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FISSION PRODUCT TRANSPORT
IN HTGR SYSTEMS - A SUMMARY

by

F. E. Vanslager, W. E. Bell,
O. Sisman, and M. T. Morgan

April 22, 1970

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F. E. Vanslager, W. E. Bell,
O. Sisman,* and M. T. Morgan*

This is a preprint of a paper to be presented at the ORNL Gas-Cooled Reactor Information Meeting, April 27-29, 1970, Oak Ridge, Tennessee, and to be printed in the Proceedings.

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FISSION PRODUCT TRANSPORT IN HTGR SYSTEMS - A SUMMARY*

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ABSTRACT

A brief summary of fission product transport in a gas-cooled reactor is presented. Analytical methods applicable to reactor situations are presented along with a variety of old and new experimental data to illustrate the calculational techniques. These methods indicate that the gaseous activity in a well-designed gas-cooled reactor system can be easily held to acceptable levels and will consist almost entirely of noble gases. Some plateout activity will result from these noble gas precursors as well as from the halogen precursors which are found to have release rates essentially equal to noble gases. The other fission products, provided normal precautions are taken, will not be released from the fuel elements in sufficient quantities to significantly increase the plateout levels above that resulting from these gaseous precursors.

INTRODUCTION

The intent of this paper is (1) to point out the main features of fission product behavior in HTGR systems, (2) to summarize recent developments in the analysis of fission product release and plateout, and (3) to summarize the results of recent investigations of fission product release.

A typical HTGR system might have fuel consisting of uranium and thorium dicarbide kernels (microspheres) overcoated with layers of pyrolytic carbon and dispersed in a bonding or filler matrix of carbonaceous

*Work supported in part by the U. S. Atomic Energy Commission.

material. Oxide kernels, as well as dicarbide kernels, can be used; also, silicon carbide layers are often incorporated in the fuel particle coatings. The fuel material is contained in fuel elements consisting of graphite blocks which are cooled by a circulating helium coolant stream.

The primary barrier to the release of fission products into the helium coolant system is the coating on each individual fuel particle. An effective secondary barrier for metallic fission products is the retention and sorption property of the fuel matrix material and the fuel element structural graphite. These barriers limit the release of fission products into the helium coolant stream. Gas-borne fission products that are released into the coolant are maintained at a satisfactory level by employing a bypass cleanup system. This helium purification system continuously processes a portion of the coolant stream through a series of adsorption beds culminating in a cryogenic bed which removes essentially all of the fission products in this helium stream.

In addition to the helium purification system, the steam generators and other reactor components are effective in removing (by plateout) fission products other than the noble gases. Consequently, the noble gases constitute the majority of the gaseous activity in a reactor circuit.

DIFFUSION OF FISSION PRODUCTS

The transport of fission products in a stagnant medium is generally described by a diffusion equation of the form

$$\frac{\partial C}{\partial t} = \nabla \cdot (D \nabla C) - \lambda C + B, \quad (1)$$

where C is the concentration, t is the time, D is the diffusion coefficient, λ is the decay constant, and B is the birth rate per unit volume. In coated fuel particles with spherical symmetry and a constant diffusion coefficient, this equation takes the form

$$\frac{\partial C}{\partial t} = \frac{D}{r^2} \frac{\partial}{\partial r} \left[r^2 \frac{\partial C}{\partial r} \right] - \lambda C + B. \quad (2)$$

This equation can be used to treat the formation, diffusion, and decay of a fission product nuclide in either the kernel or coating, or both. Generally, a buffer layer of low-density pyrolytic carbon exists between kernel and coating, but this can be neglected.

For the case where the birth rate is constant throughout a particle (e.g., a bare uranium carbide particle), the solution to Eq. (2), assuming time-independent properties and zero initial and outside concentrations, is readily obtained. The release of fission products is usually expressed in terms of the R/B value, which is the instantaneous release rate divided by the birth rate. This R/B value as a function of time is given by

$$\frac{R}{B}(t) = - \frac{D4\pi r^2 \frac{\partial C}{\partial r}}{\frac{4}{3} \pi r^3 B} \Big|_{r=a} \quad (3)$$

$$= 3 \left(\frac{D}{\lambda a^2} \right)^{\frac{1}{2}} \left[\coth \left(\frac{\lambda a^2}{D} \right)^{\frac{1}{2}} - \left(\frac{D}{\lambda a^2} \right)^{\frac{1}{2}} \right] - 6e^{-\lambda t} \sum_{n=1}^{\infty} \frac{\exp(-n^2 \pi^2 D t / a^2)}{n^2 \pi^2 + \lambda a^2 / D},$$

where a is the outside radius of the particle. This expression starts at zero at time zero and reduces to the first term as t approaches infinity. That is,

$$\frac{R}{B}(\infty) = 3 \left(\frac{D}{\lambda a^2} \right)^{\frac{1}{2}} \left[\coth \left(\frac{\lambda a^2}{D} \right)^{\frac{1}{2}} - \left(\frac{D}{\lambda a^2} \right)^{\frac{1}{2}} \right]. \quad (4)$$

This equation represents the steady-state behavior. The solid line on Fig. 1 is a plot of Eq. (4) and illustrates that the release from uncoated (bare) particles should be proportional to the square root of the fission product half-life for R/B values less than 0.1.

If there is a pyrolytic carbon coating on the particle, Eq. (2) must be solved with a step change in the birth rate at the kernel/coating interface--down to a value indicative of coating contamination. The dashed lines on Fig. 1 illustrate the behavior that would be expected for different values of the birth rate ratio B_2/B_1 , where B_2 and B_1 are the birth rates in the coating and the kernel, respectively. It should be noted that the behavior is quite different if there is no contamination in

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Fig. 1. Steady-State R/B Values vs Half-Life and Coating Contamination.

the outer coating; i.e., $B_2/B_1 = 0$. However, it should also be noted that the position of the curve for $B_2/B_1 = 0$ depends on the ratio of the kernel and coating diffusion coefficients, the sorptive abilities, and the radii of the two regions. (The equilibrium R/B value for coated particles is discussed in detail by Dunlap and Gulden.¹)

NOBLE GAS RELEASE

Numerous experiments on noble gas fission product release from coated particles have been conducted. The results of some typical experiments that were carried out by Gulf General Atomic (GGA) are shown in Fig. 2 in plots of the steady-state Kr-85m R/B values vs operating time for the GA2-284-6F2 fuel capsule (Cell I),² the GAIL IIIB loop,³ the GAIL IV loop,⁴ and the Peach Bottom reactor.⁵ The data for the 6F2 capsule and the GAIL IIIB and GAIL IV loops are all at roughly comparable temperatures. However, the Peach Bottom reactor results should be multiplied by approximately a factor of 2 to account for a somewhat lower temperature than for the capsule and the GAIL tests. The increase in the R/B values with time is probably indicative of coating failure. The coated particles used in the 6F2 capsule, in GAIL IIIB, and in the Peach Bottom reactor (Core I) had an early type of coating and showed a relatively high degree of failure, whereas the coated particles used in GAIL IV had a later, improved type of coating and showed a low degree of failure.

Figures 3 and 4 are plots of the end-of-life R/B data vs half-life of the fission products. The data generally show a $(t_{1/2})^{1/2}$ (square root of half-life) dependence and the lines were drawn with this slope. This half-life dependence indicates that the release is from either bare kernels or high contamination in the coatings. In no case was there an indication of the sharp increase in release of long-lived fission products in the manner indicated by the dashed curve with $B_2/B_1 = 0$ (see Fig. 1). (In all of these experiments the geometrical and operational complexities of the experiments should be borne in mind. Without going into details, it can be stated that a variety of difficulties usually affected the accuracy of the data points.)

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Fig. 2. Kr-85m R/B Values vs Burnup.

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Fig. 3. R/B Values vs Half-Life for 6F2 (Cell I) and GAIL IIIB.

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Fig. 4. R/B Values vs Half-Life for GAIL IV and Peach Bottom.

To more fully understand the release process, the end-of-life Kr-85m R/B values at 1300°C for a number of ORNL, Dragon, and GGA experiments are assembled in roughly chronological order in Fig. 5. The first point is the result obtained by GGA⁶ for an uncoated carbide while the next two are the results obtained by ORNL for an uncoated carbide⁷ and a mechanically uncoated carbide.⁸ The next point is a Pluto IIIA result⁹ obtained by the Dragon Project. This result was multiplied by a factor of 5 to account for the 20% particle failure. The next six are early results on coated particles obtained from the GA 309 sweep capsule⁶ series irradiated in 1960-61. The next two are the results obtained from 1962 to 1964 for the GAIL IIIA and IIIB in-pile loop experiments.^{3,10} The next three are results from the three-compartment 6F2 sweep capsule² irradiated in 1963. The first compartment of this experiment had fuel coated with multiple layers of silicon carbide and pyrolytic carbon. Since only approximately one-fourth of the particles failed during the irradiation, the Kr-85m R/B value was arbitrarily multiplied by a factor of 4 to indicate release from the kernels alone.

The next point is from the GAIL IV experiment⁴ irradiated from 1964 to 1966. Since it was known to have a particle failure of only 0.8% at end-of-life, this value was arbitrarily multiplied by 125. The last point is from Peach Bottom Core I results.⁵ As mentioned previously, this result was increased by a factor of 2 to account for the lower temperatures. This temperature correction is obtained by assuming that the release rate has an exponential temperature dependence with an activation energy of approximately 30 kcal/g mole.⁶

It can be seen from Fig. 5 that all of the end-of-life Kr-85m R/B values are approximately the same if one makes corrections for the amount of failed particles. Therefore, it is likely that the end-of-life release is from exposed kernels rather than from diffusion out of contaminated coatings. Also, it was noted that particle damage ranged from minor distortion to gross distortion with the final kernel volume more than twice the original kernel volume. Therefore, it is likely that the characteristic dimension (a) in Eq. (3) is much smaller than the radius of the kernel.

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Fig. 5. Kr-85m R/B Values Corrected to 1300°C.

The similarity of the releases from all of these experiments indicates that there is a noble gas release rate characteristic of failed fuel, and that the rate is relatively independent of initial material or of the degree of failure. Therefore, the reactor designer can establish the design noble gas release levels by determining the amount of fuel failure to be expected and using a release curve obtained from an appropriate experiment having approximately fully failed fuel. If one ignores precursor contributions, the number of noble gas atoms in the coolant stream is then governed by the equation

$$\frac{dn}{dt} = fB\left(\frac{R}{B}\right)_{\text{failed}} - n\lambda - nF, \quad (5)$$

where n is the total number in the coolant stream, f is the fraction of the fissions occurring in failed fuel, and F is the fractional removal rate by the purification system.

Although the situation is relatively simple from the reactor design point of view, the details of the noble gas release process need to be better understood. For example, there is not a clear understanding of the reason for the different krypton-to-xenon release ratios obtained from different experiments. The above-mentioned GGA experiments raise the possibility that this ratio is approximately 2-1/2 for badly failed fuel and 4-1/2 for relatively intact fuel. However, an ORNL experiment on uncoated carbide particles⁷ has indicated that there is no difference between the krypton and xenon release rates.

There is also disagreement among the various investigators about the interpretation of a half-life independent "recoil" release, as opposed to diffusive release. Vanslager has maintained that the word "recoil" does not adequately describe the release process, since the recoiling particle inevitably ends up back in a solid material from which it must be subsequently released. These problems, and many others, warrant further investigation.

HALOGEN RELEASE

The normal reactor operating temperatures are generally high enough so that the release of halogens from the fuel particles is essentially

identical to that of the noble gases. Figures 3 and 4 also include some of the iodine release results obtained from decay experiments and diffusion tube results. It can be seen that, within experimental accuracy, the release of iodine from the fuel particles is identical to that of xenon.

Further verification of the similarity of xenon and iodine release was obtained by recent work at ORNL. The release from pyrolytic carbon-coated uranium carbide particles with intentionally cracked coatings was investigated at a variety of temperatures. A series of tests was conducted by holding the cooling gas flow constant and raising the temperature in 200°C steps by increasing the flux. The Xe-133 and Xe-135 releases into the helium purge stream were measured at steady-state conditions and after a sudden capsule withdrawal, to determine iodine release. The continuing xenon release after withdrawal is from iodine decay. Figure 6, which is a typical plot of Xe-133 and Xe-135 decay at 1400°C, indicates that the release of iodine from failed particles is similar to that of xenon.

The released iodine is not adsorbed, or delayed to any significant extent, by the fuel element structural graphite. However, the metal surfaces of the reactor circuit are quite effective in removing iodine. The GAIL loop experiments have indicated that the majority of the iodine is removed in a single pass around the circuit. Computer programs have been developed¹¹ to handle this mass transport problem. The PAD code (Plateout Activity Distribution), under development at GGA, solves equations of the form

$$\begin{aligned} \frac{\partial N_j}{\partial t} &= S_{N_j} + \lambda_i N_i - N_j \left(\lambda_j + \frac{4k}{d} \right) - v \frac{\partial N_j}{\partial x} + \frac{4kK \sum M_j^p}{d(1 - \sum M_j/Ms_j)} \left(\frac{M_j}{\sum M_j} \right) \\ \frac{\partial M_j}{\partial t} &= S_{M_j} + \lambda_i M_i + kN_j - \lambda_j M_j - \frac{kK \sum M_j^p}{(1 - \sum M_j/Ms_j)} \left(\frac{M_j}{\sum M_j} \right), \end{aligned} \quad (6)$$

where

S_{N_j}, S_{M_j} = source terms for coolant concentration and surface concentration of fission product j ,

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Fig. 6. Total Xenon Released Compared with Xenon Released as Iodine from Cracked Coated UC₂ Particles at 1400°C in ORNL Experiment B9-43.

N_j, N_i = atom concentration of fission product j and its precursor i in the coolant stream,

M_j, M_i = atom surface concentration of fission product j and its precursor i on the plateout surface,

λ_j, λ_i = decay constant of fission product j and its precursor i ,

x, t = axial distance and time coordinates,

k = the mass transport coefficient,

d = the channel diameter,

V = the coolant flow velocity, and

$\frac{k \sum M_j^P}{(1 - \sum M_j / M_{s_j})} \left(\frac{M_j}{\sum M_j} \right)$ is an expression for the desorption of fission

product j from the surface into the coolant. The PAD code can handle an arbitrary number of species interacting through the summation term.

Figure 7 is a plot showing a typical calculated iodine distribution around a reactor circuit. It can be seen that the iodine concentration is greatly reduced by the reactor circuit components. For this case the loop removal rate is considerably greater than the purification system removal rate and, consequently, the total amount of loop plateout is adequately approximated by assuming instantaneous complete plateout in the circuit. The circuit distribution of iodine, however, can only be accurately estimated by employing a mass transport plateout code such as PAD.

ALKALI METAL RELEASE

The alkali metals rubidium and cesium are of intermediate volatility at normal reactor operating conditions. The short-lived isotopes are relatively unimportant, and their release can be bounded without serious consequences by using noble gas release rates from the fuel. However, Cs-134^* and Cs-137 are of special interest because of their effect on maintenance operations.

* Cs-134 is formed by an (n, γ) reaction on fission product Cs-133 .

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Fig. 7. Calculated Iodine Distribution in a Reactor Circuit.

The release of these metals from the fuel particles is likely described by diffusion equations such as Eq. (2). However, the long-lived species Cs-137 and Cs-134 will normally not reach equilibrium during the core lifetime. Consequently, the transient solution to the equation is required. Unfortunately, because of slow convergence of the series solution and added complications when considering a multiregioned fuel particle, the exact solution is not readily available. Therefore, a numerical solution to the equation is often attempted. The FIPER code,^{12,13} developed by GGA, is a numerical solution to the general diffusion Eq. (1). The code will treat any of the three basic one-dimensional geometries, i.e., slab geometry, radial cylindrical symmetry, or spherical symmetry.

Figure 8 shows some typical experimental Cs-137 data obtained from an out-of-pile isothermal anneal of an irradiated fuel particle. The solid line is the result of a FIPER analysis of the experiment using published diffusion coefficients¹⁴ for the particle coating. It can be seen that a reasonable approximation has been obtained. However, computer time limitations preclude performing a FIPER analysis of all of the fuel particles in a reactor core (each has a slightly different temperature and power history). Therefore, a closed-form approximation to the full numerical solution is required. Such an approximation is provided by the "release constant" model, in which the release rate from the fuel particles is assumed to be directly proportional to the inventory. A two-release-constant model presently under investigation by GGA is defined by the coupled equations

$$\begin{aligned}\frac{dn_1}{dt} &= B_1 - R_1 n_1 - \lambda n_1 \\ \frac{dn_2}{dt} &= B_2 + R_1 n_1 - R_2 n_2 - \lambda n_2,\end{aligned}\tag{7}$$

where n_1 and n_2 are the total inventories in regions 1 and 2 of the fuel particle, B_1 and B_2 are the respective birth rates, R_1 and R_2 are the respective release constants, and λ is the decay constant. (The single-release-constant model normally used in reactor calculations is, of course, defined by the first equation.) The dashed lines of Fig. 8 are the

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Fig. 8. Fraction of Cs-137 Released vs Annealing Time.

two-release-constant approximation to the annealing-experiment data. It can be seen that these curves provide a reasonable approximation to the actual situation and can therefore be used for reactor calculations of the release from the fuel particles.

After the release from the fuel particles, there are sorption and diffusion in the fuel element materials and evaporation into the coolant stream. The complexity of the temperature/time history and the temperature dependence of the material properties again require a numerical solution to Eq. (1). The FIPER code, with a source given by the two-release-constant approximation to the release from the fuel particles, provides a reasonable treatment of the problem. In general, the transport process is complex and, under some circumstances, involves concentration-dependent diffusion coefficients and multiple diffusion processes. The overall transport process is highly temperature-dependent.

The loss from the fuel element surfaces is by evaporation described by Freundlich adsorption isotherms.¹⁵ At moderate temperatures the overall diffusion and evaporation process represents a major barrier to cesium loss from the core. Consequently, if the reactor temperatures are low enough, the direct loss will be negligible and the noble gas precursors will constitute the major source of Cs-137 activity. As an illustration of this retention ability, Figs. 9 and 10 present some cesium activity profiles measured on the Peach Bottom D06-01 fuel element and the GAIL IV fuel element.¹⁶ Each set of profiles is from a different elevation along the length of the fuel elements. It can be seen from these steeply sloping profiles that the direct loss of cesium in these experiments was negligible.

The curves in Figs. 9 and 10 are the results of an attempt to fit the data with the FIPER code. The cesium diffusion coefficient was the same for both of these experiments and, for some reason, appeared to be very small and to have the extremely low activation energy of 3 kcal/g mole. Over the temperature range investigated, the temperature dependence of the diffusion coefficient can actually be approximated by a temperature to the 3/2-power dependence, indicating that the rate-limiting step is possibly gas-phase transport at the low concentration levels of these experiments. (It should

Fig. 9. Cesium Profiles in Peach Bottom DO6-01 Fuel Element.

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Fig. 10. Cesium Concentration in GAIL IV (Type 57KBLC Particles Were Closer to GETR Core).

be noted that these data cover three different types of graphite, two of which were impregnated.) These curves should be considered an illustration of the transient FIPER code results rather than a confirmation of a diffusion process; it is hoped that this difficulty can be resolved by examination of additional Peach Bottom elements.

The distribution of cesium activity around the coolant circuit is determined by mass transport calculations such as the PAD code. In general, it is mass-transport-limited rather than saturation-limited. Loop experiments¹⁷ have shown that an approximately exponentially decreasing distribution is obtained in isothermal regions with spikes at points of abrupt flow changes.

A silicon carbide coating is an extremely effective barrier for cesium as well as other metallic fission products.

ALKALINE EARTH RELEASE

The alkaline earth metals strontium and barium are only moderately volatile at normal reactor operating temperatures. Therefore, the release of isotopes other than Sr-89, Sr-90, and Ba-140 can usually be ignored. The 12.8-day Ba-140 chain is important for reactor maintenance considerations because of the high-energy gamma rays emitted by the La-140 daughter. The 30-year Sr-90 chain is important from the standpoint of reactor safety. This nuclide is a bone-seeking beta emitter which has a relatively high yield and long half-life.

The alkaline earth metals, being smaller in diameter than the alkali metals, show considerably more solubility in pyrolytic carbon and thus greater release from coated particles than that illustrated by the results of the Cs-137 anneal in Fig. 8. This is shown, for example, by the ORNL in-pile retention data given in Table 1. However, there is still significant holdup in the fuel material and in the graphite structural members. Silicon carbide coatings effectively retain the alkaline earth metals as indicated by ORNL data given in Table 1 and by GGA data given, for example, in Ref. 18.

Table 1. Retention of Fission Product Metals from Coated and Uncoated Fuel Particles During Irradiation (ORNL Data)

Expt. No.	Fuel	Coating	Max Fuel Temp (°C)	Burnup (%)	Fraction Found in Fuel					
					Zr-95	Ce-144	Sr-89	Ba-140	Cs-137	Ru-106
A9-13	UO ₂	Buffer	1200	5.0	0.96	0.96	NA ^a	NA	0.79	0.96
B9-36	UC ₂	Bare	1200	3.5	0.95	NA	0.067	0.21	0.56	0.98
A9-12	UO ₂	Bu-Iso ^b	1570	5.0	1.0	0.93	0.24	NA	0.37	0.95
B9-33 ^c	UO ₂	PyC	1250	6.0	1.0	1.0	0.53	0.56	0.97	NA
B9-35 ^d	UO ₂	PyC-SiC	1250	5.0	1.0	1.0	NA	1.0	1.0	NA

^aNA denotes not available.

^bFailed

^cEnd-of-life R/B for Kr-88 was 3×10^{-5} .

^dEnd-of-life R/B for Kr-88 was 10^{-8} .

Figure 11 shows the results of a series of out-of-pile strontium diffusion experiments performed at GGA. The solid lines on these plots are the results of a FIPER analysis using the GGA reference design diffusion coefficient data.* It can be seen that a reasonable fit to the data has been obtained, considering the experimental difficulties and uncertainties.

Barium, compared to strontium, has an even longer diffusional holdup; thus, 12.8-day Ba-140 will decay very significantly during transit through the structural graphite. The Xe-140 precursor will therefore constitute the major source of Ba-140 activity in a normal reactor situation.

Mass transport calculations are performed to determine the activity distribution in the reactor circuit. However, for Sr-90 an attempt must also be made to determine the physical form of the deposit in order to analyze accident situations, e.g., the rapid depressurization accident which could lead to ejection of Sr-90 from the reactor containment. The revolatilization of Sr-90 from metal surfaces is relatively negligible. Therefore, the major concern is with aerosols or with the fluid shear removal of dust particles containing Sr-90. If a plant is constructed so as to preclude the release of Sr-90 from the containment, the Sr-90 distribution is of minor importance to the reactor design.

The good retention of strontium, as well as of other fission product metals in HTGR fuel elements, is confirmed in the Peach Bottom reactor. The fractional release of Sr-90 into the main coolant after 300 effective full power days was measured to be 1×10^{-6} , even with essentially uncoated fuel particles.⁵ This release of Sr-90 can be completely accounted for as decay from the Kr-90 precursor.

NONVOLATILE AND REFRACTORY METAL RELEASE

Most of the remaining fission product metals are retained in high degree by the fuel and are essentially nonvolatile at normal reactor operating temperatures. This is shown, for example, by the ORNL retention data for Zr-95, Ce-144, and Ru-106 given in Table 1. Ruthenium-106 and Zr-95 are often used to determine the total amount of fissions that have occurred in a

*The reference diffusion coefficient data for strontium in graphite are described by the equation $\ln D = 5.7 - \frac{3.0 \times 10^4}{T}$.

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Fig. 11. Strontium Diffusion Profiles at 1000°C.

reactor experiment; i.e., the assumption is made that there is essentially no loss of these species during the experiment. The data in Table 1 verify this assumption.

HEAVY METAL RELEASE

The uranium and other heavy metals are also relatively nonvolatile at normal reactor operating conditions. However, at high temperatures there is some amount of migration into the pyrolytic carbon coatings on the fuel particles. The only important effect should be an increase in the halogen and noble gas release from the fuel particles. This could be important since, in a properly designed reactor system, the predominant source of activity is from the halogen and noble gas precursors of the metallic fission products.

Figure 12 shows experimental data on the release of noble gases from particles subjected to a high-temperature anneal until there is some uranium migration through the pyrolytic carbon coating.¹⁹ The data points in the upper right portion of the figure represent the release from fuel particles in a carbon matrix apparently saturated with uranium. The lower point at the right is the release from the particles alone, i.e., from saturated pyrolytic carbon coatings. It can be seen that the smaller granule sizes of the carbon have led to an increased R/B by providing a shorter diffusion distance for the fission products (distributed relatively uniformly by the recoil process). Thus, the release from these "failed" particles, when adjusted for the difference in the matrix granule sizes, would be comparable to the end-of-life releases plotted in Fig. 5. Therefore, this experiment does not represent a new failure mechanism, but rather is an illustration of the surrounding granule size effect on noble gas R/B values.

SUMMARY

In a well-designed reactor system the fission product activity can be easily held to acceptable levels and will result primarily from the release of noble gases and halogens from the fuel elements. The gaseous activity will consist almost entirely of noble gases, since the halogens will be

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Fig. 12. Steady-State Kr-85m R/B Values vs U-235 Concentration in the Carbon Matrix.

removed (by plateout) very effectively by the loop components. This plateout activity distribution can be reasonably approximated if sufficient experimental information is obtained.

The other fission products, provided normal precautions are taken, will not be released from the fuel elements in quantities large enough to significantly increase the plateout level above that resulting from the gaseous precursors. Some of these precautions are: (1) to limit fuel element temperatures, (2) to minimize fuel particle failure, and (3) to provide sufficient sorptive material to retain the metallic fission products. A silicon carbide coating on the fuel particles is also an effective primary barrier to the release of metallic fission products.

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