

*The Equivalent Fundamental-Mode Source*

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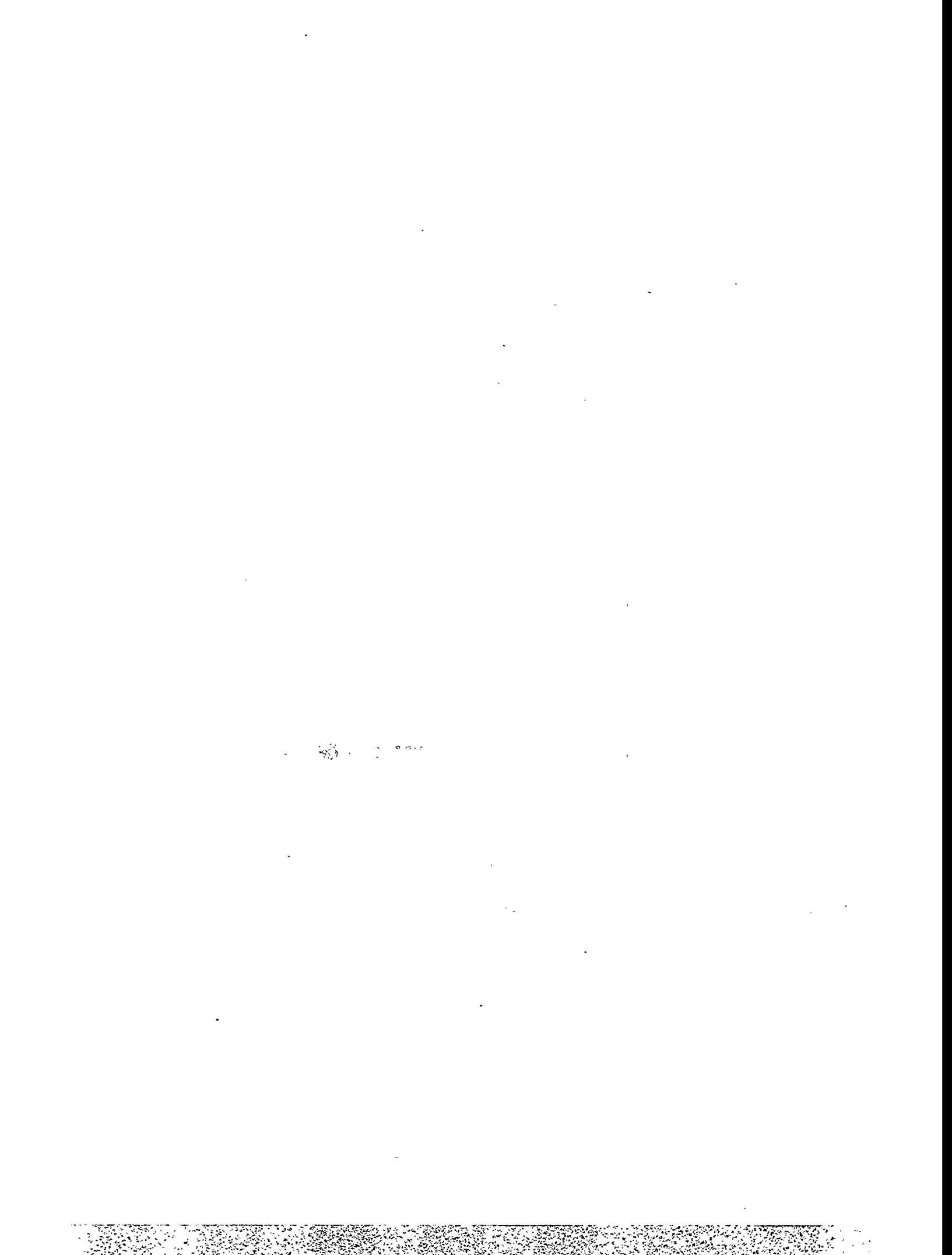
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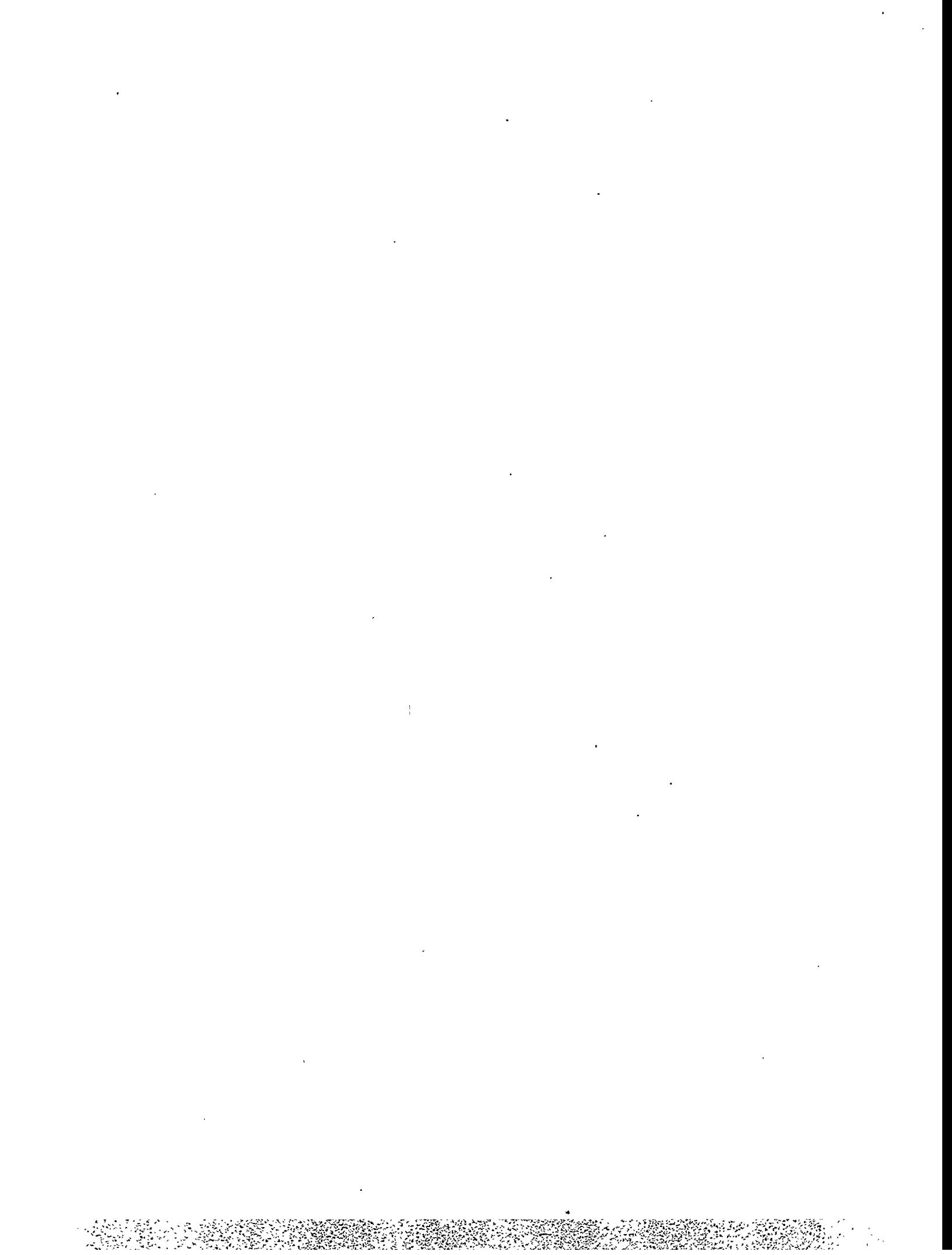
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## THE EQUIVALENT FUNDAMENTAL-MODE SOURCE

by

Gregory D. Spriggs, Robert D. Busch, Takeshi Sakurai, and Shigeaki Okajima

### Abstract

In 1960, Hansen<sup>1</sup> analyzed the problem of assembling fissionable material in the presence of a weak neutron source. Using point kinetics, he defined the weak source condition and analyzed the consequences of delayed initiation during ramp reactivity additions. Although not clearly stated in Hansen's work, the neutron source strength that appears in the weak source condition corresponds to the equivalent fundamental-mode source.

In this work, we describe the concept of an equivalent fundamental-mode source and we derive a deterministic expression for a factor,  $g^*$ , that converts any arbitrary source distribution to an equivalent fundamental-mode source. We also demonstrate a simplified method for calculating  $g^*$  in subcritical systems. And finally, we present a new experimental method that can be employed to measure the equivalent fundamental-mode source strength in a multiplying assembly. We demonstrate the method on the zero-power, XIX-1 assembly at the Fast Critical Assembly (FCA) Facility, Japan Atomic Energy Research Institute (JAERI).

## I. INTRODUCTION

In any multiplying system, prompt fission chains must be initiated by a neutron source. This source can be an external neutron source (e.g., a start-up source), an intrinsic source caused by spontaneous fission or  $(\alpha, n)$  reactions in the fuel, or delayed neutrons produced by precursors generated during previous prompt fission chains. When the reactivity of the system is below prompt critical, each prompt fission chain that is initiated is destined to eventually die out. In fact, even when the system is superprompt critical, most prompt fission chains are still destined to die out; only occasionally will a prompt fission chain actually diverge. In 1960, Wimett et al.<sup>2</sup> performed an experiment which clearly illustrated this effect. Using the Godiva-II burst assembly, 94 superprompt-critical bursts were initiated in which the average *time-to-initiation* was measured to be approximately 3 seconds, with a maximum of approximately 13 seconds. At the time each burst was fired, the effective source strength that includes the delayed neutron contribution, corresponded to approximately 1000 n/s. Therefore, on an average, 3,000 source neutrons appeared in the system prior to the initiation of each burst. Since the probability of any neutron source causing an initial fission in a uranium system is approximately 40%, it follows that an average of 1,200 fission chains were initiated prior to the first persistent chain. In the case of the burst that was delayed by 13 seconds, 13,000 source neutrons appeared in the system, initiating 5,200 prompt fission chains, which ultimately died-out before the first persistent prompt fission chain finally occurred.

In this Godiva experiment, the reactivity of the system was increased to 0.05\$ above prompt critical and held there until the burst was eventually initiated by a source neutron. Imagine a scenario in which reactivity is being ramped into a system with a large excess reactivity. If the neutron source of the system is too weak, then the potential exists to reach a point well above prompt critical before the first persistent chain can be initiated. As a consequence, a serious prompt-criticality accident could occur. In 1960, Hansen<sup>1</sup> analyzed such a scenario in which he defined a *weak source condition* that was conducive for such a scenario to occur.

$$\frac{2Q\tau^+}{\bar{v}_p\Gamma_2} \ll 1 \quad , \quad (1)$$

where  $Q$  is the neutron source strength,  $\tau^+$  is the adjoint-weighted, neutron-removal lifetime,  $\bar{v}_p$  is the average number of prompt neutrons released per fission, and  $\Gamma_2$  is the neutron dispersion factor (also known as the Diven factor).<sup>3,4</sup> Although not clearly stated by Hansen,<sup>1</sup> the neutron source strength,  $Q$ , in Eq. (1) corresponds to the *equivalent fundamental-mode source*.

In other somewhat-related work, Spriggs<sup>5</sup> showed that neutron noise experiments, such as Rossi- $\alpha$  and Feynman's variance-to-mean experiments,<sup>6,7</sup> can be performed in subcritical assemblies if the following condition is satisfied:

$$\left( \frac{2g^* S \tau^+}{g \bar{v}_p \Gamma_2} \right) \left( 1 - \frac{1}{\rho_{\$}} \right) < 100 \quad , \quad (2)$$

where  $g$  is a spatial factor derived from transport theory<sup>8-10</sup> and  $\rho_{\$}$  is the reactivity of the system in the units of  $\$$ . (Note, when the system is subcritical,  $\rho_{\$} < 0$ ; hence, the minus sign in front of the  $1/\rho_{\$}$  term.) The product,  $g^* S$ , in Eq. (2) is synonymous with the equivalent fundamental-mode source,  $Q$ , in Eq. (1) but is expressed in terms of the actual source strength,  $S$ , and a factor,  $g^*$ , that converts the actual source strength to an equivalent fundamental-mode source. The ability to perform neutron noise experiments is a direct indicator that individual prompt fission chains are still discernible; the neutron population has not yet reached a level in which the chains overlap to such an extent that an *apparent* persistent neutron population results. This level does not occur until the mean time between equivalent fundamental-mode source neutrons (including delayed neutrons) is much shorter than the average duration of a prompt fission chain.

The equivalent fundamental-mode source also appears in the reactor point kinetic model and, as such, is contained in many of its solutions.<sup>11</sup> Furthermore, the equivalent fundamental-mode source is particularly important in reactor experimentation. For example, the equivalent fundamental-mode source appears in the *Cf-source technique* originally developed by Carpenter et al.<sup>12</sup> to measure the effective delayed-neutron fraction,  $\beta_{eff}$ , in a test assembly. In this technique, a calibrated point source of strength  $S$  is placed somewhere in the system (usually in the center of the assembly) and the resulting integral fission rate,  $F$ , is measured at some known subcritical reactivity,  $\rho_{\$}$ , just below delayed critical. The effective delayed neutron fraction is related to these quantities by

$$\beta_{eff} = \frac{S \bar{\Psi}_s}{|\rho_s| \bar{v}_t F \bar{\Psi}_f} , \quad (3)$$

where  $\bar{v}_t$  is the average of the total number of neutrons released per fission, and  $\bar{\Psi}_s$  and  $\bar{\Psi}_f$  are the average importance of the source neutrons and the fission neutrons, respectively. As will be shown in this work, the quantity

$$\frac{S \bar{\Psi}_s}{\bar{\Psi}_f} , \quad (4)$$

is the equivalent fundamental-mode source where the ratio of  $\bar{\Psi}_s/\bar{\Psi}_f$  is identically equal to  $g^*$ .

Because of its potential importance in the field of reactor kinetics, criticality safety, reactor start-up operations, and reactor experimentation, it is essential to understand the equivalent fundamental-mode source. In this work, we describe the concept of an equivalent fundamental-mode source, and we derive a deterministic expression for the factor,  $g^*$ , that converts any arbitrary source distribution to an equivalent fundamental-mode source. We also demonstrate a simplified method for calculating  $g^*$  in subcritical systems using deterministic and Monte Carlo methods. And finally, we present a new experimental method that can be used to measure the equivalent fundamental-mode source strength in a multiplying assembly. We demonstrate the method on the zero-power, XIX-1 assembly at the Fast Critical Assembly (FCA) Facility, Japan Atomic Energy Research Institute (JAERI).

## II. THEORY

In a subcritical assembly in equilibrium with a fixed external/intrinsic neutron source, the neutron loss rate must equal the source rate plus the integral fission neutron production rate.

$$\frac{N}{\tau} = S + \int \chi_f \bar{v}_t \Sigma'_f \Phi' d\Omega' dE' d\Omega dV dE , \quad (5)$$

where

$\Phi = \Phi(r, \Omega, E)$  = angular flux,

$\Sigma'_f = \Sigma'_f(r; \Omega', E' \rightarrow \Omega, E)$  = macroscopic fission cross section,

$\bar{v}_t = \bar{v}_t(r, E')$  = the average of the total number of neutrons released per fission,  
 $\chi_f$  = fission spectrum (normalized to 1.0),  
 $N$  = the total, unweighted neutron population, and  
 $\tau$  = the unweighted neutron-removal lifetime.

The neutron loss rate,  $N/\tau$ , in the above expression is the sum of the leakage rate and the absorption rate. In integral form, this can be expressed as

$$\frac{N}{\tau} = \int \Omega \cdot \nabla \Phi \, d\Omega dV dE + \int \Sigma_a \Phi \, d\Omega dV dE , \quad (6)$$

where the total, unweighted neutron population,  $N$ , can be defined in terms of the angular flux,  $\Phi$ , and the neutron velocity,  $v$ , as

$$N = \int \frac{\Phi}{v} \, d\Omega dV dE . \quad (7)$$

From Eqs. (6) and (7), it follows that the unweighted neutron-removal lifetime,  $\tau$ , is defined as

$$\tau = \frac{\int \frac{\Phi}{v} \, d\Omega dV dE}{\int \Omega \cdot \nabla \Phi \, d\Omega dV dE + \int \Sigma_a \Phi \, d\Omega dV dE} . \quad (8)$$

[A more detailed explanation of the definition of a neutron-removal lifetime can be found in Spriggs et al.<sup>13</sup>]

The steady-state multiplication,  $M$ , of a subcritical system is defined as the neutron appearance rate due to the injection of source neutrons and the production of fission neutrons divided by the external/intrinsic neutron source rate,  $S$ .

$$M \equiv \frac{N}{\tau S} . \quad (9)$$

By dividing Eq. (5) by  $S$ , we note that the average number of fission neutrons produced per source neutron is equal to  $M - 1$ . Furthermore, if the source  $S$  is distributed identically to the fission

source distribution (i.e., angle, energy, and space), then the multiplication,  $M$ , will be related to the effective multiplication factor of the system,  $k_{eff}$ , as

$$M = \frac{1}{1 - k_{eff}} . \quad (10)$$

We define the quantity  $1/(1 - k_{eff})$  as the fundamental-mode multiplication,  $M_o$ , since it represents the multiplication that would occur if the source  $S$  were distributed as the fundamental mode fission source. However, in nature, external/intrinsic sources usually occur as uniformly distributed sources, such as intrinsic sources produced by spontaneous fissioning of one or more of the isotopes contained in the fuel, or as point sources that have been placed in or near the assembly, such as an external start-up source. A uniformly distributed intrinsic source or an external point source placed in or near an assembly will produce a system multiplication that can differ significantly from the fundamental-mode multiplication,  $M_o$ .

Because it is customary in reactor physics to express most quantities in terms of the effective multiplication factor,  $k_{eff}$ , we modify Eq. (10) by including a factor,  $g^*$ , that allows us to express the actual multiplication produced by an arbitrary source distribution in terms of the fundamental-mode multiplication. That is,

$$M = g^* M_o = \frac{g^*}{1 - k_{eff}} . \quad (11)$$

We can derive a deterministic expression for  $g^*$  from the steady-state transport equation. When written in terms of the angular flux,  $\Phi$ , the transport equation corresponds to

$$\Omega \cdot \nabla \Phi + \Sigma_t \Phi = \int \Sigma'_s \Phi' d\Omega' dE' + \int \chi_f \bar{v}_t \Sigma'_f \Phi' d\Omega' dE' + \chi_s s \quad (12)$$

where

$\Sigma_t = \Sigma_t(r, E)$  = total macroscopic cross section,

$\Sigma'_s = \Sigma'_s(r; \Omega', E' \rightarrow \Omega, E)$  = macroscopic scattering cross section,

$\chi_s$  = external/intrinsic source spectrum (normalized to 1.0),

$s = s(r, \Omega)$  = source distribution per unit volume per unit angle, and

$$\chi_s s = \chi_s(E) s(r, \Omega) = \text{energy-dependent source distribution.}$$

[As is customary in  $k$ -eigenvalue problems, any  $(n,2n)$ ,  $(n,3n)$ , etc. reactions occurring in the system are accounted for by altering the scattering and absorption cross sections such that

$$\Sigma_s = \Sigma_{so} + 2\Sigma_{2n} + \dots , \quad (13)$$

and

$$\Sigma_a = \Sigma_{ao} - 2\Sigma_{2n} - \dots , \quad (14)$$

thus preserving the total macroscopic cross section defined as  $\Sigma_t = \Sigma_s + \Sigma_a$ .]

Equation (12) is merely a statement of neutron conservation as applied to an infinitesimal element of direction, energy, and space. If it is integrated over all phase space, it becomes a statement of neutron conservation for the integral system.

$$\begin{aligned} & \int \Omega \cdot \nabla \Phi \, d\Omega dV dE + \int \Sigma_t \Phi \, d\Omega dV dE - \int \Sigma'_s \Phi' d\Omega' dE' \, d\Omega dV dE \\ &= \int \chi_f \bar{v}_t \Sigma'_f \Phi' d\Omega' dE' \, d\Omega dV dE + \int \chi_s s \, d\Omega dV dE . \end{aligned} \quad (15)$$

From the definition of the total cross section, the difference between the total interaction rate and the total scattering rate is equal to the total absorption rate:

$$\int \Sigma_t \Phi \, d\Omega dV dE - \int \Sigma'_s \Phi' d\Omega' dE' \, d\Omega dV dE = \int \Sigma_a \Phi \, d\Omega dV dE , \quad (16)$$

where we again stress that the absorption cross section,  $\Sigma_a$ , includes all reactions except the  $(n,2n)$ ,  $(n,3n)$ , etc. reactions multiplied by their respective multiplicities. Hence, we can rewrite Eq. (15) as

$$0 = P - L + S , \quad (17)$$

where  $P$  represents the unweighted neutron production rate due to fission,

$$P = \int \chi_f \bar{v}_t \Sigma'_f \Phi' d\Omega' dE' \, d\Omega dV dE , \quad (18)$$

$L$  represents the unweighted neutron loss rate due to leakage and absorption (i.e.,  $N/\tau$ ),

$$L = \int \Omega \cdot \nabla \Phi \, d\Omega dV dE + \int \Sigma_a \Phi \, d\Omega dV dE , \quad (19)$$

and  $S$  is the unweighted external/intrinsic source rate,

$$S = \int \chi_s s \, d\Omega dV dE . \quad (20)$$

Although Eq. (15) represents a neutron balance for the system, it does not take into account the importance in the multiplication process of a source or fission neutron born at a certain angle, at a certain energy, or at a certain location in space. Therefore, implicit in Eq. (15) is the assumption that all neutrons have equal importance even though neutrons born in the center of an assembly have a much greater probability of causing multiplication in the system than do neutrons born at the outer edge. To include the effect of neutron importance in the transport equation, Eq. (15) is multiplied by the adjoint angular flux,  $\Psi(r, \Omega, E)$ , which is a direct measure of a neutron's importance in the multiplication process. The adjoint angular flux satisfies the equation

$$-\Omega \cdot \nabla \Psi + \Sigma_t \Psi = \int \Sigma_s \Psi' d\Omega' dE' + \int \chi_f \bar{v}_t \Sigma_f \Psi' d\Omega' dE' . \quad (21)$$

Integrating the adjoint-weighted transport equation over all phase space leads to an equation of the form

$$0 = P^+ - L^+ + S^+ , \quad (22)$$

where  $P^+$  is the adjoint-weighted neutron production rate due to fission,

$$P^+ = \int \Psi \chi_f \bar{v}_t \Sigma_f \Phi' d\Omega' dE' \, d\Omega dV dE , \quad (23)$$

$L^+$  is the adjoint-weighted neutron loss rate,

$$L^+ = \int \Psi \Omega \cdot \nabla \Phi \, d\Omega dV dE + \int \Psi \Sigma_a \Phi \, d\Omega dV dE , \quad (24)$$

and  $S^+$  is the adjoint-weighted neutron source rate,

$$S^+ = \int \Psi \chi_s s \, d\Omega dV dE. \quad (25)$$

Dividing Eq. (22) by  $P^+$ , we obtain

$$0 = 1 - \frac{L^+}{P^+} + \frac{S^+}{P^+}. \quad (26)$$

If both the forward and adjoint fluxes in Eqs. (23), (24), and (25) are obtained from a  $k$ -eigenvalue solution, then the ratio  $P^+/L^+$  is identically equal to  $k_{eff}$ . Therefore, Eq. (26) can be rewritten as

$$0 = k_{eff} - 1 + \frac{k_{eff} S^+}{P^+}. \quad (27)$$

Next, we multiply and divide the last term on the right-hand side of Eq. (27) by the unweighted quantities  $S$ ,  $P$ , and  $N$ . This manipulation leads to

$$0 = k_{eff} - 1 + \left[ \frac{PS^+}{SP^+} \right] \left[ \frac{k_{eff} N}{P} \right] \left[ \frac{S}{N} \right]. \quad (28)$$

As also pointed out by Spriggs et al.,<sup>13</sup> the unweighted removal lifetime,  $\tau$ , from Eq. (8) can also be written in terms of the fission production rate by noting that  $k_{eff}$  times the neutron loss rate equals the neutron production rate.

$$k_{eff} \frac{N}{\tau} = \int \chi_f \bar{v}_t \Sigma_f \Phi' d\Omega' dE' d\Omega dV dE = P. \quad (29)$$

Hence, the unweighted removal lifetime, written in terms of the unweighted neutron population and the unweighted fission production rate, is

$$\tau = \frac{k_{eff} N}{P}. \quad (30)$$

Using the above definition of  $\tau$ , Eq. (28) reduces to

$$N = \frac{\tau g^* S}{1 - k_{eff}} , \quad (31)$$

where  $g^*$  is defined as

$$g^* = \frac{PS^+}{SP^+} . \quad (32)$$

Written in terms of the angular fluxes,  $g^*$  corresponds to

$$g^* = \frac{\int \Psi \chi_s s d\Omega dV dE}{\int \chi_s s d\Omega dV dE} \times \frac{\int \chi_f \bar{v}_t \Sigma_f \Phi' d\Omega' dE' d\Omega dV dE}{\int \Psi \chi_f \bar{v}_t \Sigma_f \Phi' d\Omega' dE' d\Omega dV dE} . \quad (33)$$

The physical meaning of  $g^*$  is easily seen by noting that the ratio of  $S^+/S$  is equal to the average importance of a source neutron, and the ratio of  $P^+/P$  is the average importance of a fission neutron. Consequently,  $g^*$  is the ratio of the average importance of a source neutron to the average importance of a fission neutron.

$$g^* = \frac{\overline{\Psi_s}}{\overline{\Psi_f}} . \quad (34)$$

Also note that we can rearrange Eq. (31) as

$$\frac{N}{\tau g^* S} = \frac{1}{1 - k_{eff}} = M_o , \quad (35)$$

which shows that when the neutron loss rate is divided by the product  $g^* S$ , we obtain the fundamental-mode multiplication,  $M_o$ . Hence, the product  $g^* S$  is defined to be the equivalent fundamental-mode source strength,  $Q$ . Obviously, when the source is distributed as the fission-source distribution,  $g^*$  will be identically equal to 1.0, and the actual source distribution will produce the same multiplication as a fundamental-mode source.

From Eq. (33), we also note that when multiple sources are in the system, the equivalent fundamental-mode source is the sum of the equivalent fundamental-mode sources of each of the constituent sources.

$$Q = g^* S = g_1^* S_1 + g_2^* S_2 + \dots . \quad (36)$$

A separate solution for  $g_i^*$  can be obtained for each source constituent. The effective  $g^*$  of a multiple-source distribution is simply a source-weighted average of the individual values of  $g_i^*$  for each of the constituent sources.

$$g^* = \frac{\sum_i g_i^* S_i}{\sum_i S_i} . \quad (37)$$

Although Eq. (31) is written in terms of the unweighted quantities  $N$ ,  $\tau$ , and  $S$ , we could just as easily have multiplied and divided the last term of the right-hand side of Eq. (27) by the adjoint-weighted neutron population,  $N^+$ , and the unweighted source,  $S$ , to obtain

$$\frac{N^+}{\tau^+} = \frac{\bar{\Psi}_s S}{1 - k_{eff}} , \quad (38)$$

where  $\tau^+$  is the adjoint-weighted removal lifetime (see Spriggs et al.<sup>13</sup>) defined by

$$\tau^+ = \frac{k_{eff} N^+}{P^+} . \quad (39)$$

The quantity  $N^+/\tau^+$  is equal to the adjoint-weighted neutron loss rate and, by comparison with Eq. (31), is equal to the average importance of a fission neutron times the unweighted loss rate.

$$\frac{N^+}{\tau^+} = \bar{\Psi}_f \frac{N}{\tau} . \quad (40)$$

It follows, therefore, that the equivalent fundamental-mode source,  $g^* S$ , is the adjoint-weighted source,  $S^+ = \bar{\Psi}_s S$ , divided by the average importance of a fission neutron. When the adjoint fluxes are normalized such that  $\bar{\Psi}_f = 1$ , then the adjoint-weighted source strength is identically equal to the equivalent fundamental-mode source.

We can readily derive Carpenter's equation<sup>12</sup> used to measure the effective delayed-neutron fraction by substituting Eq. (40) into Eq. (38). This substitution yields

$$\bar{\Psi}_f \frac{N}{\tau} = \frac{S \bar{\Psi}_s}{1 - k_{eff}} , \quad (41)$$

which, by Eq. (29), is equal to

$$\bar{\Psi}_f \frac{P}{k_{eff}} = \frac{S \bar{\Psi}_s}{1 - k_{eff}} . \quad (42)$$

Hence, the neutron production rate due to fission,  $P$ , can be written as

$$P = \frac{k_{eff} S \bar{\Psi}_s}{(1 - k_{eff}) \bar{\Psi}_f} . \quad (43)$$

By multiplying and dividing the right-hand side of this equation by  $\beta_{eff}$  and noting that  $P$  is equal to  $\bar{v}_i$  times the integral fission rate,  $F$ , we obtain

$$\bar{v}_i F = \frac{S \bar{\Psi}_s}{|\rho_s| \beta_{eff} \bar{\Psi}_f} . \quad (44)$$

Upon rearrangement of Eq. (44), we obtain Eq. (3). Although Carpenter did not explicitly state this, it is assumed that the source placed in the test assembly during this measurement will be much, much stronger than the intrinsic source. In those cases in which this condition is not satisfied, Eq. (3) must be rewritten to include the equivalent fundamental-mode of the intrinsic source as well. In terms of  $g^*$  for the intrinsic source,  $g_i^*$ , and  $g^*$  for the external point source placed in the assembly during the measurement,  $g_p^*$ , Eq. (3) becomes

$$\beta_{eff} = \frac{g_i^* S_i + g_p^* S_p}{|\rho_S| \bar{v}_t F} . \quad (45)$$

### III. CALCULATION OF $g^*$

In principle, the evaluation of Eq. (33) is rather straightforward when using a deterministic code to estimate the forward and adjoint angular fluxes from a  $k$ -eigenvalue solution. One can easily develop a post-processor code that will perform the integrals in Eq. (33).

For subcritical systems, a simpler method exists. In most deterministic and Monte Carlo codes, the user has the additional option to perform a *fixed-source* solution. From this solution, one obtains the actual multiplication of the system,  $M$ , corresponding to the specified source distribution. When combined with a  $k$ -eigenvalue solution,  $g^*$  can be determined directly from

$$g^* = M(1 - k_{eff}) . \quad (46)$$

We now demonstrate this simplified method with the following numerical example.

Consider a bare, spherical system comprised of 51.2 kg of  $^{235}U$  and 2.75 kg of  $^{238}U$  and containing a 100 n/s  $^{252}Cf$  start-up source in the center of the assembly. Using the deterministic transport code ONEDANT<sup>14</sup> and the original 16-group Hansen-Roach cross-section set,<sup>15</sup> the effective multiplication factor,  $k_{eff}$ , of this system was calculated to be 0.9914 when the outer radius of the sphere was 8.85 cm. It is well known that both  $^{235}U$  and  $^{238}U$  undergo spontaneous fission;  $^{235}U$  produces 0.01 n/s per kg and  $^{238}U$  produces 13.6 n/s per kg (see Table I). For this system, the  $^{235}U$  produces a total of 0.5 n/s, which are uniformly distributed over the volume of the assembly, and the  $^{238}U$  produces a total of 37.4 n/s, which are also uniformly distributed over the volume of the assembly. The spontaneous fission spectra for  $^{235}U$ ,  $^{238}U$ , and  $^{252}Cf$  are listed in Table II.

The equivalent fundamental-mode source for this fictitious system was determined by running a  $k$ -eigenvalue solution to determine  $k_{eff}$ , and three fixed-source problems: 1) a  $^{235}U$  spontaneous fission source distributed uniformly over the volume of the assembly, 2) a  $^{238}U$  spontaneous fission source distributed uniformly over the volume of the assembly, and 3) a  $^{252}Cf$  point source located at the center of the assembly. Because the  $^{235}U$  and  $^{238}U$  spontaneous fission spectra are

TABLE I: SPONTANEOUS FISSION DATA

Isotope	Spont. Fission Half-Life (y) <sup>16,17</sup>	$\bar{v}_p^{18-23}$	$S \left( \frac{n}{s \cdot \text{Kg}} \right)$	Watt Parameters <sup>a</sup>	
				A	B
<sup>232</sup> Th	$7.82 \times 10^{20}$	$2.14 \pm 0.20$	$1.56 \times 10^{-4}$	0.5934	8.030
<sup>233</sup> U	$2.65 \times 10^{17}$	$\sim 1.76$	0.38	0.8548	4.032
<sup>234</sup> U	$1.49 \times 10^{16}$	$\sim 1.81$	6.83	0.7712	4.925
<sup>235</sup> U	$1.00 \times 10^{19}$	$\sim 1.86$	0.01	0.7747	4.852
<sup>236</sup> U	$2.49 \times 10^{16}$	$1.66 \pm 0.11$	3.72	0.7352	5.358
<sup>237</sup> U	$9.11 \times 10^{17}$	$\sim 1.87$	0.12	0.6931	5.994
<sup>238</sup> U	$8.20 \times 10^{15}$	$2.00 \pm 0.02$	13.6	0.6483	6.811
<sup>239</sup> U	$5.55 \times 10^{16}$	$\sim 2.04$	2.03	0.7356	5.261
<sup>236</sup> Pu	$2.09 \times 10^{09}$	$2.12 \pm 0.14$	$5.70 \times 10^{07}$	0.9883	3.104
<sup>237</sup> Pu	$2.05 \times 10^{13}$	$\sim 1.88$	$5.09 \times 10^{03}$	0.9546	3.308
<sup>238</sup> Pu	$4.74 \times 10^{10}$	$2.21 \pm 0.06$	$2.59 \times 10^{06}$	0.8478	4.169
<sup>239</sup> Pu	$8.05 \times 10^{15}$	$2.24 \pm 0.10$	15.5	0.8853	3.803
<sup>240</sup> Pu	$1.14 \times 10^{11}$	$2.151 \pm 0.005$	$1.04 \times 10^{06}$	0.7949	4.689
<sup>241</sup> Pu	$5.98 \times 10^{16}$	$\sim 2.25$	2.07	0.8425	4.152
<sup>242</sup> Pu	$6.74 \times 10^{10}$	$2.141 \pm 0.006$	$1.74 \times 10^{06}$	0.8192	4.367
<sup>243</sup> Pu	$2.00 \times 10^{15}$	$\sim 2.43$	66.2	0.7354	5.387
<sup>244</sup> Pu	$6.68 \times 10^{10}$	$2.29 \pm 0.19$	$1.86 \times 10^{06}$	0.6947	6.004
<sup>252</sup> Cf	$8.55 \times 10^{01}$	$3.768 \pm 0.012$	$2.31 \times 10^{15}$	1.0250	2.926

a. Obtained by extracting A and B from the Los Alamos Model,<sup>24-26</sup> where A and B are defined by

$$f(E) = C \exp\left(-\frac{E}{A}\right) \sinh \sqrt{BE} .$$

This yields a S. F. spectrum with an average neutron energy that is accurate to within  $\pm 10\%$  in most cases.

TABLE II: SPONTANEOUS FISSION SPECTRUM

Group <sup>a</sup>	Energy Range	<sup>235</sup> U	<sup>238</sup> U	( $\alpha, n$ ) <sup>b</sup>	<sup>252</sup> Cf
1	3 - $\infty$ Mev	0.186	0.140	0.011	0.275
2	1.4 - 3 Mev	0.364	0.362	0.307	0.353
3	0.9 - 1.4 Mev	0.174	0.190	0.036	0.149
4	0.4 - 0.9 Mev	0.179	0.200	0.237	0.146
5	0.1 - 0.4 Mev	0.083	0.093	0.342	0.067
6	17 - 100 kev	0.013	0.014	0.064	0.010
7	3 - 17 kev	0.001	0.001	0.003	0.000
8-16	3 kev & $\downarrow$	0.0	0.0	0.0	0.0
Average Energy =		1.89 Mev	1.69 Mev	1.01 Mev	2.31 Mev

- a. Corresponding to Hansen-Roach group structure.<sup>15</sup>
- b. This spectrum was calculated specifically for the soft blanket region of the XIX-1 core using the SOURCES code.<sup>27,28</sup>

similar, the multiplication calculated by ONEDANT was found to be 98.76 for both source distributions. Hence,  $g^*$  for both the <sup>235</sup>U and <sup>238</sup>U sources corresponds to

$$g^* = 98.76(1 - 0.9914) = 0.85 .$$

The multiplication produced by the centrally located <sup>252</sup>Cf point source was calculated to be 204.7. So,  $g^*$  for the <sup>252</sup>Cf point source corresponds to

$$g^* = 204.7(1 - 0.9914) = 1.76 .$$

When combined, the equivalent fundamental-mode source for this particular source distribution corresponds to

$$Q = 0.85(0.5 + 37.4) + 1.76(100) = 208.2 \text{ n/s} ,$$

with an effective  $g^*$  of

$$g^* = \frac{208.2}{0.5 + 37.4 + 100} = 1.51$$

From a physical standpoint we can interpret these results as follows. A uniformly distributed  $^{235}\text{U} + ^{238}\text{U}$  spontaneous fission source producing 37.9 n/s in this spherical assembly will produce the same total neutron multiplication as an equivalent fundamental-mode source of strength 32.2 n/s (i.e.,  $0.85 \times 37.9$ ). A  $^{252}\text{Cf}$  point source emitting 100 n/s in the center of the assembly will produce the same total neutron multiplication as an equivalent fundamental-mode source of strength 1.76 n/s (i.e.,  $1.76 \times 100$ ). The combined sources ( $^{235}\text{U} + ^{238}\text{U} + ^{252}\text{Cf}$ ) of 137.9 n/s will produce the same total neutron multiplication as an equivalent fundamental-mode source of strength 208.2 n/s.

#### IV. $k_{\text{eff}}$ -DEPENDENCE OF $g^*$

The factor  $g^*$  is dependent on  $k_{\text{eff}}$ . However, depending on how the source is distributed, this dependence may be relatively weak. For example, using the assembly described in the previous section, the radius of the uranium sphere was varied from 8.85 cm to 0.85 cm in 1-cm decrements;  $k_{\text{eff}}$  varied from 0.9914 to 0.0933, respectively. As can be seen in Fig. 1,  $g^*$  for the uniformly distributed spontaneous fission source varies from 0.85 to 0.99 as  $k_{\text{eff}}$  decreases. In comparison,  $g^*$  for a  $^{252}\text{Cf}$  point source located at the center of the assembly decreases from 1.76 to 1.03 over the same range in  $k_{\text{eff}}$ .

As another example, we calculated  $g^*$  for a reflected system in which the uranium core used in the previous example was surrounded by a spherical graphite reflector 12 cm in radius. At a core radius of 7.0 cm,  $k_{\text{eff}} = 0.9842$ . When the core radius was reduced to 1.0 cm (while maintaining the outer radius of the graphite at 12 cm),  $k_{\text{eff}}$  decreased to 0.1136. For the case of a uniformly distributed source,  $g^*$  increased from 0.949 to 0.995 as  $k_{\text{eff}}$  decreased. For the case of a point source located at the center of the assembly,  $g^*$  decreased from 1.49 to 1.04 (see Fig. 1).

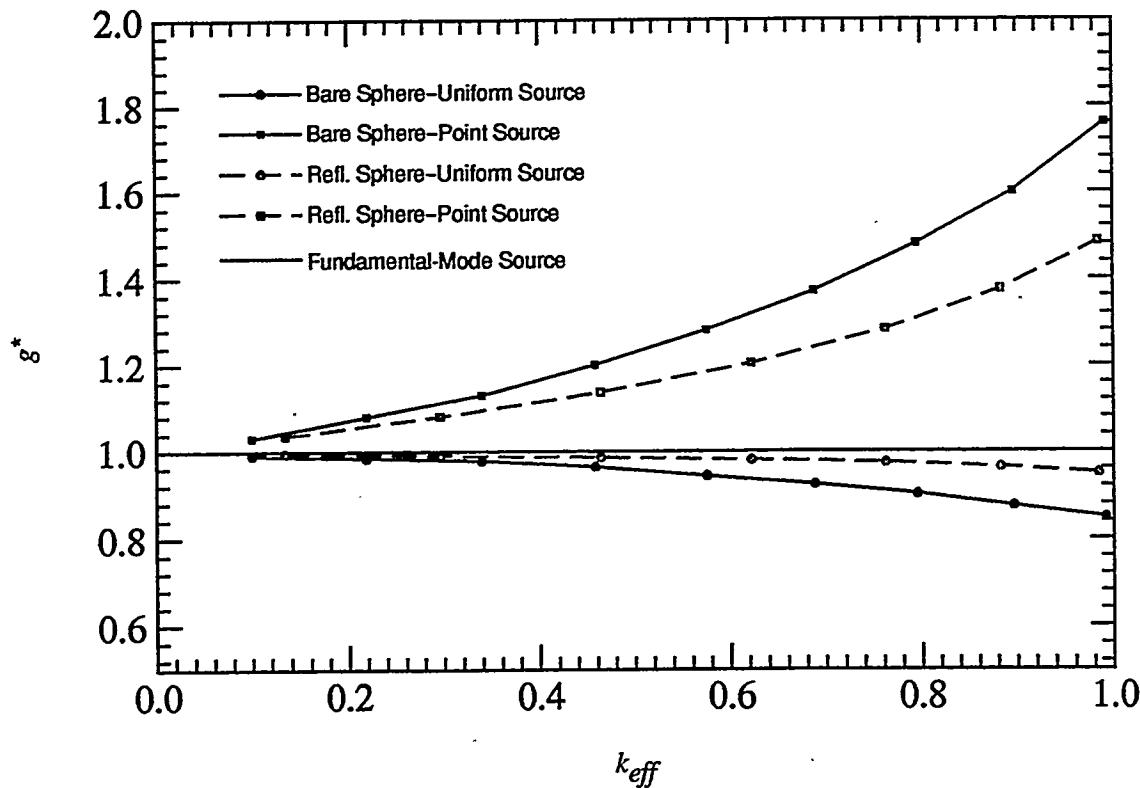


Fig. 1.  $k_{eff}$ -dependence of  $g^*$  for four different cases: 1) a bare sphere with a uniformly distributed spontaneous fission source, 2) a bare sphere with a point source at the center of the assembly, 3) a reflected sphere with a uniformly distributed spontaneous fission source, and 4) a reflected sphere with a point source at the center of the assembly.

## V. DEPENDENCE OF $g^*$ ON THE SOURCE SPECTRUM

As expected,  $g^*$  is also dependent on the energy spectrum of the source. However, this dependence can be relatively weak in some systems. To illustrate this, consider the same assembly described in Section IV in which a point source is placed in the center of the system. If we assume that the energy spectrum of this point source corresponds to that of  $^{252}\text{Cf}$ ,  $^{235}\text{U}$ , or  $^{238}\text{U}$  (see Table II), then  $g^*$  calculates to be 1.76, 1.754, and 1.75, respectively. These changes in  $g^*$  are surprisingly small when you consider the differences in the average neutron energy of the three

different sources. In terms of the Watt parameters,  $A$  and  $B$ , the average energy of each spectrum is given by<sup>17</sup>

$$\bar{E} = A \left( \frac{3}{2} + \frac{AB}{4} \right) . \quad (47)$$

Using the values of  $A$  and  $B$  from Table I,  $^{252}\text{Cf}$  has an average energy of 2.31 Mev, whereas,  $^{235}\text{U}$  and  $^{238}\text{U}$  have average energies of 1.89 Mev and 1.69 Mev, respectively.

What is even more surprising is the fact that  $g^*$  increases as the average energy of the source increases. [In fact, if one assumes that all source neutrons are born in group 1, which would represent a very hard-spectrum source in which all neutron energies are greater than 3 Mev,  $g^*$  increases to 1.84.] Although we cannot positively state that these results are wrong, they do seem to contradict intuition as they imply that a hard-spectrum source multiplies the assembly more effectively than a soft-spectrum source. One would think that as the source spectrum becomes softer in a fast system (such as the one used in this example), an average source neutron should have a smaller probability of leaking from the system and therefore have a higher probability of being absorbed in a fission reaction. Hence, the multiplication caused by a soft-spectrum source should be higher than the multiplication caused by a relatively harder-spectrum source. In fact, when we assume the point source to have an energy spectrum corresponding to the  $(\alpha, n)$  source shown in Table II,  $g^*$  increases to 1.85—which indicates that the soft-spectrum  $(\alpha, n)$  source multiplies the assembly more effectively than the relatively harder-spectrum  $^{252}\text{Cf}$  source.

Without experimental data to substantiate our logic, we can only speculate as to whether the behavior of  $g^*$  as a function of average source energy is real or not. In all likelihood, the observed minimum in calculated  $g^*$  is probably an artifact of imperfections in the cross-section set in the high-energy groups.

## VI. MEASUREMENT OF THE EQUIVALENT FUNDAMENTAL-MODE SOURCE

The equivalent fundamental-mode source can be easily measured in a multiplying system. This is accomplished by placing a neutron detector in or somewhere near the core and observing the change in the count rate produced by placing a *calibrated* neutron source at some known

location within the system. The count rate of the detector is proportional to the unweighted neutron loss rate and the detector efficiency,  $\varepsilon$ , where  $\varepsilon$  is defined as

$$\varepsilon = \frac{C}{\left(\frac{N}{\tau}\right)} , \quad (48)$$

and  $C$  is the detector count rate.

Based on the above definition of the detector efficiency, we can rewrite Eq. (31) as

$$C = \frac{\varepsilon g^* S}{1 - k_{eff}} . \quad (49)$$

Assume for the moment that we wish to measure the equivalent fundamental-mode source strength of an intrinsic source present in a critical assembly containing a large amount of one or more isotopes that undergo spontaneous fission. If the system reactivity is adjusted to be just slightly subcritical, then the detector count rate produced by the intrinsic source distribution will correspond to

$$C_i = \frac{\varepsilon g_i^* S_i}{1 - k_{eff}} , \quad (50)$$

where  $C_i$  is the detector count rate at that subcritical configuration and  $S_i$  is the intrinsic source strength.

If we then place a calibrated point source at some known location within that system and we reestablish the same subcritical configuration, the detector count rate produced by the intrinsic source plus the point source will now correspond to

$$C_{pi} = \frac{\varepsilon(g_i^* S_i + g_p^* S_p)}{1 - k_{eff}} , \quad (51)$$

where  $C_{pi}$  is the new count rate,  $S_p$  is the strength of the point source, and  $g_p^*$  corresponds to the  $g^*$ -value at the location of the point source.

Taking the ratio of Eq. (51) to Eq. (50) and solving for  $g_i^* S_i$  leads to

$$g_i^* S_i = \frac{g_p^* S_p}{\left( \frac{C_{pi}}{C_i} - 1 \right)} . \quad (52)$$

Note that we have assumed that the efficiency of the detector does not change upon insertion of the point source. In far subcritical systems, this assumption may not be valid. However, if the system is just slightly subcritical, then the multiplication of the system will be high enough to excite the fundamental mode. When this occurs, the flux distribution will be determined by the fission source distribution and will be relatively insensitive to the actual source distribution. Therefore, the detector efficiency will remain essentially constant for both source distributions used during the measurements.

Furthermore, in the above derivation, we have optimistically assumed that the observed count rates,  $C_i$  and  $C_{pi}$ , are produced only by neutrons. However, most neutron detectors are somewhat sensitive to gamma rays—which are always present in a multiplying system. When using fission chambers, this sensitivity can be effectively eliminated by adjusting the lower-limit discriminator to be high enough to detect only those pulses that are produced by the fission fragments generated within the chamber when a neutron is detected. When using  $^3\text{He}$  or  $\text{BF}_3$  detectors, it is much more difficult to discriminate out all gamma rays without simultaneously reducing the ability to detect neutrons as well. Consequently, one must be able to determine the gamma ray background count rate,  $C_\gamma$ , to obtain the correct values for  $C_i$  and  $C_{pi}$  to be used in Eq. (52). This can be accomplished if these measurements are repeated at several different subcritical configurations in the vicinity of delayed critical. Eq. (52) can be modified as follows.

When written in terms of reactivity [i.e.,  $\rho_{\$} = (k_{eff} - 1)/\beta k_{eff}$ ], Eq. (49) becomes

$$C = \varepsilon g^* S - \frac{\varepsilon g^* S}{\beta \rho_{\$}} \quad (53)$$

where  $\beta$  is the effective delayed neutron fraction. When  $\rho_{\$} \rightarrow -\infty$  (i.e.,  $k_{eff} = 0.0$ ), the count rate must approach  $\varepsilon g^* S$ , which is the count rate that would be produced by an unmultiplied source in that particular source/detector geometry. If we define  $\varepsilon g^* S$  as  $C_o$ , we can write Eq. (53) as

$$C = C_o - \frac{m}{\rho_s} \quad (54)$$

where  $m$  is  $\epsilon g^* S / \beta$ . Obviously,  $C_o = \beta m$  and since  $\beta$  is no larger than 1%, we expect  $C_o$  to be negligible relative to  $-m/\rho_s$  when  $\rho_s$  is in the vicinity of delayed critical. Therefore, to a first approximation, we can state that

$$C_i - C_{io} = -\frac{m_i}{\rho_s} \approx C_i \quad (55)$$

and

$$C_{pi} - C_{pio} = -\frac{m_{pi}}{\rho_s} \approx C_{pi} \quad (56)$$

From these two equations, we can see that the ratio  $C_{pi}/C_i$  is approximately equal to the ratio of  $m_{pi}/m_i$ . Therefore, Eq. (52) can also be written as

$$g_i^* S_i = \frac{g_p^* S_p}{\left(\frac{m_{pi}}{m_i} - 1\right)} \quad (57)$$

The slopes  $m_{pi}$  and  $m_i$  can be readily determined by plotting the detector count rate as a function of the inverse reactivity of the system and then performing a least-squares fit of these data to determine the y-intercept,  $C_o$ , and the slope,  $m$ , for both source configurations. If the detector system happens to be sensitive to gamma rays, then  $C_o$  will be noticeably greater than  $\beta m$ . When this occurs, we can infer the gamma ray background to be

$$C_\gamma = C_o - \beta m \quad (58)$$

However, determining the gamma ray background is for information purposes only. When evaluating Eq. (57), we presume (albeit, somewhat optimistically) that the gamma ray background does not change much with the reactivity of the system over a small range of reactivity in the

vicinity of delayed critical. We expect the slopes  $m_{pi}$  and  $m_i$  to be relatively independent of the background count rate, whatever it may be. We now illustrate this technique on a real system.

## VII. INTRINSIC SOURCE MEASUREMENT AND CALCULATION FOR THE XIX-1 CORE

### A. Measurement

The experimental procedure described in the previous section was performed on the zero-power XIX-1 assembly located at the Fast Critical Assembly (FCA) facility operated by the Japan Atomic Energy Research Institute (JAERI). The XIX-1 assembly is a multiregion system comprised of an inner core fueled with highly-enriched  $^{235}\text{U}$  metal. The core is surrounded by an inner blanket (referred to as the 'soft blanket') containing a significant amount of depleted uranium-oxide and sodium, and an outer blanket (referred to as the 'depleted blanket') containing only depleted uranium metal. Cross-sectional views of the core are shown in Figs. 2, 3, and 4.

For this measurement, four  $^3\text{He}$  detectors were inserted into the soft blanket region of the assembly (see Fig. 2). Count rates from these detectors were obtained at three different subcritical configurations when the system was driven by just the intrinsic source. The reactivity ranged from  $-0.072\$$  to  $-0.892\$$ . A calibrated  $^{252}\text{Cf}$  source was then inserted into the center of the assembly and the count rates from the  $^3\text{He}$  detectors were obtained at eight different subcritical configurations over the same reactivity range. A least-squares fit of count rate vs. inverse reactivity was performed on these two sets of data.

With just the intrinsic source driving the system, it was found that  $m_i = -55.8 \pm 1.3 \$/\text{s}$  and  $(C_{io} + C_{i\gamma}) = 22 \pm 10 \text{ cps}$  (see Fig. 5). The y-intercept is obviously too large; from the measured slope and the measured value of the effective delayed neutron fraction,  $\beta = 0.00729$ , the y-intercept should have been approximately 0.4 cps if the detectors were only counting neutrons. Consequently, from Eq. (58), we can infer that the gamma ray background,  $C_{i\gamma}$ , for these  $^3\text{He}$  detectors during the intrinsic source measurements was approximately 21 cps.

A similar process was repeated for the count rates obtained with the  $^{252}\text{Cf}$  source in the center of the assembly. The least-squares fit yielded  $m_{pi} = -500.6 \pm 3 \$/\text{s}$  and  $(C_{pio} + C_{pi\gamma}) = 39 \pm 25 \text{ cps}$ . This y-intercept is also too high;  $C_{io}$  should have been approximately 3.7 cps. So, the gamma ray background present during the measurements with the Cf source in the center of the assembly increased to approximately 35 cps.

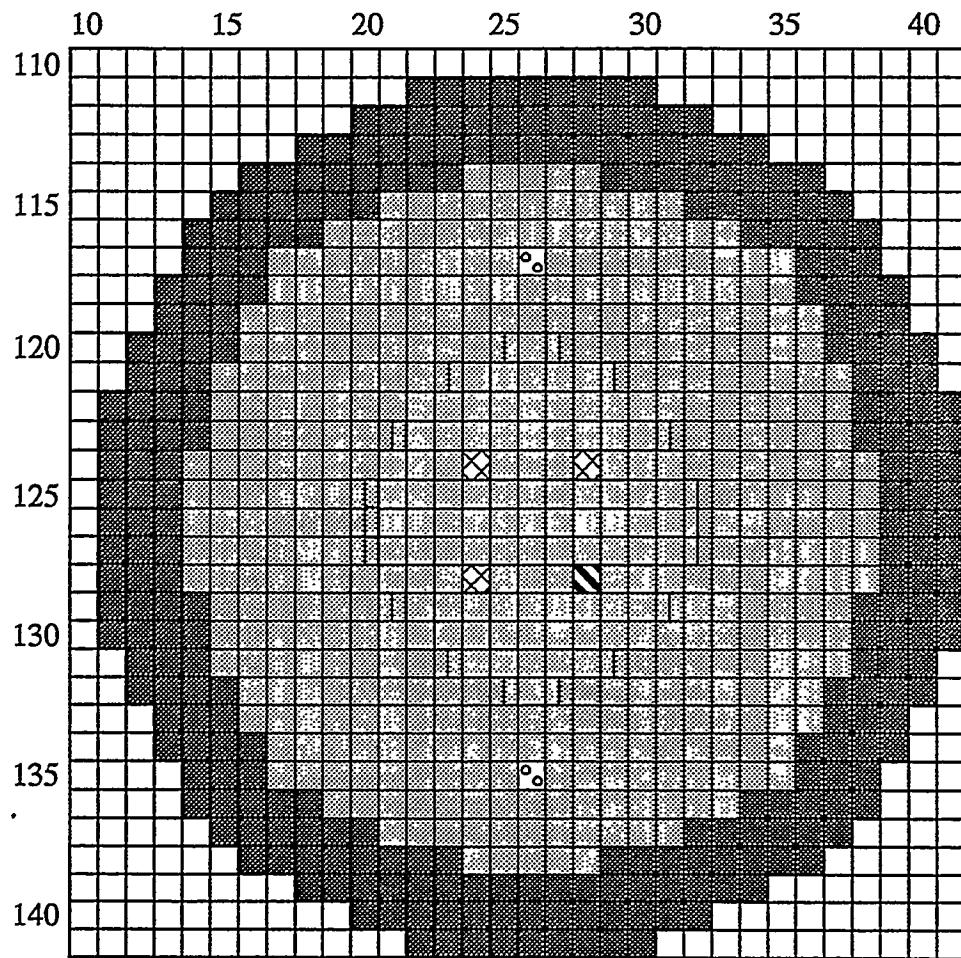


Fig. 2. Cross-sectional view of FCA XIX-1 assembly. Two pairs of He detectors are located in positions 26-118 and 26-135.

The ratio of the slopes is

$$\frac{C_{pi}}{C_i} \approx \frac{m_{pi}}{m_i} = \frac{500.6}{55.8} = 8.97 \pm 2.4\% .$$

Using the procedure described earlier in this report for calculating  $g^*$  in a subcritical assembly,  $g_p^*$  for the XIX-1 assembly was calculated to be 1.68 in the vicinity near delayed critical. And, as determined from a previous source calibration performed at JAERI, the strength of the  $^{252}\text{Cf}$  source used to perform the experiment was measured to be  $97,600 \pm 1\%$  n/s on the day of the

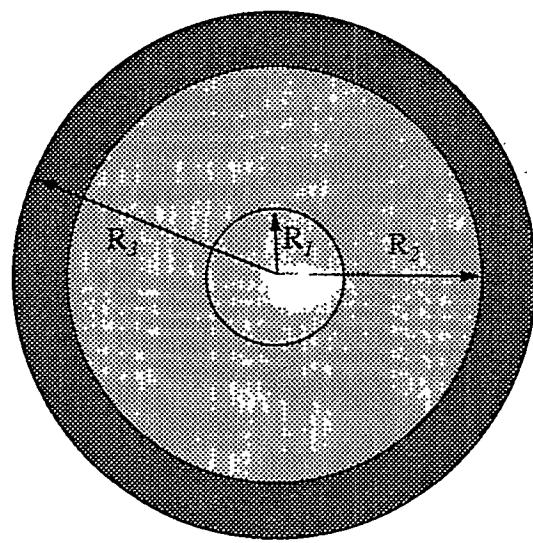


Fig. 3. Effective radii of the three regions of the XIX-1 Assembly.  $R_1 = 32.958$  cm,  $R_2 = 68.30$  cm,  $R_3 = 86.36$  cm.

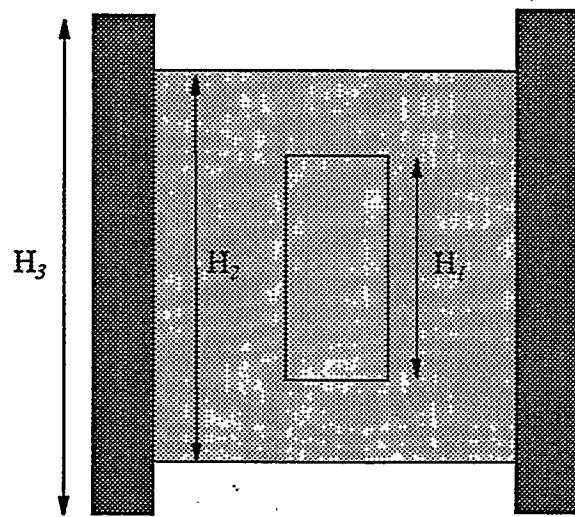


Fig. 4. Effective heights of the three regions of the XIX-1 Assembly.  $H_1 = 50.80$  cm,  $H_2 = 121.92$  cm,  $H_3 = 132.08$  cm.

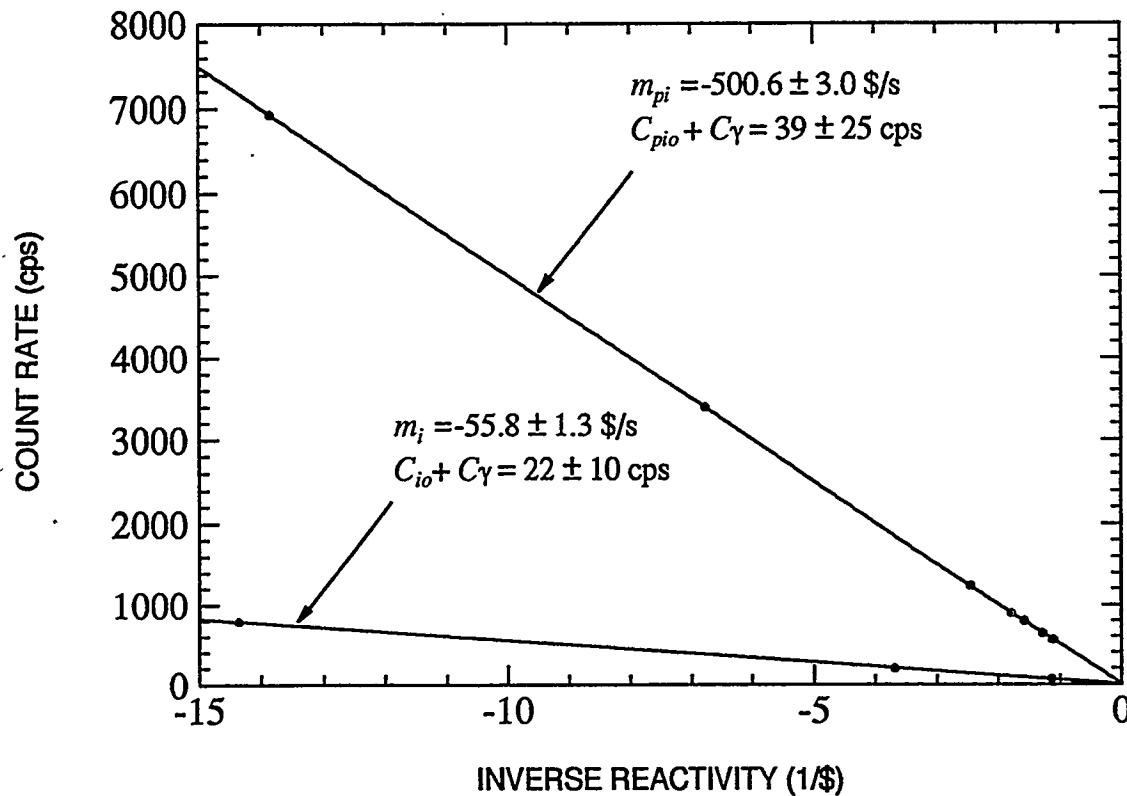


Fig. 5. Plot of the count rate of the four detectors summed together vs. inverse reactivity.

measurement. Substituting these values into Eq. (52) yields an equivalent fundamental-mode *intrinsic* source strength of

$$g_i^* S_i = \frac{1.68 \times 97,600}{8.97 - 1} = 20,600 \pm 3\% \text{ n/s} . \quad (59)$$

## B. Calculation

Using TWODANT,<sup>14</sup> the equivalent fundamental-mode intrinsic source strength was calculated using the data listed in Tables I and II. The intrinsic source was separated into its various constituents so that the neutron spectrum from each spontaneous fission source and the

$(\alpha, n)$  source could be properly accounted for in each region of the assembly. The results of this analysis are shown in Table III.

Although there is a 30% difference between the measured and calculated equivalent fundamental-mode source strength, this difference is not particularly significant for this system. A small increase of 2% in  $g^*$  in the soft blanket (i.e., 0.0896 to 0.1096) and a 2% increase in  $g^*$  in the depleted blanket (i.e., 0.0277 to 0.0477) would increase the calculated effective intrinsic source strength to 20,800 n/s—which would compare very favorably with the measured value. A 2% increase in these two values of  $g^*$  is very plausible considering the fact that the original 16-group Hansen-Roach cross-section set<sup>15</sup> used to perform these calculations is not well suited for this particular system since those cross sections are known to yield biased results. This bias is best seen when comparing the measured and calculated adjoint-weighted neutron lifetimes. Near delayed critical, the adjoint-weighted neutron lifetime was calculated to be 550 ns, which is 21% lower than the measured value of 700 ns. Because the neutron lifetime varies as the inverse of the absorption cross section, it follows that the 16-group Hansen-Roach cross-section set overpredicts

TABLE III: EQUIVALENT FUNDAMENTAL-MODE INTRINSIC SOURCE

Region	Isotope	Mass (Kg)	$S$ (n/s)	$g^*$	$g^*S$ (n/s)
Core	$^{235}\text{U}$	153.0	2	0.994	2
	$^{238}\text{U}$	11.7	159	0.991	158
Soft Blanket	$^{235}\text{U}$	11.7	<0.1	0.0916	0
	$^{238}\text{U}$	5,841.2	79,400	0.0896	7,114
	$(\alpha, n)^a$	—	772	0.0796	61
Depl. Blan.	$^{235}\text{U}$	38.8	<0.4	0.0285	0
	$^{238}\text{U}$	18,713.7	255,000	0.0277	7,064
			$\sum S = 343,160$	$g_i^*S_i = 14,399$	

a. The uranium in the soft blanket is in the form of uranium-oxide plates. The alpha particles from the uranium interact via  $(\alpha, n)$  reactions with other elements in the soft blanket (primarily the  $^{18}\text{O}$  and  $^{23}\text{Na}$ ) to produce an additional source of intrinsic neutrons.

the absorption rate in this particular system. Hence, in accordance to the calculations, source neutrons born in the soft and depleted blankets will have a much harder time penetrating those regions than they would in reality. This shorter mean-free path, in turn, yields a calculated  $g^*$  that will be lower than the actual value.

Finally, we note that the measured value of the effective  $g^*$  corresponds to

$$g^* = \frac{20,600}{343,160} = 0.060 ,$$

which shows that the equivalent fundamental-mode source is only 6% of the total neutron source present in this system.

The XIX-1 assembly is a prime example of the importance of understanding the equivalent fundamental-mode source. If we were to evaluate the weak-source condition using the total source strength present in the system, we would obtain

$$\frac{2S\tau}{\sqrt{\Gamma_2}} = \frac{2 \times 343,160 \times 700 \times 10^{-9}}{2.6 \times 0.80} = 0.23 ,$$

which marginally satisfies the condition of  $\ll 1$ . However, when we use the equivalent fundamental-mode source strength of 20,600 n/s, then the weak-source condition becomes 0.0138, which clearly satisfies the condition of  $\ll 1$ . Hence, the equivalent fundamental-mode source strength for this assembly is clearly not strong enough to ensure consistent behavior of the system during start-up despite the fact that the total source strength is in excess of 340,000 n/s.

### VIII. DISCUSSION

In addition to the applications previously mentioned in this manuscript, the factor  $g^*$  also has an impact on one of the most basic experimental techniques used in the field of nuclear engineering—the classical *approach-to-critical* experiment using a  $1/M$  plot. In this experiment, a source is placed in an assembly, and a reference count is taken with a nearby detector. Then a small amount of fuel is added to the assembly, and a new count is obtained. The ratio of the reference count to the new count (i.e., the inverse of the multiplication,  $M$ ) is plotted as a function of the amount of fuel in the system and then extrapolated to the point where  $1/M = 0.0$  to obtain

an estimate of the critical mass. In principle, if this process is repeated for a series of small mass additions, the  $1/M$  plot will be a linear function of the amount of mass in the system assuming each mass addition produces the same  $\Delta k_{\text{eff}}$ . In general, this is not a very good assumption; if the system is loaded with fuel from the center out, then the initial mass additions will produce a much greater change in  $k_{\text{eff}}$  than will the later mass additions at the outer surface of the assembly.

For the sake of argument, let us assume that we are able to load a new system in such a way as to produce equal changes in  $k_{\text{eff}}$ . Since  $1/M = (1 - k_{\text{eff}})/g^*$ , the  $1/M$  plot will be linear with  $k_{\text{eff}}$  only if  $g^*$  is constant with  $k_{\text{eff}}$ . However, as shown in a previous section, having a constant  $g^*$  over a wide range of  $k_{\text{eff}}$  is highly unlikely. Using the numerical example presented in Section V, when a point source is positioned in the center of that assembly,  $g^*$  increases as the radius of the assembly increases (i.e.,  $k_{\text{eff}}$  increases). As shown in Fig. 6, the  $1/M$  plot for this example is noticeably non-linear with  $k_{\text{eff}}$ . However, in this particular instance, the curve is considered to be conservative from the standpoint that an extrapolation to  $1/M = 0.0$  using any two successive points underestimates the critical mass. Nevertheless, it is conceivable to perform an approach-to-critical experiment in which  $g^*$  decreases as  $k_{\text{eff}}$  increases, thereby, producing a nonconservative estimate of the critical mass. When preparing to perform an approach-to-critical experiment, it would be most helpful to understand how  $g^*$  is going to change as  $k_{\text{eff}}$  increases before the experiment is actually performed.

## IX. CONCLUSIONS

The factor  $g^*$  is the parameter that converts any arbitrary source distribution to an equivalent fundamental-mode source. The equivalent fundamental-mode source, in turn, is the effective source strength that is multiplied by the factor  $1/(1 - k_{\text{eff}})$  to yield the correct neutron production rate in the system ( $k_{\text{eff}}$  in this expression corresponds to the  $k$ -eigenvalue of the system and is, by definition, independent of the source distribution).

The equivalent fundamental-mode source has many applications in reactor kinetics, particularly point kinetics, as well as being an important parameter in criticality safety, reactor operations, and reactor experiments. In subcritical systems, the equivalent fundamental-mode source is easily calculated using deterministic or Monte Carlo methods and can be readily measured in real critical assemblies using the technique described in this work.

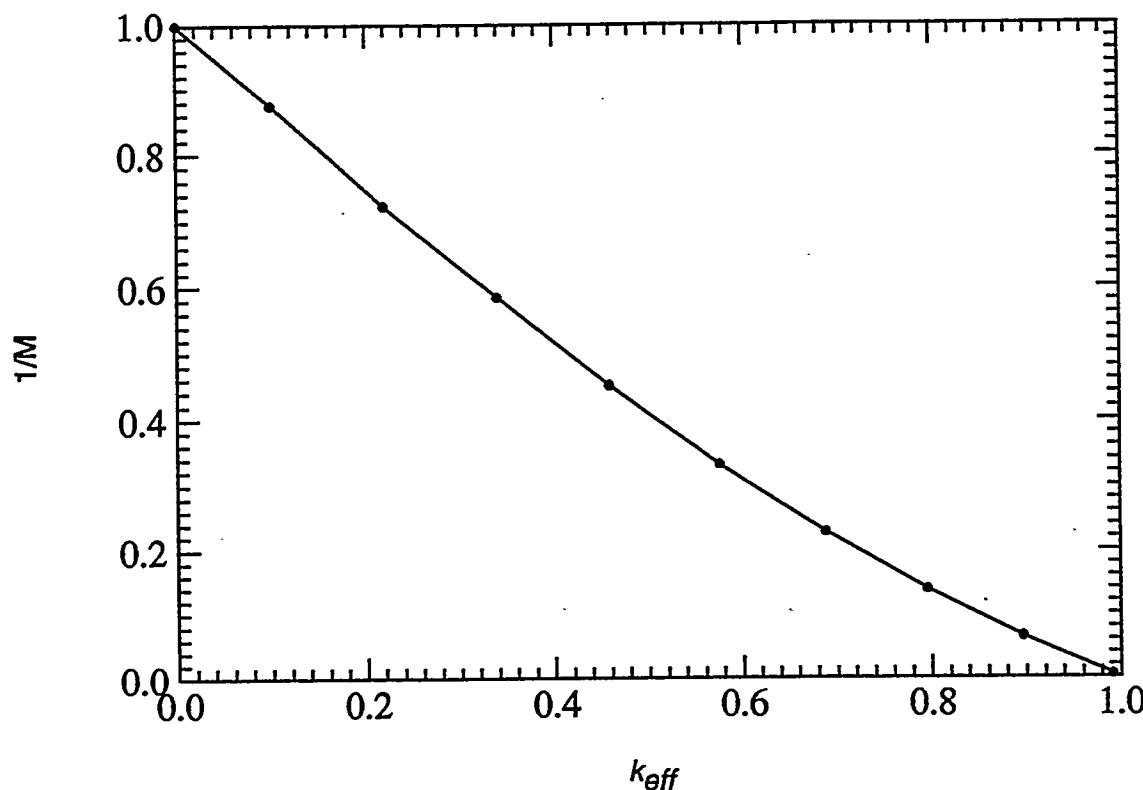


Fig. 6.  $1/M$  plot vs.  $k_{eff}$  for a spherical, uranium assembly with a  $^{252}\text{Cf}$  point source positioned in the center of the assembly.

#### ACKNOWLEDGMENT

The authors would like to thank Dr. Bill Wilson and Dr. David Madland (Los Alamos National Laboratory) for supplying the spontaneous fission and  $(\alpha, n)$  data used in this analysis. We would also like to thank Dr. Murao, Mr. Mukaiyama, and Mr. Osugi (JAERI) for allowing us to use the FCA facility to test our theory and our experimental technique.

#### REFERENCES

1. G. E. Hansen, "Assembly of Fissionable Material in the Presence of a Weak Neutron Source," *Nucl. Sci. Eng.* **8**, 709 (1960).

2. T. F. Wimett, R. H. White, W. R. Stratton, and D. P. Wood, "Godiva-II—An Unmoderated Pulse-Irradiation Reactor," *Nucl. Sci. Eng.* **8**, 691 (1960).
3. B. C. Diven, H. C. Martin, R. F. Taschek, and J. Terrell, "Multiplicities of Fission Neutrons," *Phys. Rev.* **101**, 1012 (Feb. 1956).
4. J. Terrell, "Distribution of Fission Neutron Numbers," *Phys. Rev.* **108**, 1012 (Nov. 1958).
5. G. D. Spriggs, "The Reactor Noise Threshold," *Nucl. Sci. Eng.* **116**, 62 (1994).
6. R. P. Feynman, F. de Hoffmann, and H. Serber, "Statistical Fluctuations in the Water Boiler and the Dispersion of Neutrons Emitted per Fission," Los Alamos Scientific Laboratory report LA-101 (June 1944).
7. R. P. Feynman, "Statistical Behavior of Neutron Chains," Los Alamos Scientific Laboratory report LA-591(del) (July 1946).
8. G. I. Bell, "On Stochastic Theory of Neutron Transport," *Nucl. Sci. Eng.* **21**, 390 (1965).
9. D. R. Harris, "Neutron Fluctuations in a Reactor of Finite Size," *Nucl. Sci. Eng.* **21**, 369 (1965).
10. M. Otsuka and T. Iijima, "Space-Dependent Formula for Rossi- $\alpha$  Measurements," *Nukleonik*, **7**, 488 (1965).
11. D. L. Hetrick, Dynamics of Nuclear Reactors, Chicago Press (1971).
12. S. G. Carpenter, R. J. M. Gasidlo, and J. M. Stevenson, "Measurements of the Effective Delayed-Neutron Fraction in Two Fast Critical Experiments," *Nucl. Sci. Eng.* **49**, 236 (1972).
13. G. D. Spriggs, R. D. Busch, K. J. Adams, D. K. Parsons, L. Petrie, and J. S. Hendricks, "On the Definition of Neutron Lifetimes in Multiplying and Non-Multiplying Systems," Los Alamos National Laboratory report, to be published (1997).
14. R. E. Alcouffe, "DANTSYS: A Diffusion Accelerated Neutral Particle Transport Code System," Los Alamos National Laboratory report LA-12969-M (1995).
15. G. E. Hansen and W. H. Roach, "Six and Sixteen Group Cross Sections for Fast and Intermediate Critical Assemblies," Los Alamos National Laboratory report LAMS-2543 (Dec. 1961).
16. R. B. Firestone et al., Table of Isotopes, John Wiley & Sons, Inc. (1996).
17. C. Wagemans, The Nuclear Fission Process, CRC Press (1991).

18. F. Manero and V. A. Konshin, "Status of the Energy-Dependent  $\bar{\nu}$ -Values for the Heavy Isotopes ( $Z > 90$ ) from Thermal to 15 Mev and of  $\bar{\nu}$ -Values for Spontaneous Fission," *At. Energy Rev.* **10**, 637 (1972).
19. D. L. Johnson, "Evaluation of Neutron Yields from Spontaneous Fission of Transuranic Isotopes," *Trans. Am. Nucl. Soc.* **22**, 673 (1975).
20. R. T. Perry and W. B. Wilson, "Neutron Production from  $(\alpha, n)$  Reactions and Spontaneous Fission in  $\text{ThO}_2$ ,  $\text{UO}_2$ , and  $(\text{U}, \text{Pu})\text{O}_2$  Fuels," Los Alamos National Laboratory report LA-8869-MS (June 1981).
21. V. V. Malinovskij, V. G. Vorob'eva, and B. D. Kuzminov, "Review of Measurements of the Average Number of Prompt Fission Neutrons," INDC (CCP)-239/L, 24 (1985).
22. N. E. Holden and M. S. Zucker, "Prompt Neutron Multiplicities for the Transplutonium Nuclides," *Proc. Int. Conf. Nucl. Data for Basic and Applied Sci.*, Young, P. G., et al., Eds., Gordon and Breach, NY, 1631 (1986).
23. D. Reilly, N. Ensslin, H. Smith, Jr., and S. Kreiner, Passive Nondestructive Assay of Nuclear Materials, NUREG/CR-5550, Los Alamos National Laboratory (March 1991).
24. D. G. Madland and J. R. Nix "New Calculation of Prompt Fission Neutron Spectra and Average Prompt Neutron Multiplicities," *Nucl. Sci. Eng.* **81**, 213 (1982).
25. D. G. Madland, "Calculation of Prompt Fission Neutron Spectra and Average Prompt Neutron Multiplicities for the Spontaneous Fission of the Even Isotopes of Plutonium," *Inter. Conf. Nucl. Data for Sci. and Tech.* Gatlinburg (May 9-13, 1994).
26. D. G. Madland, Los Alamos National Laboratory private communication (Jan. 1997).
27. W. B. Wilson, M. Bozoian, and R. T. Perry, "Calculated  $\alpha$ -Induced Thick-Target Neutron Yields and Spectra, with Comparison to Measured Data," *Nucl. Data for Sci. and Tech.* Mito, Japan (1988).
28. M. Bozoian, R. T. Perry, and W. B. Wilson, "Calculated  $\alpha$ -Induced Thick-Target Neutron and Radionuclide Yields for  $E_\alpha \leq 50$  Mev," *Workshop on High Energy and Heavy Ion Beams in Materials Analysis*, Alb, NM (June 14, 1989).

