (Watts/cc)
in Dry Canal Storage ^(a)

Time (Days) After Shutdown

	50	75	100	150	200
Al	.17	.11	.08	.53	.033
Cd	.71	.46	.34	.20	.13

(a) Based on operation at 621 W/cc for 45.7 days.

TABLE IV

Peak Heating Rates in Fuel Meat
Due to Beta Decay in Canal Storage^(a)

Time (Days)	50	75	100	150	200
Total β Energy Rate	.114	.078	.059	.039	.028
Peak β Heating Rate	3.14	2.13	1.62	1.07	.77

(a) Heating rates in watts/cc are calculated for fuel which served in position 7I for 45.7 days. The rates are at the tips of the edge fuel plates in the axial plane of highest source density. Equations (12a, 12b) yield

$$p_{\max} = 27.5 \bar{p}_{\beta}(t)$$

$\bar{p}_{\beta}(t)$ is the average watts/cc smeared over the entire core. However, this heat is actually concentrated in the fuel "meat." Furthermore, it is peaked at the outer edges of the fuel plates, and in the outer plates. Equation 11a and 11b express this heat distribution. The factor 27.5 converts the average $\bar{p}_{\beta}(t)$ into the maximum heating rate in the meat at the corners of the fuel element which occupied position 7I. In particular, using the data below Equations 11a, 11b, the factor is obtained from

$$\begin{aligned}
 27.5 &= 9.87 \times F_{7I} \times \frac{V}{V_E} \times x_{\max}(x) y_{\max}(y) z_{\max}(z) \\
 &= 9.87 \times 0.0308 \times 52 \times 1.25 \times 1.13 \times 1.23
 \end{aligned}$$

TABLE V

Compositions of Core Regions

Region	Volume (cc)	Material	Volume Fraction	Atoms/cc $\times 10^{-24}$	Density (g/cc)
ETR Fuel Element (3.034x3.034x36) in ³	5430	H ₂ O	0.607	0.0202	0.604
		H		0.0404	0.0675
		O		0.0202	0.536
		6061 Al	0.388	0.0233	1.041
		U	0.00533	0.000252	0.0983
		U ²³⁵		0.000233	0.0916
		U ²³⁸		0.000017	0.0067
Control Rod Fuel Section	5430	H ₂ O	0.466	0.0156	0.465
		H		0.0312	0.0521
		O		0.0156	0.413
		6061 Al	0.532	0.0320	1.433
		U	0.00243	0.000116	0.0454
		U ²³⁵		0.000108	0.0423
		U ²³⁸		0.00001	0.0031
Filler Pieces (SFR)*	5430	H ₂ O	0.197	0.00659	0.197
		H		0.0132	0.0221
		O		0.00569	0.174
		6061 Al	0.803	0.0483	2.163
5430	H ₂ O	0.190	0.00637	0.191
		H		0.0127	0.0213
		O		0.00637	0.169
		6061 Al	0.702	0.0423	1.894
		Fe(steel)**	0.108	0.00914	0.848

TABLE V (cont.)

Region	Volume (cc)	Material	Volume Fraction	Atoms/cc $\times 10^{-24}$	Density (g/cc)
5430	H ₂ O	0.184	0.00616	0.185
		H		0.0123	0.0206
		O		0.00616	0.164
		6061 Al	0.600	0.0361	1.619
		Fe(steel)**	0.216	0.0183	1.700
5430	H ₂ O	0.177	0.00593	0.178
		H		0.0119	0.0198
		O		0.00593	0.157
		6061 Al	0.499	0.0300	1.346
		Fe(steel)**	0.324	0.0275	2.55
5430	H ₂ O	0.171	0.00573	0.172
		H		0.0115	0.0191
		O		0.00573	0.152
		6061 Al	0.397	0.0239	1.071
		Fe(steel)**	0.432	0.0366	3.40
FEFPL (complete homogenization)	21,700	H ₂ O	0.051	0.0017	
		H		0.0034	
Type 316 Stainless Steel**		O		0.0017	
		Al	0.309	0.0186	
		Fe(steel)**	0.277	0.0235	
		Na	0.257	0.0056	0.213
		U		0.000404	(985° F)
		U ²³⁵		0.000376	
		U ²³⁸		0.000028	

TABLE V (cont.)

* Notation for filler pieces:

○ denotes a 6061 Al SFR plug

● denotes a type 304 stainless steel filler plug

** All stainless steels are treated as pure iron for calculating gamma ray attenuation. The chief minor constituents Cr and Ni are very much like iron for this purpose.

TABLE VI
Homogenized Core Composition
(for leakage calculation)

Region	Volume (cc)	Material	Volume Fraction	Atoms/cc $\times 10^{-24}$	Density (g/cc)
Entire Core Homogenized	4.24×10^5	H ₂ O	0.44	.015	.44
		H		.030	.05
		O		.015	.30
		Al	0.49	.029	1.32
		Steel**	0.05	.0039	.36
		U		.00015	.06
		U ²³⁵		.00014	
		U ²³⁸		.000011	
		Na	0.01	.00029	.011

** See Table V

TABLE VII

ISOSHL D Fission Gamma Sources for 435 W/cc and 45.7 days Operation*

Group	Energy (meV)	0 s	6 s	60 s	600 s	3600 s
		Photons/ cc sec	Photons/ cc sec	Photons/ cc sec	Photons/ cc sec	Photons/ cc sec
		($\times 10^{-12}$)	($\times 10^{-12}$)			
1	.015	.00644	.00644	.00644	.00643	.00623
2	.025	.0475	.0475	.0474	.0465	.0413
3	.035	.0814	.0814	.0813	.0814	.0812
4	.045	.0317	.0316	.0316	.0312	.0297
5	.055	.198	.198	.198	.197	.195
6	.065	.482	.471	.386	.0613	.00592
7	.075	.0132	.0132	.0130	.0120	.00801
8	.085	.491	.491	.489	.471	.406
9	.095	.0912	.0912	.0911	.0901	.0880
10	.15	4.42	4.30	3.71	2.87	1.93
11	.25	3.61	3.56	3.18	2.33	1.51
12	.35	3.66	3.63	3.37	2.50	1.11
13	.475	6.78	6.71	6.19	4.42	3.00
14	.65	7.14	7.11	6.87	5.80	4.49
15	.825	5.82	5.77	5.45	4.52	3.18
16	1.0	3.08	3.07	2.97	2.39	1.22
17	1.225	3.34	3.29	2.87	1.81	.885
18	1.475	4.64	4.58	4.07	3.00	1.97
19	1.7	.510	.498	.415	.291	.173
20	1.9	1.03	1.03	.923	.643	.443
21	2.1	.639	.635	.598	.420	.139
22	2.3	.505	.500	.455	.321	.102
23	2.5	.637	.634	.602	.499	.321
24	2.7	.406	.395	.290	.116	.0503
25	3.0	1.04	1.03	.876	.352	.0901

* The 435 W/cc is the average power density in the core, excluding the FEFPL loop volume, but including all filler pieces in the homogenization. The resulting average source densities were used in the leakage calculation in which the FEFPL loop volume was included in the core homogenization. This is a slight inconsistency but has no bearing on the relative amounts of energy leaking out and deposited in the core.

TABLE VIII

SINBAD Fission Gamma and Beta Sources for 621 W/cc for 45.7 Days^(b)

Group	MeV	0		1 s		6 s		60 s		600 s		3600 s		1d	
		$\bar{p}_j$ (a)		$\bar{p}_j$ (a)		$\bar{p}_j$ (a)		$\bar{p}_j$ (a)		$\bar{p}_j$ (a)		$\bar{p}_j$ (a)		$\bar{p}_j$ (a)	
1	.03	3.35		3.35		3.33		3.18		2.59		2.49		2.02	
2	.08	1.75		1.75		1.74		1.74		1.73		1.72		1.60	
3	.15	4.08		4.07		4.07		4.05		3.83		2.98		1.67	
4	.25	7.05		7.02		6.86		5.41		2.10		1.21		1.06	
5	.425	54.2		48.8		30.3		10.9		7.75		4.80		2.86	
6	.65	38.0		33.7		19.0		4.48		4.12		3.73		1.93	
7	.825	3.82		3.82		3.82		3.69		2.74		2.30		1.33	
8	1.0	3.99		3.97		3.88		3.14		1.97		.696		.239	
9	1.225	14.6		13.6		9.94		6.27		5.55		3.71		.192	
10	1.475	3.37		3.11		2.55		2.14		2.01		1.57		.222	
11	1.7	1.70		1.70		1.68		1.53		1.09		1.05		1.02	
12	1.9	2.17		2.17		2.17		2.04		1.03		.545		.191	
13	2.1	13.7		12.1		6.75		1.19		.986		.480		.0511	
14	2.3	.978		.978		.978		.975		.844		.360		.0475	
15	2.5	2.56		2.00		1.32		1.20		1.16		.870		.179	
16	2.7	.517		.516		.510		.460		.364		.317		.153	
17	3.0	1.05		1.05		1.05		1.03		.789		.276		.112	
18	3.35	.353		.353		.353		.350		.282		.0646		0	
19	3.75	.232		.232		.232		.231		.215		.130		0	
20	4.25	.777		.683		.365		.0343		0		0		0	
21	4.75	.0449		.0449		.0449		.0449		.0446		.00387		0	
22	5.25	.382		.382		.378		.241		0		0		0	
Total		159		145.		101.		54.3		41.2		29.3		14.9	
Betas ccxsec		105		101		90.7		72.3		51.3		34.7		15.5	
β watts/cc			25.6		23.7		19.0		12.5		6.78		3.73		.910

(a) $\bar{p}_j$ = average photons per cc x sec (Cf. Eq. 9).(b) Multiply all disintegration rates by 10^{12} .

TABLE IX

Energy Fluxes at Selected Points External to the Core

Detector Position (cm) (Relative to Core Center)			Total Energy Flux (watts/cm ²)		Flux Ratio	Comments
X	Y	Z	With Core Attenuation	Zero Core Attenuation		
400	0	0	0.295	1.12	0.27	Out along x 366 cm from yz face
0	0	400	0.255	1.37	0.19	Out along z 354 cm from xy face
283	283	0	0.524	2.00	0.26	Out through core edge normal to z
239	0	321	0.295	1.20	0.25	Out through core edge normal to y
34.5	17	29.6	36.3	138	0.26	0.5 cm out from yz face along x
17	17	92	38.8	137	0.28	2 cm out from xy face along z
Average					0.25	

TABLE X

Overall Distribution Functions for Row (x)
and Column (y) Directions in Columns G - K

(For Gamma Sources)

Column	F_{nI}	$\bar{P}_{nI} \left(\frac{W}{cc} \right)$	$x/7.7046$	$X(x)$	$y/7.706$	$Y(y)$
G	.0446	288	0.5	1.0	0.5	0.58
					1.5	0
					2.5	2.44
					3.5	1.97
					4.5	0
H	.0959	619	1.05	0	0.5	1.32
			1.174	1.34	1.5	1.38
			1.375	1.20	2.05	0
			1.625	1.20	2.175	1.04
			1.825	1.34	2.375	0.93
			1.95	0	2.625	0.93
					2.825	1.04
					2.95	0
					3.5	1.23
					4.5	0.29
I	.0720	465	2.05	0.0	0.5	1.12
			2.175	1.34	1.5	0
			2.375	1.20	2.05	0
			2.625	1.20	2.175	2.86
			2.825	1.34	2.375	2.57
			2.95	0.0	2.625	2.57
					2.825	2.86
					2.95	0
					3.5	0
					4.5	1.74
J	.1182	763	3.05	0.0	0.5	1.14
			3.175	1.34	1.5	1.20
			3.375	1.20	2.05	0
			3.625	1.20	2.175	.98
			3.825	1.34	2.375	.88
			3.95	0.0	2.625	.88
					2.825	.98
					2.95	0
					3.5	1.26
					4.5	.67
K	.0523*	338	0.5	1.0	0.5	1.50
					1.5	0.48
					2.5	0.48
					3.5	2.54
					4.5	0.0

* Positions 7K and 8K were arbitrarily assigned .5% of core power.

Table X (Cont.)

Notes:

x coordinates measured from left edge of Column G

y coordinates measured up from bottom of row 9

$$\sum X(x_i) \Delta x_{i/a} = 1.0$$

$$\sum Y(y_i) \Delta y_{i/a} = 5.0$$

a = 7.7064 cm = width of a lattice location

TABLE XI

Source Volume Subdivisions Along Row (x)

and Column (y) Directions

(For Gamma Sources)

Column	Row	Intervals (cm)	
		Δx	Δy
G	5	7.7064	No Source
	6	"	7.7064
	7	"	7.7064
	8	"	No Source
	9	"	7.7064
H	5		7.7064
	6	In each column H, I, J, every row is divided beginning at the left edge into six intervals with widths as follows:	7.7064
	7		Same as for x in G, H, I
	8		No Source
	9		7.7064
I	5		7.7064
	6	1st interval 0.77064	No Source
	7	2nd interval 1.15596	Same as for x in G, H, I
	8	3rd interval 1.92660	No Source
	9	4th interval 1.92660	7.7064
J		5th interval 1.15596	
	5	6th interval <u>0.77064</u>	7.7064
	6		7.7064
	7	Sum of all 7.70640	Same as for x in G, H, I
	8		7.7064
K	5	7.7064	No Source
	6	"	7.7064
	7	"	7.7064
	8	"	7.7064
	9	"	7.7064

TABLE XII

SINBAD Group Structure and Associated Energy
Absorption and Conversion Coefficients^a

Group	E_j (MeV)	Al		H ₂ O		Fuel Meat		Cd	
		μ (cm ⁻¹)	Conv (x 10 ¹³)	μ (cm ⁻¹)	Conv (x 10 ¹³)	μ (cm ⁻¹)	Conv (x 10 ¹³)	μ (cm ⁻¹)	Conv (x 10 ¹³)
1	0.03	2.24	0.1077	0.139	.00670	30.9	1.486	125.	6.0
2	0.08	0.140	0.0179	0.0250	.00320	2.56	0.328	18.7	2.40
3	0.15	0.0739	0.0178	0.0269	.00649	1.11	0.266	3.53	0.85
4	0.25	0.0728	0.0292	0.0309	.0124	0.479	0.192	1.17	0.47
5	0.425	0.0774	0.0528	0.0329	.0224	0.225	0.153	0.440	0.30
6	0.65	0.0754	0.0787	0.0319	.0333	0.143	0.149	0.288	0.30
7	0.825	0.0741	0.0981	0.0319	.0423	0.121	0.160	0.250	0.33
8	1.0	0.0723	0.116	0.0310	.0497	0.109	0.174	0.231	0.37
9	1.225	0.0687	0.135	0.0299	.0589	0.0987	0.194	0.214	0.42
10	1.475	0.0655	0.155	0.0290	.0686	0.0913	0.216	0.199	0.47
11	1.7	0.0638	0.174	0.0269	.0736	0.0873	0.238	0.194	0.53
12	1.9	0.0616	0.188	0.0259	.0792	0.0837	0.255	0.187	0.57
13	2.1	0.0599	0.202	0.0249	.0842	0.0810	0.273	0.184	0.62
14	2.3	0.0590	0.218	0.0249	.0922	0.0799	0.295	0.182	0.67
15	2.5	0.0581	0.233	0.0249	.100	0.0783	0.314	0.180	0.72
16	2.7	0.0565	0.245	0.0240	.104	0.0764	0.331	0.180	0.78
17	3.0	0.0552	0.266	0.0220	.106	0.0765	0.363	0.179	0.86
18	3.35	0.0532	0.286	0.0219	.118	0.0741	0.398	0.171	0.92
19	3.75	0.0519	0.312	0.0210	.127	0.0731	0.440	0.163	0.98
20	4.25	0.0501	0.342	0.0202	.138	0.0720	0.491	0.191	1.3
21	4.75	0.0490	0.374	0.0195	.149	0.0715	0.545	0.190	1.45
22	5.25	0.0483	0.407	0.0188	.159	0.0675	0.570	0.196	1.65

^aConversion coefficients are calculated from $(\text{Conv})_j = \mu_j E_j \times 1.603 \times 10^{-13}$.

Data from E. Storm and H. I. Israel, "Photon Cross Sections from 0.001 to 100 MeV for Elements 1 through 100", LA-3753, November 1967.

TABLE XIII

Activation Gamma and Beta Data

Atom i-1	N_{i-1} (atoms/cc)	σ_{i-1} (barns)	$\bar{\sigma}_{i-1} N_{i-1} \phi$ atoms cc sec	$\bar{E}_{\gamma}$ (MeV)	$\bar{E}_{\beta}$ (MeV)	γ Heat (w/cc)	β Heat (W/cc)
$\text{Na}^{23} \rightarrow$ Na^{24} (15.0 h)	2.13 -04	0.53	2.51 +10	1.369 (100) 2.754 (100)	1.389 (max)	0.017	0.0019
$\text{Al}^{27} \rightarrow$ Al^{28} (2.3 m)	290 -04	0.235	151 +10	1.780	2.85	0.43	0.23
$\text{Cr}^{50} \rightarrow$ Cr^{51} (27.8 d)	0.32 -04	17.0	12.2 +10 $\times 0.68 =$ 8.2 +10	0.320 (9%)	e^{-} 0.315	0.0004	0.0041
$\text{Cr}^{54} \rightarrow$ Cr^{55} (3.52 m)	0.18 -04	0.38	0.15 +10	No γ	2.59 (max)	---	0.0002
$\text{Mn}^{55} \rightarrow$ Mn^{56} (2.57 h)	0.65 -04	13.3	19.2 +10	0.847 (99) 1.811 (29) 2.11 (15)	2.85 (max)	0.052	0.029
$\text{Fe}^{54} \rightarrow$ Fe^{55} (2.6 y)	1.68 -04	2.9	10.8 +10 $\times 0.033 =$ 0.36 +10	Mn x-rays	---	---	---
$\text{Fe}^{58} \rightarrow$ Fe^{59} (45 d)	0.089 -04	1.1	0.22 +10 $\times 0.5 =$ 0.11 +10	0.143 (.8) 0.192 (2.8) 1.095 (56) 1.292 (44)	1.57 (max) (0.3) 0.475 (max)	0.0002	--
$\text{Ni}^{62} \rightarrow$ Ni^{63} (100 y)	0.145 -04	15	4.8 +10 $\times 0.00087 =$ 0.00042 +10	No γ	0.067 (max)	---	---
$\text{Ni}^{64} \rightarrow$ Ni^{65} (2.56 h)	0.047 -04	1.5	0.16 +10	0.368 (4.5) 1.115 (16) 1.481 (25)	2.13 (max)	0.0001	0.0002
TOTAL						0.50	0.27

Table XIII (Cont'd)

Notes:

Table XIII gives data for use in Equation (18) which expresses the rate of decay, $\lambda_i N_i$, of atoms i which are formed by neutron capture of stable atoms one atomic number lower. These are denoted by $i-1$. The first column identifies the stable isotope $i-1$ which captures a neutron to become atom i . The next three columns give the atom densities, capture cross section and activation rates of these stable isotopes. Succeeding columns give the gamma energies and beta energies released by decay of atom i and the resulting gamma and beta heating rates in the core.

The numbers -04 and +10 denote powers at 10. Where only a fraction of the activated isotopes decay to yield gammas, this fraction is shown under $\bar{\sigma}_{i-1} N_{i-1} \phi$.

The σ_i is a 2200 m/sec activation cross section. $\bar{\sigma}_i$ is the average over a Maxwellian spectrum.

Under $\bar{E}_\gamma$ the parentheses enclose the % of decays yielding the gamma ray.

The 0.27 total W/cc of betas is high because the average beta energy is usually 1/3 of the maximum.

Fig. 1 Horizontal Plane Configuration for ETR with FEFPL in Place

Al Filler
Piece

Fuel Element

Control Rod

Experiment

Numbers (e.g., 3.08) are percentage of total power carried by the fuel element
See Table V for further information about experiment types.

Fractional distance across column I from left side

Fig. 9 U, Beta Column

Beta decay heating rate in fuel meat from fission product betas. Profile across tips of fuel plates in position 7I and in the axial plane of highest gamma source density (plates run row-wise)

Fractional distance across column 1 from left edge

Fig. 11 U, fission Column

Fission product decay heating in fuel meat from delayed neutron induced fissions. Profile across tips of fuel plates in position 7I and in the axial plane of highest gamma source density. (Plates run row-wise)

Fig. 15 Horizontal Cross Section of the FEFPL Loop

	G	H	I	J	K
5		0.56	2.51	1.58
6	1.76	2.36		2.97	2.66
7	2.18	1.49	3.08	1.73
8		2.64		2.84	
9	0.52	2.54	1.61	2.70	1.57

Numbers (e.g., 3.08) are percentages of total power carried by the fuel element

- Aluminum SFR plug in aluminum filler piece
- Same as but with 5 in. cylinder of 40 mil Cd for axial flux tailoring
- 304 Stainless Steel SFR plug in aluminum filler piece

Aluminum Filler Piece

Fuel Element

Control Rod

Fig. 16 Local Core Configuration for Calculating Heat Production Around the Hot Spot (numbers are percentages of power carried in fuel bearing locations)

$X(x) = Y(y)$

Fig. 22 Fuel configuration used to calculate gamma heating in dry canal storage

Fig. 23 Cross section of canal storage unit

Fig. 25 Maximum heating rate due to beta decay in the fuel element removed from 71 hole after operating 45.7 days at 5.4 MW. (The highest element power in ETR)