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IN A FAST REACTOR*

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by

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ABSTRACT

Experimental and analytical techniques have been developed for delayed neutron (DN) signal analysis and characterization that can provide diagnostic information to augment data from cover-gas analyses in the detection and identification of breached elements in an LMFBFR. Eleven flow reduction tests have been run in EBR-II to provide base data support for predicting DN signal characteristics during exposed fuel operation. Results from the tests demonstrate the feasibility and practicability of response-analysis techniques for determining (a) the transit time, T_{tr} , for DN emitters traveling from the core to the detector, and (b) the isotopic holdup time, T_h , of DN precursors in the fuel element. T_{tr} has been found to vary with the relative grid location of the DN source, and T_h is affected by the form of fuel exposed to the coolant as well as the condition of the breach site. These parameters are incorporated into a mathematical formulism that enables one to compute for any exposed-fuel test an "equivalent recoil area." This concept provides a basis for comparison of different run-beyond-cladding-breach tests in fast reactors.

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I. INTRODUCTION

Fuel-element failures in EBR-II are detected by increases in cover-gas activity and by the presence of delayed-neutron (DN) emitters in the primary sodium. Delayed-neutron emitters are detected by the fuel-element rupture detector (FERD) system.⁽¹⁾ A separate sodium loop pumps sodium from the intermediate heat exchanger (IHX) outlet to the FERD detectors located external to the reactor vessel. Here the sodium is counted for its DN precursor content and returned to the primary tank.

Until recently, fission-gas releases from breached elements in EBR-II were well separated from each other in time and were seldom accompanied by measurable DN signals. When DN signals were detected, gas release was usually large enough to allow easy identification of the subassembly containing the breached element by xenon-tag analysis⁽²⁾ with a minimal loss of reactor operating time. By contrast, several recent failures, generally in elements at high burnup or of advanced design, (e.g., carbide or nitride elements) have released very little fission gas (and tag), but have produced significant DN signals. Occasionally, more than one such element has been in the reactor at the same time. Experience with these elements has demonstrated that cover-gas analyses alone may be insufficient for positive identification without adversely affecting the plant factor, and that as much diagnostic information as possible must be obtained from DN signals. It has therefore become important to have available well characterized response data from the DN detection system, and also

the analytical tools capable of deriving meaningful diagnostic information from those data. This is especially true during run-beyond-clad-breach (RBCB) operation, when DN-signal analysis is the only means available for monitoring the extent of exposed fuel and the potential for flow-blockage formation in subassemblies.

This report documents techniques developed at EBR-II for DN signal analysis and characterization that can provide diagnostic information to augment data from cover-gas analyses in the detection and identification of breached elements. The development of an analytical transport and decay model for DN precursors is presented, and the capabilities, limitations and results of the model are discussed.

II. FERD-LOOP FLOW REDUCTION TEST

An important tool in analyzing the response of the FERD system is the FERD-loop Flow Reduction test (FFR test). This test allows the time required for DN emitters to travel to the detectors from the outlet of the IHX (which is at the inlet to the FERD loop) to be varied experimentally. This transport time is inversely related to the FERD-loop flow rate as shown in Fig. 1. Results from this test have provided (a) information about the flow paths of the coolant from different locations in the core to the FERD loop, (b) information concerning the release of DN emitters and their fission-produced precursors from various fuel forms, and (c) insight into the nature of "tramp" uranium in the EBR-II primary sodium.

The FFR tests are performed by sequentially reducing the flow rate of the coolant through the FERD loop in steps and monitoring the DN count rate

Fig. 1. Transit time from IHX to DN detector as a function of flow rate in FERD loop.

at each flow level. Each test is performed with the reactor at full power and in a normal operating posture. Typical flow-rate levels for FFR tests performed during the period from Jan. 1977 to April 1978 were 6.31, 3.79, 2.21, 1.26, and 0 L/s. The flow rates and countrates for an FFR test run with EBR-II's nickel-uranium fission-product source (FPS)⁽³⁾ in the core are shown in Fig. 2. Experience gained with the first several tests indicated that the data were more amenable to analytical treatment if the flow-rate levels were chosen closer together and if the low flow rates were avoided. Accordingly, most tests performed after April 1978 were performed at 6.31, 5.68, 5.05, 4.42, 3.79, and 3.16 L/s.

The FFR test was originally developed to provide data that could be used with a numerical model that computed the transit time from the core to the DN detectors for recoil fission sources (i.e. sources that release fission products primarily by direct recoil to the coolant). The decay model used, herein called the recoil model, was developed in Ref. 4 by Fryer, et al., and independently in Ref. 5 by Jacobi, et al.

When RBCB testing began in EBR-II, it was found that DN signals produced by oxide fuels were often several times larger than would be calculated for simple recoil from the geometrical area of fuel exposed to the coolant. Moreover, arrival-time analyses⁽⁶⁾ indicated that the apparent age of DN emitters was, in some cases, significantly longer than the transit time of sodium from the defect to the detector.⁽⁷⁾ The observed behavior could be explained by either or both of two factors: that DN emitters are released from the fuel by a mechanism (or combination of mechanisms) in addition to direct recoil (e.g. sodium-leaching, bubble trapping and migration of volatile DN precursors, or possibly a diffusion mechanism), and that sodium in

Fig. 2. Variation in sodium flow and FERD count rate during FERD reduced-flow test with recoil fission-product source in core.

the gap between fuel and cladding allows transport (and aging) of DN precursors from a substantially much larger surface than that under the gap. In any event, attempts to apply the recoil model to FFR test data obtained with non-recoil fission sources in core could not produce meaningful estimates of the DN-precursor transit time.

III. ANALYTICAL MODEL FOR NON-RECOIL FISSION SOURCES

As a result of the experience described above a new, more general calculational model has been developed to treat non-recoil fission sources. This model accounts for intermediate decay of the precursors of DN emitters in the fuel and the coolant; enhancement of the total DN signal from non-recoil effects; isotopic holdup in the fuel and the effect of this holdup on the relative proportions of short- vs. long-lived isotopes; and the compound decay of "tramp" DN emitters within the FERD loop.

The new analytical model has been incorporated into the GIRAFFE^(a) computer code.⁽⁸⁾ Calculations will be presented in Sec. V that demonstrate the usefulness of the GIRAFFE code for computing the time required for DN emitters to travel from the core to the inlet of the FERD loop and isotopic holdup times for non-recoil fission sources. Applications of the model for determining surface area of exposed fuel are suggested.

A. Derivation of Equations

The model treats parent and daughter terms separately and assumes that the release of both parent and daughter isotopes from the fuel matrix is controlled by an Arrhenius rate equation with an effective escape-rate coefficient λ_e . The units of λ_e are s^{-1} , and its value is inferred from

(a) General isotope release analysis for failed elements.

empirical data obtained from FFR tests. A further assumption is that residence times for the nuclides of interest in the vicinity of the core is sufficiently short that isotopic depletion effects can be ignored.

1. Fuel:

Based on the above assumptions the number of atoms of parent or daughter isotopes of species i in the fuel at equilibrium [i.e. at times that are long compared to $1/(\lambda_e + \lambda_{\text{decay}})$] are given by

$$N_{PF}^i = \frac{S_P^i}{\lambda_P^i + \lambda_e} \quad (1)$$

$$\text{and } N_{DF}^i = \left[\frac{\lambda_P^i S_P^i}{\lambda_P^i + \lambda_e} + S_D^i \right] / (\lambda_D^i + \lambda_e) \quad , \quad (2)$$

where N_{PF}^i = number of atoms of parent isotope i in fuel at equilibrium,
 N_{DF}^i = number of atoms of daughter isotope i in fuel at equilibrium,
 S_P^i, S_D^i = fission production rates for parent and daughter isotopes
of species i , atoms/s,
and λ_P^i, λ_D^i = decay constants for parent and daughter isotopes of
species i , s^{-1} .

2. Primary Sodium:

In the coolant it becomes convenient to focus attention on a differential slug of Na and work on a unit volume basis. Accordingly, we begin by defining $N_{P,Na}^i$ to be the initial concentration at the defect site, in

atoms per cm^3 of sodium, of parent isotopes that are injected into a differential volume of coolant flowing past the defect. Let F to be the primary coolant flow rate in cm^3/s . Then

$$N_{P,Na}^i = \frac{\lambda_e}{(\lambda_P^i + \lambda_e)} \frac{S_P^i}{F} \quad (3)$$

Similarly, for the daughter isotopes,

$$N_{D,Na}^i = \frac{\lambda_e N_{DF}^i}{F} = \frac{\lambda_e}{(\lambda_D^i + \lambda_e)} \left[\frac{\lambda_P^i}{\lambda_P^i + \lambda_e} \frac{S_P^i + S_D^i}{F} \right] \quad (4)$$

where, as before, the subscripts P and D are used to signify parent and daughter terms, respectively.

3. Atom Concentrations at Detector:

For the precursor atoms of species i the concentration in the sodium arriving at the FERD detector is given by

$$N_{P,Na}^i(t) = \frac{\lambda_e}{(\lambda_P^i + \lambda_e)} \frac{S_P^i}{F} e^{-\lambda_P^i t} \quad (5)$$

For the daughter isotope of species i the governing differential equation is

$$\frac{dN_{D,Na}^i}{dt} = N_{P,Na}^i \lambda_P^i - \lambda_D^i N_{D,Na}^i, \quad (6)$$

subject to the initial condition

$$N_{D,Na}^i(0) = \frac{\lambda_e}{(\lambda_D^i + \lambda_e)} \left[\frac{\lambda_P^i}{\lambda_P^i + \lambda_e} \frac{S_P^i + S_D^i}{F} \right] \quad (7)$$

where expressions again are on a unit volume basis.

The solution to Eqn. 6 subject to the initial condition (7) is obtained by standard techniques. The result is

$$N_{D,Na}^i(t) = \frac{\lambda_P^i}{(\lambda_D^i - \lambda_P^i)} \frac{\lambda_e}{(\lambda_P^i + \lambda_e)} \frac{S_P^i}{F} (e^{-\lambda_P^i t} - e^{-\lambda_D^i t}) + \frac{\lambda_e}{(\lambda_D^i + \lambda_e)} \left[\frac{\lambda_P^i}{\lambda_P^i + \lambda_e} \frac{S_P^i + S_D^i}{F} \right] e^{-\lambda_D^i t} \quad (8)$$

Note carefully that this expression assumes that all of the isotopes initially injected into the coolant at time zero remain together for the

entire trip to the detector. Several flow-related effects can be identified in an operating reactor which can cause deviations from this ideal situation (the principal effects in EBR-II are thought to be large-eddy mixing in the upper plenum, and flow stream splitting and recombination). Nevertheless, while these effects can create significant dispersion in the measured signals, mean values obtained by averaging large numbers of data should follow the behavior governed by the model formulated here. It is in this sense that the assumption employed in formulating Eqns. 5 and 8, which carries through the remainder of the development, will be substantiated when compared to experimental observations in Sec. V.

As a final step we separate the total flow-delay time t into two components by making the substitution

$$t = T_{tr} + t_f \quad (9)$$

In this expression T_{tr} represents the transit time from the core to the inlet of the FERD loop, and t_f is the transit time within the FERD loop. The separation of these variables is necessary since t_f is a control variable whose value is a well known function of the FERD-loop flow rate (Fig. 1), whereas T_{tr} is an unknown system parameter whose value must be computed numerically. If Eqn. 9 is introduced into Eqn. 8, the latter becomes

$$N_{D,Na}^i(T_{tr}, t_f) = \frac{\lambda_P^i}{(\lambda_D^i - \lambda_P^i)} \frac{\lambda_e}{(\lambda_P^i + \lambda_e)} \frac{S_P^i}{F} \left[e^{-\lambda_P^i(T_{tr} + t_f)} - e^{-\lambda_D^i(T_{tr} + t_f)} \right] + \frac{\lambda_e}{(\lambda_D^i + \lambda_e)} \left[\frac{\lambda_P^i}{\lambda_P^i + \lambda_e} \frac{S_P^i + S_D^i}{F} \right] e^{-\lambda_D^i(T_{tr} + t_f)} \quad (10)$$

4. DN Activity at Detector

Equation 10 gives the atom concentration of daughter isotope i in the coolant as a function of the core flow delay time T_{tr} and the FERD-loop transit time t_f . The total activity at the FERD detector for isotope i , $A(T_{tr}, t_f, \lambda_e)^i$, is equal to the concentration of the daughter isotope times the probability of decay for that isotope:

$$A(T_{tr}, t_f, \lambda_e)^i \equiv \lambda_{D,Na}^i(t) \quad (11)$$

or, in view of Eqn. (10),

$$A(T_{tr}, t_f, \lambda_e)^i = \frac{\lambda_D^i \lambda_e}{F} \left\{ \lambda_P^i S_P^i \frac{\exp[-\lambda_P^i(T_{tr} + t_f)] + \exp[-\lambda_D^i(T_{tr} + t_f)]}{(\lambda_D^i - \lambda_P^i)(\lambda_P^i + \lambda_e)} + \frac{\left(\frac{\lambda_P^i}{\lambda_P^i + \lambda_e} S_P^i + S_D^i \right)}{\lambda_D^i + \lambda_e} \exp[-\lambda_D^i(T_{tr} + t_f)] \right\}, \quad (12)$$

where the dependence on the escape-rate coefficient, λ_e , has now been explicitly indicated.

Finally, the total DN activity at the FERD detector is obtained by summing the decay rate for each daughter isotope times P_n^i , the probability of neutron emission for that isotope:

$A(T_{tr}, t_f, \lambda_e) \equiv$ total DN activity at FERD detector, cps per cm^3 of Na.

$$A(T_{tr}, t_f, \lambda_e) = \sum_{i=1}^9 A(T_{tr}, t_f, \lambda_e)^{i_{Pn} i} \quad (13)$$

5. Mean Isotopic Holdup Time

The effective mean isotopic holdup time, T_h , is equal to the reciprocal of the isotopic escape-rate coefficient.

$$T_h \equiv 1/\lambda_e \quad (14)$$

This value, when added to the total transit time from the core to the FERD, provides an approximate estimate of the total age of DN emitters "seen" by FERD. It should be emphasized here that the physical interpretation of T_h as a holdup time can be, at best, only a rough approximation. This is because application of the Arrhenius model cannot account for differences in individual nuclide escape rates that could arise, for example, from atomic mass differences or differences in chemical properties of the various DN precursors. Nevertheless, the usefulness of T_h lies in the experimental observation that gross differences in the form of fuel exposed to the coolant, or in the character and condition of the breach site, give rise to measurable differences in the computed value of λ_e , and hence also T_h .

8. Treatment of Background

In order to accurately correct measured DN activities for the presence of DN emitters produced by the fissioning of tramp uranium it is necessary to determine the behavior of the latter emitters as a function of flow rate within the FERD loop.

It is assumed first that the tramp uranium acts like a recoil source, ejecting both parent and daughter isotopic species with essentially zero holdup. A further assumption is that the location of the tramp source is in the reactor core. In accordance with these assumptions, decay equations were applied for the DN emitters and their parents traveling from the core to the FERD inlet using a transit time of 15.0 secs. For this calculation the tramp-produced parent and daughter species were assumed to behave in the same manner as those released from a fuel defect, and the composite decay equation derived earlier (Eqn. 13) was applied.

Knowing the activity at the FERD inlet and the relative mixture of long-versus short-lived parents and daughters, it was possible to compute the activity within the FERD loop as a function of FERD flow. The resulting total activity (in counts/s) seen at the detectors is shown in Fig. 3.

Superimposed on the predicted curve in Fig. 3 are four activities that were measured during a reduced-flow test⁽⁴⁾ with tramp background (i.e., with no subassembly DN leakers in the core). The agreement between experimentally observed and predicted tramp signals is very good and stands as a justification for the earlier assumptions, and also provides a preliminary verification of Eqn. (13).

Fig. 3. DN activity due to fissioning of tramp uranium.

IV. UTILIZATION OF REGRESSION METHODS TO INFER PHYSICAL PARAMETERS

The equations developed above permit one to predict the DN activity at the FERD station as a function of three parameters: T_{tr} , the transit time from the core to the FERD-loop inlet; λ_e , the isotopic escape-rate coefficient for the fission-product source; and t_f , the flow delay time within the FERD loop. The latter parameter, t_f , is a control variable that can be varied experimentally by systematically increasing or decreasing the flow rate of sodium through the FERD loop in an FFR test. The values of the remaining model parameters, T_{tr} and λ_e , are then determined with an optimization algorithm that minimizes a nonlinear error function of the form

$$E(T_{tr}, \lambda_e) = \sum_{k=1}^J \left\{ W_k(T_{tr}, \lambda_e) [A_{mk} - A_{pk}(T_{tr}, \lambda_e)] \right\}^2, \quad (15)$$

where

A_{mk} = measured DN activity (corrected for tramp background)
at flow step k ,

A_{pk} = predicted activity (i.e., Eq. 13) at flow step k with
parameter set (T_{tr}, λ_e) ,

W_k = weighting function assigned to A_{pk} ,

and J = total number of FERD flow steps.

Data that can be obtained from one FFR test are not sufficient to permit solution of Eqn. 15 for T_{tr} and λ_e simultaneously. The procedure, therefore, is to first perform a series of tests with a source having a known value of λ_e . (In the EBR-II tests a recoil fission-product source is used, for which

$\lambda_e = \infty$). Data from these tests define, via constrained minimization of Eqn. 15, the value of T_{tr} . This value is then used as an input parameter for subsequent tests in which λ_e is unknown. The ability to accurately determine the value of T_{tr} is desirable, since tests have shown its value to depend on the relative core location of the defective element. The value of λ_e is also a useful parameter from a physical standpoint in that its reciprocal provides a suitable measure of the effective holdup time at the defect site.

A. Descriptive Statistics Made Available from TRANSIT Techniques

In addition to the important transport and release parameters described above, several descriptive statistics are made available during the course of the foregoing minimization procedure. These statistical parameters have interesting and useful physical interpretations; they enable one to derive information relating to the fission rate in the DN emitter, the overall efficiency of the FERD detection system, the background due to gamma decay, and ultimately, in principle at least, a measure of the surface area of fuel exposed to the coolant.

Illustrations will be helpful in visualizing the following descriptions. Thus, Fig. 4 shows a plot of measured-vs.-predicted DN signals from a hypothetical FFR test run at 5 flow steps. The Y axis represents the measured countrates, corrected for tramp background. The X axis represents the corresponding predicted countrates computed with the model equations developed in Sec. III. Line I is the best-fit straight line generated by minimization of error function $E(T_{tr}, \lambda_e)$ in Eqn. 15. Frequent reference will be made to this figure in the discussions that follow.

Fig. 4. Plot of five pairs of measured and predicted DN activities and best-fit line from GIRAFFE regression algorithm.

1. Y-Intercept

The Y-intercept, signified by the symbol γ in Fig. 4, is obtained by extrapolation of the best-fit line I to $X = 0$. Physically γ is equal to the flow-independent contribution to the measured DN signal. This is the photoneutron component due to the gamma activity in the coolant (primarily from decay of Na^{24}).

Earlier numerical methods employed with the recoil model⁽⁴⁾ required the gamma background signal to be measured directly. Thus, as shown here, an additional benefit of the regression approach is that the constant term can remain unknown and its value will be determined by the optimization algorithm. (Alternatively, for FFR tests in which the flow is reduced to zero, γ can still be measured directly and used to provide an additional data point, thereby increasing the degrees of freedom in the regression analysis by one).

2. Residuals and Correlation Coefficient

Also produced during minimization of Eqn. 15 are the residuals and the coefficient of correlation for the measured-vs.-predicted data.

The residuals, in counts/s, are a measure of the vertical deviations of the data from the best-fit line. These values can be useful in identifying local deficiencies in the analytical model (e.g. over- or under-predicting at low flow rates); or for spotting anomalous measured data due to, say, a spurious drift in the counting system characteristics or a slightly changing source strength (caused, for instance, by the gradual opening of a cladding defect during the time the FFR test is being performed).

The correlation coefficient, XFIT, is an important quantity from a theoretical standpoint in expressing the effectiveness of the isotopic transport and decay model for predicting the outcome of actual FFR experiments.

3. Slope of Measured-vs.-Predicted Line

Referring once again to Fig. 4, we define C to be the slope of the best-fit measured-vs.-predicted line I. This dimensionless parameter contains some potentially very useful physical information. Theoretical considerations reveal that C is equal to $V_{Na} \alpha \epsilon \rho V / F$, where V_{Na} is the volume of sodium in cm^3 that is "seen" by the DN detectors; α is the fission rate in the source, in fissions per second per g of fissionable material; ϵ is the efficiency of the detector arrangement (dimensionless); ρ is the density of the fuel, in g/cm^3 ; V is the volume of fuel from which recoil fission products are emitted, in cm^3 ; and F is the flow rate of primary sodium, in cm^3/s (not to be confused with the FERD-loop flow rate f).

The magnitude of C is determined quite accurately by the regression technique (typically, $\sigma_C/C < 6\%$). It is therefore possible to extract some reasonably quantitative physical information from C . The only limitation to the accuracy with which one can extract any combination of the six physical parameters comprising C is the uncertainty associated with the complimentary parameters. Put another way, since the uncertainty in C is small, if one can obtain accurate values for N of the parameters (by external measurements), a FFR test will provide reasonably accurate estimates for the remaining $(6-N)$ parameters. The derivations presented in parts B and C below explicate these ideas for a special case.

B. Equivalent Recoil Area

We develop here the mathematical formalism needed to compute the equivalent recoil area and the signal enhancement factor for nonrecoil fission sources. These concepts play a crucial role in the monitoring and characterization of DN signals during RBCB irradiation testing in a fast reactor.

As a first step we set forth a method for comparing the value of C computed from RBCB tests to the value of C computed from FPS data. The decision to base comparisons upon the FPS results stems from many considerations. The FPS has been used in many calibration tests with the FERD; a large body of base data has therefore been accumulated. Furthermore, the relatively large signals produced with the FPS provide excellent regression statistics in transit-time analyses. A final consideration reflecting perhaps the most important feature of the FPS test results is that the total surface area of fuel material is well known.⁽³⁾ In accordance with these considerations, the fission source terms in the GIRAFFE code (variables S_D and S_P in Section III.A) have been normalized to the fission rates of the FPS in core location 5B2. This results in a nominal value of unity for C computed from FFR data using the FPS as the DN source.

The interplay of the various physical parameters which combine to influence the magnitude of C was discussed in Part A above. To compare the value of C for an RBCB test to the value for the FPS tests, we note first that three of the parameters (namely the bulk sodium flow rate, the detector efficiency, and the volume of sodium "seen" by the detectors) do not change; dependence of C on these parameters can therefore be dropped. Accordingly, the value of C should be proportional only to the product of the fission rate in the source and the quantity of fissionable material contained in the "equivalent recoil volume." Thus, for an RBCB experiment we can write

$$C = \frac{\alpha_{RBCB} \times \rho_{RBCB} \times V_{RBCB}}{\alpha_{FPS} \times \rho_{FPS} \times V_{FPS}}, \quad (16)$$

where

α = total fission rate in the fuel (fiss/g/s),

ρ = density of fissionable material (g heavy metal/cm³),

and V = equivalent recoil volume (cm³).

The recoil volume can be alternatively expressed as:

$$V = S_{rc} \times W, \quad (17)$$

where W is the experimentally determined recoil distance in the source material, and S_{rc} is defined to be the "equivalent recoil area" for the fuel. The units of S_{rc} are cm²; it should be pointed out, however, that only in the case of a recoil emitter does S_{rc} represent an actual physical area. For non-recoil sources S_{rc} provides a measure of the square centimeters of the source material that would produce a given FERD response if direct recoil were the only mechanism for release. Defined in this manner, S_{rc} gives us a basis for comparison for different RBCB fuels.

Substituting Eqn. (17) into the earlier Eqn. (16) and rearranging gives

$$[S_{rc}]_{RBCB} = C \times \frac{\alpha_{FPS} \times \rho_{FPS} \times W_{FPS}}{\alpha_{RBCB} \times \rho_{RBCB} \times W_{RBCB}} \times [S_{rc}]_{FPS} \quad (18)$$

D. Enhancement Factor

Depending upon the predominant mechanism responsible for DN-precursor release in a given exposed-fuel experiment it is possible for the value of S_{rc} to differ significantly from the actual surface area S of the defect. We define an enhancement factor, Ω , to be the ratio of the equivalent recoil area S_{rc} to the actual area S of fuel beneath the cladding defect.

$$\Omega = S_{rc}/S \quad (19)$$

To determine Ω for a given fuel form it is necessary to perform a calibration test with either a predefected element for which S is known, or a naturally-breached element for which S can be measured in post-irradiation examination. Knowledge of Ω is desirable since it can then be used in the design of subsequent irradiation tests with similar fuels to estimate, for example, the magnitude of the signal expected for a given defect size. Alternatively, since calculation of S_{rc} , and hence also Ω , takes into account the value of λ_e for the fuel, one can work backwards, using information from a flow reduction test to estimate the defect size during RBCB operation.

Examples of the determination of T_h , S_{rc} , and Ω for some prototypic LMFBR fuels will be presented and discussed in the following section.

V. NUMERICAL RESULTS

The analytical models developed in the foregoing sections have been incorporated into the FORTRAN computer code GIRAFFE. The theoretical considerations and computational strategy employed in the code for transit time or isotopic holdup time calculations are described in detail in Ref. 7. Production and decay constants used by the code are listed in Table I for the daughter isotopes (i.e. the principle DN emitters), and in Table II for the precursor isotopes. In the first part of this section we apply the code to compute transit time values from FFR test data obtained with the FPS in two core locations. Part B summarizes results of applying the code to compute isotopic holdup times for several nonrecoil fission-product source irradiations.

A. Results of Transit Time Calculations for a Recoil Source

Five FFR experiments were performed during full-power irradiation tests with the FPS in EBR-II to provide data needed to evaluate the transit time

TABLE I.
Production and Decay Constants for Daughter Isotopes

<u>Isotope</u>	<u>T_{1/2}, s</u>	<u>λ_D, s⁻¹</u>	<u>Fission^a Yield</u>	<u>Prob of Neutron Emission</u>
⁸⁷ Br	55.7	.01244	.01305	.0235
⁸⁸ Br	15.9	.04359	.01609	.0550
¹³⁶ Te	21.0	.03301	.01189	.0075
¹³⁷ I	24.6	.02818	.02421	.0665
⁸⁹ Br	4.5	.154	.01084	.1360
⁹³ Rb	5.8	.1195	.02564	.0173
¹³⁸ I	6.5	.1066	.01264	.0514
⁹⁴ Rb	2.7	.2477	.01555	.0930
¹³⁹ I	2.4	.2888	.00424	.1390

^aIndependent yields from Reference 9.

TABLE II.
Production and Decay Constants for Precursor Isotopes

Isotope	$T_{1/2}$, s	λ s^{-1}	Cumulative Fast Fission Yield, ^a Fraction	
			^{235}U Fission	^{239}Pu Fission
^{87}Se	5.6	0.124	6.045×10^{-3}	5.724×10^{-4}
^{88}Se	1.5	0.462	2.543×10^{-3}	1.137×10^{-4}
^{136}Sb	0.82	0.850	1.390×10^{-4}	2.357×10^{-5}
^{137}Te	3.5	0.198	2.972×10^{-3}	8.471×10^{-4}
^{89}Se	0.41	1.691	5.901×10^{-4}	1.601×10^{-5}
^{93}Kr	1.27	0.546	2.815×10^{-3}	2.911×10^{-4}
^{138}Te	1.6	0.433	5.367×10^{-4}	1.186×10^{-4}
^{94}Kr	0.21	3.301	9.197×10^{-4}	4.020×10^{-5}
^{139}Te	~ 0	10^9	5.797×10^{-5}	1.196×10^{-5}

^aFrom Reference 9.

T_{tr} . During these tests flow-rate data and DN signal measurements for FERD channels A, B, and C were recorded at a sampling frequency of two per second. At each flow step, approximately 700 recorded observations were averaged to provide input to GIRAFFE. Table III shows a typical set of input and computed results for a GIRAFFE run using data from the April 1978 test, performed with the FPS in core location 5B2. Mean flow rates and measured DN countrates are listed in columns 1 and 2 for each of the six FFR flow steps. Column 3 contains the corrected activities (i.e. measured minus tramp). Activities predicted from the analytical model are listed in column 4. The final column lists the residuals expressed as percentages of the corresponding measured activities.

The computed value of the transit time for this set of data is 14.9 s. Note that λ_e was set to infinity for this calculation, since the FPS is a recoil emitter (Sec. IV). The value of the slope of the measured-vs.-predicted line is 0.94. A plot of the measured-vs.-predicted data is given in Fig. 5. That the slope turned out to be close to unity for this case reflects the normalization of the parent and daughter source terms in GIRAFFE to the source strength and surface area of the FPS in position 5B2. The value of the correlation coefficient is 0.9999, indicating an excellent agreement between the theoretical model and the experimentally measured activities (Fig. 6).

Results for the other FPS tests are summarized in Table IV. The results show that the average transit time from the core to the IHX outlet for position 5B2 is 14.5 s. The average time from position 5E4 is 19.4 s.

These results indicate that a difference of ~ 5.3 s exists in the time taken for DN-emitting fission products to reach the IHX outlet from positions 5B2 and 5E4. The location geometrically nearest the primary outlet pipe (5B2) had a shorter transit time than the location on the other side of the core from

TABLE III. TRANSIT Input and Output Values for FERD
Channel C in FFR Test No. 1: λ_e Infinite

Input		Output		
FERD Flow, L/s	Measured Activity, counts/s	Corrected Activity, counts/s	Predicted Activity, counts/s	Residuals, percent of measured
3.34	676	652.2	677	0.17
1.99	481	464.6	473	-0.73
1.02	257	249.0	251	1.38
.00	17	17.0	0	-6.90

Other Computed Output Values

Transit Time = 14.9 s

C = Slope = $0.94 \pm .004$

GAMMA = Y Intercept = 15.8 ± 3.7

XFIT = Correlation Coefficient = 0.99988

Fig. 5. Measured-vs-predicted activity pairs and best-fit regression line for FFR Test No. 1.

Fig. 6. Plot of predicted and measured DN signals as a function of FERD flow rate for FFR Test No. 1.

TABLE IV. Results of Transit Time Calculations From FPS Data

FFR Test Number	FPS Core Location	Computed Transit Time, s		
		Channel A	Channel B	Channel C
1	5B2	13.0	16.0	14.9
2	5B2	N.A. ^a	14.7	17.0
3	5B2	12.7	12.4	12.7
4	5B2	15.5	15.6	15.2
5	5E4	18.7	20.4	19.2

Average Transit Time from Core Location 5B2 = 14.5 s

Average Transit Time from Core Location 5E4 = 19.4 s

^aNot available.

the outlet pipe (5E4). This shorter time is consistent with a model of the flow in the upper plenum, which has a flow component that rises over the baffle and flows around the circumference of the baffle; such a component would be maximum for position 5E4 and minimum for 5B2.

B. Results of Isotopic Holdup Time Calculations

In this section we summarize results of applying the GIRAFFE code to data obtained from several FFR experiments performed with nonrecoil fission sources in core. Results obtained using channel A data recorded during the first non-recoil source test, designated FFR Test No. 6, are reproduced in Table V. For this test a predefected UO_2 element⁽¹⁰⁾ containing an axial cladding slit was irradiated to simulate the signal-behavior one would expect from a defective element in which the breach site had enlarged to allow significant fuel-sodium reaction.

From Table V the computed value of λ_e for the UO_2 test is seen to be 0.073 s^{-1} . This gives an effective isotopic holdup time of $T_h = 13.7 \text{ s}$. Interpretation of the other input and output parameters given is the same as discussed above for Table III.

When the results obtained using the channel A data from the UO_2 test are combined with results obtained using data from channels B and C, the average value of the slope of the measured-vs.-predicted line is $C = 0.137$. Using Eqn. 18 and the element parameters provided in Ref. 8, the equivalent recoil area is found to be $S_{rc} = 7.29 \text{ cm}^2$. The geometrical area of the cladding defect was 0.2 cm^2 . The enhancement factor, Ω , is therefore 36.5. Thus, in accordance with the definition of Ω (Sec. IV.D), the observed DN signals were about 36.5 times larger than could be explained by recoil release

TABLE V. GIRAFFE Input and Output Values for FERD
Channel A in FFR Test No. 6: $T_{tr} = 15.0$

Input		Output		
FERD Flow, L/s	Measured Activity, counts/s	Corrected Activity, counts/s	Predicted Activity, counts/s	Residuals, percent of measured
3.53	110.9	86.0	437	0.08
2.04	87.9	70.7	321	0.03
1.02	60.1	52.1	179	-0.44
0.00	28.0	28.0	0	0.53

Other Computed Output Values

$$\lambda_e = 0.073 \text{ s}^{-1}$$

Effective Mean Isotopic Holdup Time 13.7 s

C = Slope = 0.13 ± 0.000

GAMMA = Y Intercept = 28.1 ± 0.2

XFIT = Correlation Coefficient = 0.99995

from the 0.2 cm^2 of UO_2 beneath the cladding slit. This signal-enhancement factor as well as the observed aging of the DN emitters strongly suggests other release mechanisms in addition to direct recoil were at work. A possible alternative explanation is that the sodium bond allowed transport (and aging) of DN precursors emitted by recoil from a substantially much larger fuel surface than the exposed surface area directly beneath the defect. This latter transport mechanism is regarded as less likely, however, in light of the significant amount of fuel-sodium reaction product observed at the defect during postirradiation examination. (10)

Table VI summarizes key computed parameters from GIRAFFE runs using data from the predefected UO_2 test just described, and for five other nonrecoil source irradiation tests. The values listed represent mean values obtained by averaging GIRAFFE results from the three independent FERD channels. The last five tests, designated FFR test Nos. 7-11, all involved naturally breached elements. The fuel types, listed in column 2 of the table, included two mixed oxide elements and one advanced uranium-plutonium carbide element. Computed values of the equivalent recoil area given in the final column range from 7.29 cm^2 to 94.10 cm^2 . Noteworthy is the relatively large difference in the computed values of the holdup time between the mixed oxide tests and the mixed carbide tests. The results for the last two tests shown indicate that the mixed carbide element behaved almost like a recoil emitter during tests 10 and 11.

FFR Test No. 7 is the only RBCB test for which an estimate of the defect size is available that can be used to compute an Ω factor. The cladding breach for the element irradiated during that test consisted of an axial cladding crack about 3.81 cm long and 0.0762 cm wide at its midpoint. Post-

TABLE VI. Summary of GIRAFFE Results for FFR Tests Run
with Nonrecoil Fission Sources

FFR Test Number	Fuel Type	FERD Signal At 6.31 L/s, counts/s	λ_e , s^{-1}	T_h , s	C	XFIT	S_{rc} , cm^2
6	UO ₂	110	0.066	15.1	0.137 ± .001	0.9995	7.3
7	(U,Pu)O ₂	460	0.078	12.1	1.11 ± .007	0.9998	49.2
8	(U,Pu)O ₂	550	0.042	23.8	1.24 ± .005	0.9999	57.6
9	(U,Pu)O ₂	675	0.045	22.2	2.03 ± .009	0.9999	94.1
10	(U,Pu)C	255	$\sim \infty$	~ 0	2.28 ± .002	0.9997	11.5
11	(U,Pu)C	310	0.337	2.9	0.42 ± .001	0.9998	17.3

irradiation examination gave no indication of any fuel loss from the defect. The measured diameter increase on the element was a maximum of 7% at the axial center of the breach. The geometric area of the defect was found to be 0.21 cm^2 after the test. The value of S_{rc} was computed to be 49.2 cm^2 . Thus, the enhancement factor for this test was at least 234. (The actual defect area at the time of the FFR test is not known, but may have been smaller than 0.21 cm^2).

It was remarked in Sec. IV that an estimate of the equivalent recoil area imports far greater meaning than is contained in a mere statement of the signal magnitude. The difference between these two indicators of the source condition is made apparent in Table VI. For example, if one compares the values obtained for FFR Test Nos. 6 and 9, it is seen that the signal magnitudes were different by a factor of 6.1, while the corresponding S_{rc} values were different by a factor of 12.9. The reason for the large difference in these two indicators is that the ratio of the signal magnitudes does not take into account differences in λ_e or T_{tr} . More meaningful information is conveyed, in terms of how severe or benign an exposed-fuel test may be, by a comparison of S_{rc} values, since these values automatically accommodate differences in the escape rate and transit time for the DN emitters.

VI. LIMITATION OF TECHNIQUES AND DIRECTION OF FUTURE RESEARCH

The techniques, both experimental and analytical, treated in this report are limited to steady state analyses. Methods required to reduce FFR test data taken during periods of changing source conditions (e.g., during a rise to power or an abrupt change in defect size) have not yet been developed. A further limitation arises from the assumption that only one source (in addition to tramp and gamma background) is in the core. Thus, if two sources are present with significantly different transit times and/or isotopic holdup times, the model equations do not produce an accurate estimate of either parameter. However, the statistical information made available by the GIRAFFE code does then indicate the presence of more than one leaker. In the absence of well characterized tag signals, this fact may very well provide the only means of determining whether one or more than one leaker is in the core.

The Arrhenius model introduced in Sec. III to account for isotopic removal terms in the transport equations was not derived from physical principles. It is, rather, an empirically-based model that "adjusts" the effective removal rate in a manner that best agrees with the observed flow-dependent signal behavior. Use of the resulting escape-rate coefficient in the general transport and decay model provides a particularly tractable technique for accommodating differences in the relative composition of parent and daughter species that result from gross differences in the overall holdup time, however that holdup time arises physically. In this regard use of the Arrhenius equation represents a considerable improvement over the earlier, recoil model. Nevertheless, further research is needed to better understand the interplay of the complex physical mechanisms that are responsible for transport of

fission products and their progeny from the fuel matrix to the coolant under the various conditions that attend sustained operation with exposed fuel in a fast reactor. Ongoing research at EBR-II is focused on reducing the considerable uncertainty that currently exists in this area.

VIII. SUMMARY AND CONCLUSIONS

Experience at EBR-II has shown that the leaker-diagnostic information obtainable from gas analyses alone may be quite limited, especially when operation continues well past the point of initial breach. Even in cases where good gas data are available, fission-gas signals cannot be correlated with breach size. Increased reliance has therefore been placed on DN signal analysis, which remains the only means for monitoring the extent of exposed fuel and the potential for flow-blockage formation in subassemblies.

Response-analysis techniques have been developed and incorporated into the FORTRAN computer code GIRAFFE. The GIRAFFE code has been used to compute the transit time, T_{tr} , for DN emitters traveling from the core to the FERD, and the isotopic holdup time, T_h , for various LMFBR fuels. Results have demonstrated a dependence between the computed value of T_{tr} and the relative core location of the DN source. The quantity T_h has been found to be affected by the form of fuel exposed to the coolant as well as the condition of the breach site. Additional information provided by the GIRAFFE code has been used to infer such physical parameters as the DN signal attributable to gamma activity in the coolant, and the equivalent recoil area for the DN source. This latter concept has been introduced in this study to provide a basis for comparing different RBCB tests. It is believed that this concept conveys far greater

meaning, in terms of how severe or benign an exposed-fuel experiment may be, than could be obtained by simply monitoring the magnitude of the signal.

Finally, although the present investigation has focused on EBR-II's DN detection system, the analytical methods are general and could be used to provide useful diagnostic information in any fast reactor. A major conclusion of this study is that consideration should be given during the design of new plants to provide DN detection systems that are capable of providing useful diagnostic information in addition to the magnitude of the DN signal. Careful design of the reactor and DN systems should make it possible for DN signal analysis to be used for the location of a breached element in addition to monitoring the condition of the breach. Such systems would require multiple DN detection stations (with an appropriate flow-delay time between stations) on a loop having a constant flow velocity, or a single detection station on a side-stream loop having an adjustable flow velocity.

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