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NEUTRON TRANSPORT MEASUREMENTS
AND SOURCE MONITOR DEVELOPMENT
FOR URANIUM BOREHOLE LOGGING SONDE

FINAL REPORT

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FOREWORD

The material presented in this report is conveniently divided into two parts:

- I. Benchmark Experiments
- II. 14-MeV Neutron Monitor

The first part discusses the design and results of a ^{252}Cf -neutron and a 14-MeV neutron benchmark experiment for verifying previously developed theoretical methods for use in the design of a neutron logging tool for uranium borehole exploration. The second part discusses the Science Applications, Inc. (SAI) development of a fast fission monitor for measuring 14-MeV neutrons for a D-T borehole sonde in a high neutron moderating and absorbing environment. The 14-MeV pulsed neutron monitor was used in carrying out the 14-MeV neutron benchmark experiment, and a variation on the method of quantitatively counting many events in one burst of "pileup counts," developed for the 14-MeV-pulsed neutron monitor, was employed in successfully counting the epithermal neutrons produced by a short (2 μsec) burst of 14-MeV neutrons. Thus, the development of the 14-MeV-neutron monitor and the measurements with 14-MeV neutrons were intimately related.

The work presented here was performed under subcontract to Bendix Field Engineering Corporation, operating contractor for the Department of Energy National Uranium Resource Evaluation Program.

Part I. Benchmark Experiments

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1. INTRODUCTION

Fission-neutron logging as a means of direct rock interrogation for determining the presence and abundance of underground uranium ore reserves has been explored for some time in the United States and abroad.

A recent Science Applications, Inc. (SAI) study¹ has verified and further evaluated the potential of the fission-neutron borehole logging technique by theoretical models and has developed criteria for the optimum design of tools to be used in the field for uranium exploration work.

The present study was designed as an experimental test of the analytic models used in the above computational work, and its results will serve to strengthen or refute the conclusions based on theoretical models as well as the validity of the models themselves.

Active neutron logging is based on the premise that the presence of fissionable material in the ground will alter the time characteristics of the epithermal neutron flux impinging on a detector as a result of thermal-neutron fission resulting from the introduction of an external neutron source. (The thermal neutron flux is also observable long after the thermal neutron die-away in the ground because of the presence of delayed fission neutrons (DFN) in the fertile zone.) The ability to measure these effects and, hence, measure quantities related to the presence of an ore of fissionable material (e.g., uranium) depends on a number of parameters related to the size and composition of the ore, the energy and strength of the primary neutron source, and the sensitivity of the experimental

equipment. Pulsed 14-MeV neutrons and isotopic ^{252}Cf neutrons are the two most popular sources for borehole logging use today. The present experimental program has used both types of neutron sources.

1.1 MEASUREMENTS WITH A ^{252}Cf NEUTRON SOURCE

Specifically, the present ^{252}Cf experiments have been designed to measure the thermal neutron detector signal (and, hence, the thermal flux) in an ore-free borehole model as a function of distance from the californium source and also to measure the neutron detector response in the borehole due to fission neutrons produced in an ore model at some radial distance from the borehole. The latter effect was simulated by placing a spontaneous fission source (^{252}Cf) at radial positions of the borehole model and by measuring the neutron flux in the central borehole.

In addition, the ^{252}Cf experiments were designed to measure the epithermal neutron flux which also contributes to the production of fission events in the uranium ore. The thermal and epithermal components of the neutron flux at various distances from the ^{252}Cf fission source were separately measured by using bare ^3He counters and cadmium-covered counters. Since the thermal neutron flux is highly sensitive to the amount of neutron absorbing materials in the matrix, while the epithermal neutron flux is more dependent on the slowing down power (moisture content) of the borehole test bed, the separation of thermal and epithermal components made it possible to correct for the difference in absorption between the concrete test bed and the sandstone used in the modeling calculations (see Appendix II). It was thus possible to directly test the calculations in sandstone with measurements in the concrete test bed.

1.2 MEASUREMENTS WITH A PULSED 14-MeV NEUTRON SOURCE

The radial distributions of thermal and epithermal neutrons were also measured for the pulsed 14-MeV neutron source from a borehole sonde in order to determine the accuracy of the computational model ¹ (i.e., one-dimensional discrete-ordinates with P_3 angular distributions and S_{16} quadrature) for these higher energy neutrons. These differ from the ²⁵²Cf fission neutrons in that for some elements, such as calcium (in concrete), the neutron elastic scattering angular distributions are much more forward peaked. Also, the cross section for charged-particle production by 14-MeV neutron reactions is much larger than for fission neutrons. Moreover, the neutron cross sections in the 14- to 5-MeV region, very important to neutron thermalization, are not as well known as those for the lower energy region.

Different measurement techniques had to be utilized with the 14-MeV pulsed source than with the ²⁵²Cf steady-state source to circumvent the pulse pileup problem: this is the problem of several pulses being counted during the resolving time of the counter-amplifier system. Thus, additional detector calibration techniques and additional precautionary measures had to be instituted to achieve reliable calibration and flux counting with this necessarily more complex system. These are described in Section 8.

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2. EXPERIMENTAL ARRANGEMENT

The measurements described here were carried out in an experimental borehole model (Model C) at the Bendix Field Engineering Corporation (BFEC), Grand Junction, Colorado.

The axial and radial cross section views of Model C are shown schematically in Fig. 1 and 2. The model is a seven-foot diameter concrete cylinder, physically located partially above ground, that contains three separate zones: a bottom barren zone, 4.3 feet high, a middle uranium ore zone, 6.7 feet high; and a top barren zone 4.1 feet high. The flux measurements were made in the bottom barren zone with the experimental pit being saturated with water. Model C (Test Pit C) has 13 vertical holes, including a 4.5-inch diameter borehole, four 4-inch diameter holes, and eight 1.5-inch diameter detection holes. The detection holes, used for measuring neutron flux when a neutron source was placed in the borehole, were arranged radially so that they did not shadow each other when viewed from the borehole axis.

Two ^3He detectors were used for measuring the neutron flux, one being 1-inch in diameter and 6 inches long and filled to a pressure of 4 atmospheres, and the other having a 0.25-inch diameter, 3-inch length, and filled to 1 atmosphere. Flux measurements were taken with bare and cadmium-covered ^3He detectors. Bare ^3He detector count rates were used as a measure of the total neutron flux (thermal plus epithermal) while cadmium-covered ^3He detector count rates were used as a measure of the epithermal neutron flux. The procedure for

Figure 1. Axial Cross Section of the Test Pit C

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Central Borehole: 4.5" Diameter
 Holes A₁, A₂, A₃, A₄: 4" Diameter
 Holes 1 through 8: 1.5" Diameter Detector Holes
 Angles α , β , γ are 26°, 19°, and 45°, respectively
 Radial Distances of Holes 1 through 8 are
 7, 11, 15, 19, 23, 28, 33, and 38", respectively

Figure 2. Radial Cross Section of the Test Pit C.

determining thermal and epithermal neutron fluxes from bare and cadmium-covered ^3He detector count rates is described in a subsequent section. The flux calibration of ^3He detectors was achieved by using thermal activation foils. The manufacturer's data on the ^{252}Cf source strength was verified using threshold foil activation as described in Sections 3 and 4.

Information on parasitic neutron capture (poison) and saturation moisture contents of Model C was provided to SAI by BFEC and was based on experimental results obtained by Robert Wilson of BFEC. ²

3. SUMMARY OF FOIL ACTIVATION PRINCIPLES

Foil activation was used in this experiment for calibrating both the neutron source (threshold foils) and the neutron detector (thermal foils). Foil activation principles are, therefore, reviewed here for the reader's convenience.

The net number of radionuclides of a particular species that are produced per unit time and unit volume by activating a foil in a neutron flux, ϕ , is given by the following linear differential equation:

$$\frac{dN}{dt} = \Sigma\phi - \lambda N \quad (1)$$

where N = atomic concentration of the activation-produced radionuclides (atoms/cm³)

Σ = macroscopic activation cross section (cm⁻¹)

λ = decay constant (sec⁻¹)

In both source and detector calibration, the quantity of interest to be extracted from the foil activity is the neutron flux. Solving Equation (1) for ϕ with the usual substitutions for N and Σ yields the following expression:

$$\phi = \frac{C\lambda A}{\epsilon FN_0 f_{\text{om}}} e^{\lambda t_w} \left[(1 - e^{-\lambda t_i})(1 - e^{-\lambda t_c}) \right]^{-1} \quad (2)$$

where C = total number of activity counts

ϵ = gamma detector efficiency

F = gamma ray yield per decay from the activation product
 f = fractional isotopic abundance of the target (activated) isotope in the sample (foil)
 A = mass number of the irradiated (target) nuclide (gm)
 λ = decay constant of the activation product (sec^{-1})
 t_w = waiting time between the end of irradiation and the beginning of the counting period (sec)
 t_i = irradiation period (sec)
 t_c = counting period (sec)
 σ = microscopic activation cross section (barns)
 m = mass of the irradiated sample (foil) in grams
 N_0 = Avogadro's number - $0.6022169 \text{ (cm}^2/\text{barn-mol)}$
 ϕ = neutron flux ($\text{n/cm}^2 \cdot \text{sec}$)

4. CALIFORNIUM SOURCE CALIBRATION WITH FOILS

Six ^{252}Cf sources were used in the calibration experiment designed to measure the neutron source strength and to verify the manufacturer data available with the sources. Four of the sources belong to BFEC and two to SAI. Table 1 gives the original manufacturer's data on the ^{252}Cf source strengths and the corresponding calculated strength at the time of the calibration experiment. The experimental arrangement is shown in Fig. 3.

Aluminum foils were used for measuring the neutron flux from the ^{252}Cf sources according to the threshold reaction:

which yields ^{27}Mg , a radioactive nuclide that decays with a 9.46-minute half-life, emitting gamma radiation of 0.84 MeV (70 percent) and 1.013 MeV (30 percent). Both gamma line activities were measured for better accuracy.

The flux was evaluated from the $^{27}\text{Al}(n,p)^{27}\text{Mg}$ foil activations according to Equation (2). The corresponding data are given in Table 2. The counts and the efficiencies referred to in this table are for a Ge(Li) detector used to measure the photopeaks of the two gamma lines of ^{27}Mg .

Because of the approximately uniform radial distribution of the sources in this experiment, the neutron flux incident on the aluminum foil is given by the point-source expression:

$$\phi = \frac{S}{4\pi R^2} \quad (3)$$

Table 1. ^{252}Cf Neutron Sources

Source Ownership	Original Amount (μg)	Original Neutron Yield (n/s)	Decay Time (y)	Corrected Yield (n/s)
BFEC	4.421	$1.0217 + 7^*$	0.668	$8.572 + 6$
BFEC	4.596	$1.0621 + 7$	0.668	$8.911 + 6$
BFEC	4.270	$0.9868 + 7$	0.688	$8.279 + 6$
BFEC	4.382	$1.0127 + 7$	0.668	$8.497 + 6$
SAI	12.5	$2.8880 + 7$	0.668	$2.423 + 7$
SAI	8.75	$2.0213 + 7$	3.683	$7.681 + 6$
TOTAL		$8.993 + 7$		$6.617 + 7$

* read as 1.0217×10^7

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Figure 3. Geometry of the ^{252}Cf Source Calibration Experiment

Table 2. $^{27}\text{Al}(n,p)^{27}\text{Mg}$ Foil Activation Data

	<u>Gamma 1</u>	<u>Gamma 2</u>
$E_{\gamma}(\text{MeV})$	0.844	1.013
A	27	27
F	0.7	0.3
f	1.0	1.0
$\lambda(\text{sec}^{-1})$	$1.2213 \cdot 10^{-3}$	$1.2213 \cdot 10^{-3}$
ϵ	0.076	0.060
$\sigma(\text{barns})$	4.94	4.94
$t_w(\text{sec})$	120	120
$t_i(\text{sec})$	600	600
$t_c(\text{sec})$	600	600
C (counts)	2128	702

where S = source strength (n/s)

R = distance from foil to geometric
center of sources (cm)

The neutron flux at the foil positions as calculated from the time-corrected manufacturer data is 4.597×10^5 n/cm²·sec, while the mean flux evaluated from the aluminum foil activation data, based on both the 0.844-MeV and 1.013-MeV gamma lines (that were different from each other by 2.48 percent), was 4.696×10^5 n/cm² sec. The 2 percent disagreement between the calculated and measured fluxes is well within the experimental error.

One of the most critical quantities for evaluating ϕ from Equation (2) in this activation experiment is the "effective" activation cross section. This is because the ²⁵²Cf neutron spectrum extends below the 3-MeV threshold of the ²⁷Al(n,p) ²⁷Mg reaction. Therefore, the σ used in Equation (2) had to be a ²⁵²Cf spectrum averaged value that accounted for the aluminum foil threshold. According to the formalism of Calamand,³ this effective cross section can be written as:

$$\begin{aligned}\sigma_{\text{eff}} &= \int_0^{\infty} \sigma(E) \chi(E) dE \\ &= a A^{2/3} \int_0^{\infty} P(E) \chi(E) dE \\ &= a A^{2/3} \int_{E_{\text{eff}}}^{\infty} \chi(E) dE\end{aligned}\tag{4}$$

where a = units constant (see Reference 3)

$\chi(E)$ = ²⁵²Cf fission spectrum

$P(E)$ = penetrability of the Coulomb barrier
which confronts the charged particle
(proton) while leaving the compound
nucleus (²⁷Al+n)

The energy variation of $\chi(E)$ and $P(E)$ are shown graphically in Fig. 4. The effective cross section value obtained by this procedure for the ^{252}Cf neutron-induced $^{27}\text{Al}(n,p)^{27}\text{Mg}$ activation reaction is $4.0 \pm 0.45 \times 10^{-3}$ barn. The detailed fission spectrum used ³ is generally representative not only of ^{252}Cf spontaneous fission neutrons but also of fission neutrons induced by thermal neutrons in fissile nuclides (e.g., ^{235}U).

Figure 4. The Weighted Penetrability Function. E_{eff} is defined such that areas A and B are equal.

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5. NEUTRON DETECTOR CALIBRATION AND SENSITIVITIES

The thermal and epithermal neutron fluxes as a function of the radial distance from the source were measured in the detection holes with bare and cadmium-covered ^3He detectors. The count rates from these detectors were transformed into neutron fluxes by calibrating with thermal neutron foils of gold (^{197}Au) and manganese (^{55}Mn) according to the following capture reactions:

and

The foil activation data are summarized in Table 3 and include quantities defined in connection with Equation (2). The gamma photo-peak activities were measured with a 3-inch x 3-inch NaI(Tl) detector at Grand Junction and were repeated with a Ge(Li) detector at SAI. The latter results were more complete and more accurate; hence, they were used in the ^3He detector calibration calculations.

The cadmium cover was 0.76-mm thick and the gold foil was 63.5 μ thick.

In order to obtain the purely thermal activation from the total activation of a foil, one must subtract the epicadmium activation multiplied by a Cd correction factor, F_{Cd} , that accounts for the neutron spectrum change when Cd is used as a cover to absorb thermal neutrons incident upon the foil as well as other effects.

Table 3. Thermal Foil Activation Data

	$^{197}\text{Au}(n, \gamma)^{198}\text{Au}$	$^{55}\text{Mn}(n, \gamma)^{56}\text{Mn}$
Cover	Bare	Bare
E_{γ} (MeV)	0.412	0.847
A	197	55
C (counts)	6600	673
F	0.95	0.989
f	1.0	0.813
$\lambda(\text{sec}^{-1})$	2.9757-6	7.4744-5
ϵ	0.017	0.0072
σ	98.8	13.3
m(g)	0.1495	0.1772
t_w (sec)	10,980	360
t_i (sec)	349,500	60,100
t_c (sec)	20,000	12,753

Thus:

$$C_{th} = C_{bare} - F_{Cd} C_{Cd} \quad (5)$$

where C_{th} = counts due to thermal neutrons
 C_{bare} = counts due to all activations
 C_{Cd} = counts due to epicadmium neutrons.

With the aid of Equation (2), C_{th} gives the thermal flux ϕ_{th} directly. The correction factor can be obtained analytically according to a procedure discussed in Reference 4 which also provides appropriate graphs for F_{Cd} . The value of 1.13 for F_{Cd} was used for the present experiment with Au foils.

Thus, one could calculate a thermal to epithermal flux ratio as given by the following expression:

$$\frac{\phi_{th}}{\phi_{epi}} = \frac{\phi_{bare} - \phi_{Cd} \cdot F_{Cd}}{\phi_{Cd} \cdot F_{Cd}} \quad (6)$$

Here, ϕ_{bare} refers to the "bare foil" flux obtained by using the bare foil activation (C_{bare}) and Equation (2) of Section 3. This gives ϕ_{epi} directly at the calibration point (Detector Hole #4). Thus, ϕ_{th} and ϕ_{epi} are known for Hole #4.

Using this flux ratio, the equivalent correction factor, F_{eq}^{Cd} , was calculated for the bare and cadmium-covered ^3He detectors of both sizes used, i.e., 1 inch x 6 inches and 0.25 inch x 3 inches, from the following expression:

$$\frac{C_{bare}(^3\text{He}) - C_{Cd}(^3\text{He}) \cdot F_{eq}^{Cd}}{C_{Cd}(^3\text{He}) \cdot F_{eq}^{Cd}} = \frac{\phi_{th}}{\phi_{epi}} \quad (7)$$

Here, C_{bare} and C_{Cd} were the count rates in Detector Hole #4, so that ϕ_{th} and ϕ_{epi} are the same as in Equation (6), as measured with the Au foils. From these, the 3He detector sensitivities were calculated for thermal neutron measurements with both 3He detectors as per the following expression, using Equation (2) to obtain $\phi_{th}(foil)$:

$$S = \frac{C_{th}(^3He)}{\phi_{th}(foil)} \quad (8)$$

For the 1-inch x 6-inch detector, the values of S from Au and Mn foil measurements were averaged to 13.9 counts-cm²/neutron. The value of S for the 0.25-inch x 3-inch 3He detector was found to be 0.232 counts-cm²/neutron. These values were then used to calculate radial-dependent thermal and epithermal neutron fluxes in Model C, utilizing Equation (7) to obtain ϕ_{epi} from ϕ_{th} , $C_{bare}(^3He)$, and $C_{Cd}(^3He)$.

The validity of the above experimentally determined S values was confirmed by comparing them with approximate theoretical sensitivities evaluated as follows.

The neutron sensitivity of a 3He detector as used in the present context is the ratio of the neutron induced (n,p) reaction rate in the 3He detector and the neutron flux:

$$S = \frac{(V\Sigma\phi)\gamma}{\phi} = V\Sigma\gamma \quad (9)$$

where V = 3He detector volume
 Σ = macroscopic $^3He(n,p)$ cross section
 γ = self-shielding factor*.

*To be discussed in Section 6. See also SAI report SAI-78-551-LJ (1978).

The value of S should be, in principle, evaluated analytically by using an energy-integrated reaction rate since Σ is a function of energy. The present calculation, however, serves only the purpose of establishing the validity of the experimentally-determined sensitivities and can, therefore, produce adequate results if an effective thermal cross section value is used.

From Equation (9), one can obtain by algebraic manipulation:

$$S = N_L P \sigma V \gamma \quad (10)$$

where σ = microscopic $^3\text{He}(n,p)$ cross section (barns)
 N_L = Loschmidt number = 2.6872×10^{-5} molecules/
barn-cm of ideal gas
 P = filling pressure (atmospheres)

The cross section value used was 4237 barns*. The rest of the ^3He detector data are given in Table 4: γ was obtained as per Section 6. The results of this approximate calculation are shown in Table 5 where they are compared with the previously described experimental values.

*See analytic justification in Appendix I.

Table 4. ^3He Detector Data for Sensitivity Evaluation

<u>Detector</u>	<u>Pressure (atm)</u>	<u>γ (Thermal)</u>	<u>Volume (cm³)</u>
1" x 6"	4	0.493	77.22
0.25" x 3"	1	0.947	2.413

Table 5. ^3He Detector Sensitivity

<u>Detector</u>	<u>Pressure</u>	<u>S_{calc}</u>	<u>S_{exp}</u>
1" x 6"	4	17.3	13.9
0.25" x 3"	1	0.26	0.232

6. THE SELF-ABSORPTION EFFECT IN A ^3He DETECTOR

^3He detectors are routinely used for thermal neutron flux measurements, and the rationale of this procedure is that the measured reaction rate, RR , in a detector of volume V is proportional to the flux because:

$$RR = \Sigma_a V \phi \quad (11)$$

or, more accurately,

$$RR = \int_V d\vec{r}^3 \int_0^\infty dE N(\vec{r}) \sigma_a(E) \phi(E, \vec{r}) \quad (12)$$

where σ_a , Σ_a are the microscopic and macroscopic absorption cross sections, respectively, and N is the atomic concentration of the absorber (here ^3He). Since ^3He has a large (n,p) absorption cross section, the inner layers of the detector "see" neutron fluxes that are reduced by the outer layers and, consequently, are not equally "effective" in detecting the incident neutron flux. This effect, which increases with the mass of the absorber (i.e., the detector volume and its filling pressure) can be also described analytically through a "self-shielding" function, $\gamma(E)$, that multiplies either the flux or the cross section. Thus the "effective" reaction rate, $\overline{RR}$, can be written as:

$$\overline{RR} = \int_V d\vec{r}^3 \int_0^\infty dE N(\vec{r}) \sigma(E) \phi(E, \vec{r}) \gamma(E). \quad (13)$$

An approach for calculating $\gamma(E)$ was explored in a previous publication⁵ and yielded the same results as the approach presented here.

Equation 13 can be evaluated as follows. Consider a long cylindrical detector of radius R on which an isotropic neutron flux is incident. Let (θ, ϕ) be the direction of arbitrary neutron incidence and ℓ the traveling distance through the cylinder. Ignoring scattering by the ^3He atoms, which is very small compared to the absorption, the (n, p) reaction rate per unit length of cylinder⁶ can be written as:

$$\begin{aligned} \rho &= 2\pi R \frac{\Sigma_{n,p}}{\Sigma_a} \frac{\phi}{4\pi} \int_0^{2\pi} d\phi \int_0^{\pi/2} d\theta \left[1 - e^{-\Sigma_a \ell(\theta, \phi)} \right] \cos\theta \sin\theta \\ &= 2\pi R \frac{\phi}{4\pi} \int_0^{2\pi} d\phi \int_0^{\pi/2} d\theta \left[1 - e^{-\Sigma_a \ell(\theta, \phi)} \right] \cos\theta \sin\theta \end{aligned} \quad (14)$$

since Σ_a equals $\Sigma_{n,p}$.

$$\text{Then} \quad \rho = \pi R \frac{\phi}{2} \chi(\Sigma_a R) \quad (15)$$

$$\text{where } \chi(\Sigma_a R) = \frac{1}{\pi} \int_0^{2\pi} d\phi \int_0^{\pi/2} d\theta \left[1 - e^{-\Sigma_a \ell(\theta, \phi)} \right] \cos\theta \sin\theta \quad (16)$$

and

$$\ell = \frac{2R \cos\theta}{1 - \sin^2\theta \cos^2\phi} \quad (17)$$

The integration of Equation (16) leads to the following complicated combination of modified Bessel functions of the first and second kind:

Figure 5. Neutron Absorption in a Long Cylindrical Detector

$$\begin{aligned} \chi(\Sigma_a R) = & \frac{4}{3} (\Sigma_a R)^2 \{ 2 [\Sigma_a R \{ K_1(\Sigma_a R) I_1(\Sigma_a R) \\ & + K_0(\Sigma_a R) I_0(\Sigma_a R) \} - 1] + \frac{1}{\Sigma_a R} [K_1(\Sigma_a R) I_1(\Sigma_a R)] \\ & - K_0(\Sigma_a R) I_1(\Sigma_a R) + K_1(\Sigma_a R) I_0(\Sigma_a R) \} . \end{aligned} \quad (18)$$

Quantity $\chi(\Sigma_a R)$ is related to the escape probability from the cylinder, P_e , as follows: ⁶

$$P_e = \frac{\chi(\Sigma_a R)}{\Sigma_a R} . \quad (19)$$

Fortunately, Sauer ⁷ has developed a simple approximation to P_e for infinite cylinders that is very good so long as the length of the cylinder is much larger than its radius. In terms of $\chi(\Sigma_a R)$, the approximation gives:

$$\chi(\Sigma_a R) = 1 - \left[1 + \frac{2\Sigma_a R}{n+1} \right]^{-(n+1)} \quad (20)$$

where n equals 3.58 for an infinite cylinder. Then, Equation (14) can be written as:

$$\rho = \pi R^2 \frac{\Sigma_a}{2\Sigma_a R} \phi \chi(\Sigma_a R) \quad (21)$$

and, for a cylinder of length L , the reaction rate is given by:

$$\begin{aligned} \overline{RR} = \rho L &= \pi R^2 L \frac{\Sigma_a}{2\Sigma_a R} \phi \chi(\Sigma_a R) \\ &= V \Sigma_a \phi \gamma \end{aligned} \quad (22)$$

where

$$\gamma = \frac{1}{2 \Sigma_a R} \left\{ 1 - \left[1 + \frac{2 \Sigma_a R}{n+1} \right]^{-(n+1)} \right\} \quad (23)$$

This result is the same as Equation (35) of Reference 5. For 2200 m/s neutrons, ($\sigma_a = 5330$ barns), this becomes:

$$\gamma_{2200} = \frac{3.491}{PR} \left[1 - (1 + 0.06254 PR)^{-4.58} \right] \quad (24)$$

where R is the radius of the cylinder and P is its filling pressure in atmospheres.

The validity of this model* for predicting the flux self-shielding due to a ^3He detector was verified by comparing the calculated and experimental results using a 1-atm 0.25-inch x 3-inch detector and a 4-atm 1-inch x 6-inch detector at the same spatial point. If the theoretically evaluated self-shielding factors for the two detectors were evaluated exactly, then the quantity:

$$\frac{C_1}{P_1 V_1 \gamma_1} \cdot \frac{P_2 V_2 \gamma_2}{C_2},$$

where C_1 and C_2 are the counts from the two detectors and the rest of the parameters for the two detectors are as defined above, should equal unity. The ratio was found to be 1.07, which implies that the approximation used in evaluating the self-shielding factor is quite good.

* and of that given in Reference 5

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7. RESULTS FROM ^{252}Cf -NEUTRON-SOURCE EXPERIMENTS AND COMPARISON WITH CALCULATIONS

The thermal and epithermal fluxes as a function of radial distance from the center of the borehole in the lower barren zone of Test Pit C were measured with the ^{252}Cf source in the center and the detector in the detection holes (Fig. 6 and 7), and vice-versa, with the source in the detection holes and the detector in the borehole (Fig. 8 and 9). These two sets of curves are very similar in slopes, while the forward thermal data (Fig. 6) are 42 percent higher than the backward (Fig. 8), and forward epithermal data (Fig. 7) are 10 percent higher than the backward (Fig. 9). Since the slopes are the same, the high-energy neutron transport is identical both ways. The differences, being very small for epithermal and larger for thermal, are due to the larger volume of water around the detector in the backward case, i.e., detector in borehole. Since the cross section for this (excess) water increases with decreasing energy faster than it does in most concrete, the count rate with the detector in the borehole will be smaller.

The identity of the slopes indicates that neutron leakage to the outside of the test pit is not very important. In general, if a medium is large enough relative to the slowing-down length of the source neutrons, the flux should depend on the source-detector distance and not on the absolute positions of either the source or the detector. This seems to be the case in the present measurements. Thus, the geometry of Model C with ^{252}Cf as a neutron source is adequate as a mockup of an actual borehole geometry in the ground. Since the

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Figure 6: Thermal Neutron Flux per Source Neutron from a ^{252}Cf Source at the Center of Model C

Figure 7.. Epithermal Neutron Flux per Source Neutron
(Weighted by the ^3He (n,p) Reaction Rate)
from a ^{252}Cf Source at the Center of Model C

Figure 8 . Thermal Neutron Flux per Fission Neutron from a ^{252}Cf Source in Detection Holes of Model C and with the Detector in the Central Borehole.

Figure 9. Epithermal Neutron Flux per Fission Neutron (Weighted by the ^3He (n,p) Cross Section) from a ^{252}Cf Source in Detection Holes of Model C and with the Detector in the Central Borehole

theoretical calculations¹ were done for sandstone, and the measurements in concrete, the agreement between the theoretical and experimental curves for both thermal and epithermal fluxes (Fig. 6 and 7) implies that, for a ^{252}Cf neutron source, the slowing-down properties of both media are similar. Also, the magnitudes of the measured and calculated epithermal fluxes agree very well. Some discrepancy was observed, however, between the magnitudes of the measured and calculated thermal neutron fluxes. These were determined to be due to differences in composition between sandstone and concrete that result in different thermal neutron absorption cross sections for the two media. The concrete of Model C contains traces of materials² having large neutron absorption cross sections (boron, cadmium, europium, gadolinium, samarium, etc.) and as a result, its macroscopic thermal absorption cross section is considerably higher than that of sandstone which was used in the theoretical calculations. When the absorption effect of these "poisons" was subtracted from the experimental data (correction factor = 2.8) the resulting agreement between measurements and calculations was very good (Fig. 6). The procedure by which this correction was applied is fully described in Appendix II and is based on analytic models reviewed in Reference 8.

Since all measurements were carried out with the borehole and the detection holes flooded with water, it was necessary to first determine the moisture content of concrete before the proper analytic flux curve was selected for comparison. R. Wilson of BFEC has carried out independent moisture measurements for Model C,² and he determined the saturation content to be 13.1 percent by weight. Thus, our measurements were compared with available analytic curves¹ for 13 percent moisture in sandstone and for a 4.5-inch diameter borehole. The theoretical calculations were made with ANISN, a discrete ordinates code, using a 37-group neutron energy structure.

The results of group 37 (0.4 eV to 10^{-5} eV) were used for comparison with the measured thermal fluxes.

For the epithermal results, however, it was necessary to process the calculated data into a form that is directly comparable with the measured data (Fig.7 and 9). Although the epithermal experimental results were corrected for self-shielding effects and calibrated with foils, as discussed before, they are still not truly epithermal fluxes; they are epithermal-neutron-induced (n,p) reaction rates in ^3He detectors. Thus, rather than unfold these data with the $1/v$ cross section of the (n,p) reaction in ^3He , it was easier to fold (weight) the calculated epithermal neutron flux with the inverse-velocity response of the ^3He detector. The ^3He cross sections for this procedure were obtained from the analytic study.¹ This forward procedure was found to be simpler and quite adequate for comparisons between experimental and theoretical results. It must, therefore, be borne in mind that the comparisons shown in Fig.7 and 9 are not for true epithermal neutron fluxes, but for ^3He (n,p) cross-section-weighted epithermal neutron fluxes.

In addition to flux measurements in the source plane, vertical scans were made to help ascertain the accuracy of locating the true source plane with the counter and to determine if the presence of the fertile ore zone, a few feet above the source plane, would affect the measurements in the source plane. The corresponding count rate data are shown in Fig.10 and 11 for the thermal and epithermal fluxes, respectively. These plots peak at a vertical position very near the level of the measured source plane. In addition, the vertical plots are approximately symmetric about the measured vertical position of the source plane. We can conclude from these data that the presence of the fertile zone has no appreciable effect on the measurements.

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Figure 10. Vertical Thermal Flux Profile, ^{252}Cf Neutron Source

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Figure 11. Vertical Epithermal Flux Profiles, ^{252}Cf Neutron Source

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8. FLUX MAPPING TECHNIQUES WITH A PULSED 14-MeV NEUTRON SOURCE

The thermal neutron die-away time for a very short burst of 14-MeV neutrons ($\sim 2 \mu\text{sec}$) is of the order of hundreds of microseconds in concrete with 13 percent moisture content as available for these measurements. With the use of a short amplifier time-constant, pulses occurring within a few microseconds of each other can be resolved with the discriminator and scaler circuits that were employed. The thermal pulses out of the amplifier were observed on the oscilloscope at each measuring point, and the sonde output was adjusted so that a negligible fraction of the pulses occurred within a few microseconds of each other. The ^{238}U neutron monitor output provided proper 14-MeV source normalization.

When a cadmium cover was placed on the ^3He counter, these delayed (thermal) neutrons all disappeared. A very large single (unresolved) pulse of epithermal neutrons was present. Thus, the epithermal epicadmium neutron events all piled up into a single count with one count occurring per 14-MeV burst, i.e., one count per second. This single "epithermal" count, one per burst, was of course, subtracted from the thermal counts obtained with the bare counter as described in the preceding paragraph.

The pulse-height "spectrum" of the epithermal pileup pulses was measured with a multichannel pulse-height analyzer at each radial distance from the 14-MeV sonde, and the pulse height channel number for each pulse was divided by the average pulse height produced by a single neutron count to obtain

the stochastically-averaged number of epithermal neutrons for the flux measurement.

The average pulse height for a single neutron event, on the other hand, was determined by a separate measurement in which the counter was placed outside a ^{252}Cf shipping cask containing the ^{252}Cf sources. The pulse height distribution from these "singles", as measured with the same pulse height analyzer but with higher known amplifier gain (which was properly taken into account), was used to obtain the average pulse height of a single neutron event.

The epithermal neutrons are measured during the slowing-down time (as differentiated from the much longer thermal die-away time) in concrete with 13 percent moisture. This slowing-down time is the order of 5 to 10 μsec . Since nearly all of the epithermal neutrons are counted within 10 to 15 μsec , the amplifier time-constant was set to 8 μsec . This produced a positive-going pulse of ~ 20 μsec duration, long enough to linearly add the contribution of all epithermal pulses.

To ensure proper preamplifier and amplifier linearity, i.e., absence of saturation effects in the electronics, the system was tested with a precision pulser that was fed to the input of the preamplifier in the standard fashion. The system saturation and, correspondingly, the upper limit for a linear response was at a pulse height well above any epithermal-neutron "pileup pulse" measured in these experiments.

9. RESULTS FROM 14-MeV PULSED-NEUTRON-SOURCE MEASUREMENTS, AND COMPARISON WITH CALCULATIONS

The goal for both the ^{252}Cf and the 14-MeV measurements was to perform a set of benchmark radial flux measurements in the test holes #1 through #8 (Fig. 2) that were especially made for these measurements. The source plane was to be about 2 feet from the bottom of the lowest concrete barren zone (see Fig. 1) which would place it at the center of this zone. From the set of Pit C drawings initially provided to us, this would locate the source plane at 14 feet from the top. But at this depth, as shown in Fig. 1, we were actually 11 inches above the bottom of the barren zone.

The ^{252}Cf measurements were made at this depth before the proper drawing was made available. This does not in any way invalidate them because the densely packed earth below has a density close enough to that of concrete to produce neutron albedo (backscattering) effects similar to those of concrete at a point 11 inches above.

However, when the long 14-MeV-neutron sonde was lowered into the central borehole, it was discovered that the same source plane could not be reproduced. The borehole was not as deep as the original drawings portrayed. The lowest attainable depth of the 14-MeV source was 29 inches from the bottom. An investigation was made of this inconsistency, as a result of which the correct drawings were found (illustrated in Fig. 1). Consequently, the measurements were made with the source of the 14-MeV sonde at 29 inches above the bottom. However, the detector-depth position marker was inadvertently left at the point corresponding to a 14-foot depth, or a

distance of 18 inches below the source plane. It was, therefore, necessary to correct the epithermal-neutron flux plots to a source-to-detector distance

$$D(\text{in}) = \sqrt{r^2 + (18)^2}$$

(Here, r is the radial distance out to each 1.5-inch diameter measuring hole.) The resulting plot is shown in Fig. 12. Note that, due to this inconsistency, measurements for a maximum source-to-detector distance of nearly 100 cm (for hole #7) were obtained, rather than only about 80 cm for ^{252}Cf . This turned out to be quite useful for the 14-MeV data. Although shorter range measurements are absent, it was possible to attain greater neutron ranges than with the ^{252}Cf fission neutrons. (Compare Fig. 7 and 12.) This is important because the 14-MeV-neutron penetration is much greater.

The same comments hold true for the 14-MeV-source thermal-neutron flux plot shown in Fig. 13. The thermal-neutron flux was corrected for the difference in absorption cross section ("poison") between Pit C concrete and the sandstone (with added water) that was used in the calculation. This resulted in raising the magnitude of the measured flux by a factor of 2.8 for direct comparison with the calculations as plotted in Fig. 13.

For the measured epithermal neutron flux shown in Fig. 12, the difference in poisons between concrete and sandstone has a negligibly small effect.

It is worth noting that the agreement between measurements and calculations in both the thermal and the epithermal fluxes is quite good at the radial positions where experimental data are available.

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Figure 12. Epithermal Flux per Source Neutron versus Distance from a 14-MeV Neutron Source at the Center of Model C

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Figure 13. Thermal Flux per Source Neutron versus Distance from a 14-MeV Neutron Source at the Center of Model C

10. CONCLUSIONS

The immediate objective of this study has been to experimentally test the validity and accuracy of previously developed computational models for the analysis of uranium exploration techniques by active neutron interrogation. This borehole technique employs a source of neutrons (^{252}Cf spontaneous fission or 14 MeV) that travel to the uranium-bearing zone and induce secondary fission neutrons whose detection back in the borehole produces a measure of the uranium ore concentration. In order to separately study the forward transport of source neutrons from the borehole to the uranium ore and the transport of fission-induced neutrons back to the borehole, two sets of measurements were performed. The forward source-neutron transport in a borehole was experimentally simulated with a neutron source (^{252}Cf or 14 MeV) in the borehole and with neutron (thermal and epithermal) detectors at radial positions in a wet cylindrical concrete test pit. The transport of induced fission-neutrons back to the borehole was simulated with ^{252}Cf fission sources at radial positions in the test pit and with neutron detectors (thermal and epithermal) in the borehole.

The comparison between these measurements and neutron transport calculations have shown good agreement, although the calculations and the measurements were not performed in the same medium (concrete versus sandstone). The difference between the two media was seen to insignificantly affect the slowing-down of source neutrons while affecting the magnitude (but not the shape) of the thermalized flux through thermal-neutron absorption (poison). This effect was successfully

accounted for in comparing the measured and the calculated thermal-neutron radial distributions.

No significant difference in slope was seen between the measurements with a ^{252}Cf fission source in the borehole and a neutron detector at radial positions of the test pit and measurements with a ^{252}Cf fission source at radial positions of the test pit and a detector in the central borehole. This indicates that the test-pit system is large enough to simulate a real borehole configuration in which, due to the absence of neutron leakage from the system, this phenomenon is expected. Differences in magnitude between these forward and reverse measurements were seen, however. These are due to local differences in water volume in the immediate neighborhood of the neutron detectors. The difference is insignificantly small (v10 percent) for epithermal neutrons, but noticeably large (42 percent) for thermal neutrons whose water-to-concrete cross section ratio is much larger than for epithermal neutrons.

Although the reported comparisons between measurements and calculations have served to separately investigate various aspects of the primary (source) and secondary (fission-induced) neutron transport in a borehole-like test system, it would be useful to integrate these test studies by performing measurements in a test ore zone for comparison with the already available computed data. These will also serve to experimentally determine the overall sensitivities of the experimental system to the uranium concentration in a low-grade ore.

The ultimate objective of comparative studies between calculated and measured results of this type is to optimize the design and performance of a uranium exploration tool that operates based on the fission-neutron interrogation technique. This can best be achieved by developing

simple and direct correlations between important design parameters and the environment-dependent operational parameters. The relative sensitivity between these two sets of parameters can be most practically determined by developing simple analytic relations that synthesize the experimentally validated, computed results. The detailed results of neutron transport calculations are of use in this connection, but they are too voluminous to provide a direct avenue between the tool design parameters and the environment-dependent operational parameters. They can, nevertheless, serve as an invaluable basis for the development of such a direct avenue via analytic relationships.

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Part II. 14-MeV NEUTRON MONITOR

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1. INTRODUCTION

In uranium borehole logging with a pulsed 14-MeV neutron sonde, the amount of uranium present is determined by measuring either the production of epithermal prompt fission neutrons (PFN) from thermal neutron fission of ^{235}U atoms or the intensity of the delayed fission neutrons (DFN). A measure of the ore concentration is given by the intensity of fission neutrons per 14-MeV source neutron, so that the quantification of the ore concentration is directly limited by the accuracy of measuring the mean number of 14-MeV neutrons per pulse.

To facilitate accurate source measurements for the 14-MeV sonde, a neutron detector was chosen for the monitor such that only high-energy neutrons could be detected. In this way, the monitor is insensitive to all "moderated" neutrons and to most high-energy neutrons backscattered from the surrounding matrix. The extent to which the monitor was insensitive to the matrix was first investigated by SAI with the use of 14-MeV neutron transport calculations utilizing the geometry shown in Fig. 14. Here, r_b is the borehole radius. Water occupies the (spherical) region between radius $r_s = 1.125$ inches and r_b . The sandstone, between r_b and $r_c (= 5$ feet), varied in moisture content from 7 percent to 22 percent by weight. The results, shown in Fig. 15, indicate that a ^{238}U -lined fission counter will follow the source strength to ± 1 percent in the midst of all these matrix variations: i.e., $r_b = 2.25$ to 10.00 inches and the percentage of moisture from 7 percent to 22 percent.

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Figure 14. Borehole Calculation Model

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Figure 15. Relative Effect of Borehole Diameter and Moisture Variations on Monitor Performance. 14 MeV/TOTAL refers to ratio of counts directly from source to source plus backscattered counts.

The relative insensitivity of ^{238}U to the matrix is due to the energy dependence of the neutron fission cross section: it drops rapidly below 2 MeV, is flat from 2 to 6 MeV; and increases to about twice this plateau value at 14 MeV. The backscattered thermal and epithermal neutrons produce no fissions in the ^{238}U lining of the monitor.

The results of these studies were presented to the Bendix Field Engineering Corporation (BFEC), Grand Junction, Colorado, who subsequently sponsored a development effort for a 14-MeV neutron monitor.

This development consisted of procuring and testing a pair of ^{238}U -lined and unlined counters; the unlined to determine the size of pulse obtained from combinations of a) rf pickup from the sonde pulse, b) neutron-recoil atoms from counter gas and counter wall, and c) "gamma flash;" and the lined to detect neutrons having energies in excess of the $^{238}\text{U}(n,f)$ reaction threshold.

A description of the counters and electronics is given in Section 2, and the test results are presented in Section 3. A brief evaluation and recommendations for packaging and utilizing the ^{238}U monitor for pulsed 14-MeV sources are presented in Section 4.

2. EXPERIMENTAL ARRANGEMENTS

Measurements were made at the Department of Energy (DOE), Grand Junction, Colorado facilities in a model borehole with the geometry shown in Fig. 16. A ^{238}U -lined and an unlined counter were located on opposite sides of the 14-MeV source. The lined counter provides a large output pulse that is the result of many fission counts per pulse, while the unlined counter provides a measure of the "background" pulse due to a) 14-MeV elastic atomic recoils in the filling gas and walls of the chamber, b) "gamma-flash," and c) rf pickup.

The circuitry for summing up all the fission pulses and subtracting out these possible sources of background is shown in Fig. 17. An amplifier with common-mode noise rejection subtracts the background pulse from the lined fission counter pulse. The resultant difference pulse is analyzed by the pulse height analyzer, from which the address scaler output is taken ($\div 100$). This output is proportional to the analyzer pulse height and is divided by 100 to make it slow enough for most common scalers to follow without count loss (the address scaler output, undivided, is 50 megahertz in the INO-TECH multichannel analyzer, which is too fast for most available scalers).

The ^{238}U -lined and the unlined counters are Centronic type FC4A, 0.25 inch OD, 2.75 inch overall length, and 0.975 inch active length. The lined counter was coated with $1000 \frac{\mu\text{gm}}{\text{cm}^2}$ of highly depleted ^{238}U (depleted of ^{235}U to ~ 200 ppm).

Calibration of the counting arrangement shown in Fig. 17 was carried out by irradiating threshold activation

Figure 16. Sonde and 14-MeV Monitor Geometry for Grand Junction Measurements

U-238 LINED
FISSION
COUNTER

Figure 17. Block Diagram of Electronics for Measuring Mean Number of Fission Events per 14-MeV Neutron Pulse

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foils simultaneously with obtaining total counts from the address scaler, as discussed in Section 3. This provided the number of source neutrons per scaler count.

3. EXPERIMENTAL RESULTS

3.1 MONITOR CALIBRATION

Utilizing iron and aluminum foils to monitor the 14-MeV neutron source intensity (via the $^{27}\text{Al}(n,p)^{27}\text{Mg}$, $^{27}\text{Al}(n,\alpha)^{24}\text{Na}$, and $^{56}\text{Fe}(n,p)^{56}\text{Mn}$ reactions whose cross sections are known with reasonably good accuracy at 14 MeV), the sonde output was measured to be 4.6×10^7 neutrons/pulse. This is to be compared to the rated output of 8×10^7 neutrons/pulse, presumably when in top operating condition. This measurement was used to calibrate the monitor output in terms of source neutrons per pulse for the duration of each run.

3.2 SENSITIVITY TO MATRIX

Next, measurements were made with the monitor seven inches away from the nearest solid concrete shielding brick (4 x 6 x 10 inches) and with the brick adjacent to the monitor. The measured difference was 18 percent, or ± 9 percent from the mean. This seems to contradict the results shown in Fig. 15. However, this change of scattering is much larger than for the case of Fig. 15 where the sandstone was replaced by water. Here, the concrete shielding brick material near the counter was removed.

Measurements in the dry and wet (filled with water) borehole showed only a ± 2 percent (4 percent between extremes) effect, which is reasonably small. The larger output was obtained with the borehole full of water, a 0.5-inch thick layer between sonde and borehole.

With no water in the borehole, the sonde was moved so that the monitor was adjacent to the borehole wall, then

moved $\sim 3/8$ inch away from it. The monitor output was nearly constant for both cases, showing a variation of ± 1.4 percent from the mean (2.7 percent between extremes), with the higher output obtained with the monitor against the borehole concrete wall.

From these results, we conclude that the monitor is sufficiently stable to normal matrix variations (density and composition) that can be expected under borehole measurement conditions. The calculations show that, as the borehole diameter is increased from 2.25 inches to 10 inches (with water replacing sandstone) and as the moisture content increases from 7 percent to 22 percent, the maximum monitor variation is ± 1 percent. The measurements show that complete removal of the water in the 4.5 inch borehole resulted in only a 2 percent variation from the mean. Furthermore, with no water in the borehole, moving the monitor against the borehole and then away from it showed a variation of ± 1.4 percent from the mean. It should be pointed out that the monitor was mounted at the outer surface of the sonde where it experienced much greater local variations than it would have if located inside the sonde, at the center and farther from the large local variations in scattering material (as it was in the calculational model, the results of which are summarized in Fig. 15).

3.3 "BACKGROUND" EFFECTS

The "background" pulses were observed separately from the "signal-plus-background" pulses of the ^{238}U -lined counter by viewing the amplifier output on an oscilloscope with one of the counters alternately removed. These were only ~ 10 percent of the "signal-plus-background" pulses, and they dropped off rapidly with increasing distance from the source. This indicates that the background could arise from 14-MeV neutron elastic scattering with gas atoms in the counter or from gamma rays. Measurements with varying amounts of lead between the sonde and the

monitor indicated that the background pulses were mostly due to 14-MeV-neutron elastic collisions with the counter-gas atoms. This means that this background effect will be very nearly proportional to the pulse from the sum of the ^{238}U fission pulses and need not be subtracted. The size of the background pulse was seen to vary rapidly with distance along the sonde, with the counter touching the side of the sonde (except for the heat-shrink tubing placed over the counter to avoid severe "ground loop" problems in the counter). Its variation in this case was approximately the same as that of the ^{238}U -lined counter, i.e., dropping off rapidly as the counter was moved away from the source toward either end of the sonde. The expected effect of rf pickup would be either a very slow variation with distance or no variation. Therefore, rf pickup outside the sonde is ruled out.

If, however, the monitor is eventually placed inside the sonde, rf effects might become large. An unlined counter should be used in initial tests, at least, along with a lined counter to measure any possible large rf pickup effect (and to facilitate shielding out this pickup, if present).

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4. SUMMARY AND CONCLUSIONS

The ^{238}U -lined counter serves well as a 14-MeV pulsed neutron source monitor with the counting arrangement shown in Fig. 17. The background pulse from the unlined counter is mostly due to 14-MeV-neutron gas-atom collisions and need not be subtracted since it is proportional to the summed fission pulses from the $^{238}\text{U}(\text{n},\text{f})$ reactions at the counter wall. Thus, the common-mode rejection configuration in Fig.17 can be eliminated and the circuitry correspondingly simplified.

The pulse height analyzer in Fig.17 can be replaced with a peak-reading voltmeter type of circuit, followed by a voltage-to-frequency converter chip and a timer-scaler, all of which can be mounted on a rather small printed circuit (PC) board. The operation would very much parallel that of the SAI portable adenosine triphosphate photometer (ATP) (used to measure trace quantities of life forms) which is a small and lightweight unit.

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APPENDICES

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Appendix I

Cross-Section Averaging for $1/v$ Absorber over a Maxwellian Thermal Distribution

Let σ_0 be the 2200 m/s cross section and $\bar{\sigma}$ the Maxwellian average thermal cross section. Then:

$$\bar{\sigma} = \frac{\int_0^{\infty} \sigma_0 \sqrt{\frac{E_0}{E}} \phi_M(E, T) dE}{\int_0^{\infty} \phi_M(E, T) dE} = \sigma_0 \sqrt{\frac{\pi}{4}} \sqrt{\frac{T_0}{T}} \quad (I1)$$

where

$$\phi_M(E, T) = \frac{\text{Const}}{T^{3/2}} E e^{-E/kT} \quad (I2)$$

and E_0 = energy of a 2200 m/s neutron

k = Boltzmann constant

$T_0 = 293.15^\circ\text{K}$ (room temperature).

For $T = T_0$ we have:

$$\bar{\sigma} = \sqrt{\frac{\pi}{4}} \sigma_0 = 0.886 \sigma_0 \quad (I3)$$

In addition, for a $1/E$ flux wing up to $\sim 5kT$, we obtain:

$$\bar{\sigma} = 0.8969 \sqrt{\frac{\pi}{2}} \sigma_0 = 0.795 \sigma_0 \quad (I4)$$

which is the effective $1/v$ thermal neutron cross section.

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Appendix II

Effect of Concrete Poisons on the Thermal Flux

In this section, we examine the effect of poison elements* in concrete and their effect on the thermal neutron flux. We also present a method to correct for this effect in order to compare with the analytic results which were obtained for a formation of sandstone and water without significant amounts of poison.

For the purpose of this discussion, we will use an analysis based on removal and age-diffusion theory. The general characteristics of the flux predicted by this analysis will be seen to justify its use.

We will write the thermal flux from a point source of strength S as the product of the fast flux, $\phi_f(r)$, and a buildup factor, $B(E, r)$:

$$\phi_{th}(E_{th}, r) = B(E_{th}, r) \phi_f(r) \quad (III1)$$

where ϕ_f is given by removal theory:

$$\phi_f(r) = \frac{S}{4\pi r^2} e^{-\Sigma_r r}, \quad (II2)$$

*high neutron absorbers

and Σ_r is the removal cross section. The buildup factor is given by age-diffusion theory:

$$B(E_{th}) = \frac{\Sigma_r}{\Sigma_a [1 - (L\Sigma_r)^2]} \exp [T_{th}\Sigma_r^2 - h(E_{th})] \quad (II3)$$

$$(\text{for } \Sigma_a/D > \Sigma_r^2)$$

and

$$B(E_{th}, r) = \frac{\Sigma_r}{\Sigma_a (L\Sigma_r - 1)} \exp [T_{th}\Sigma_r^2 - h(E_{th})] \quad (II4)$$

$$\{\exp [(\Sigma_r - \frac{1}{L})r]\}$$

$$(\text{for } \Sigma_a/D < \Sigma_r^2).$$

L is the diffusion length:

$$L^2 = \frac{1}{3\Sigma_a\Sigma_s(1 - \bar{\omega})} \quad (II5)$$

where $\bar{\omega}$ is the mean cosine of the scattering angle. The quantities T_{th} , age to thermal, and $h(E_{th})$, the inverse of the moderating ratio, are given by:

$$T_{th} = \int_{E_{th}}^{E_{max}} \frac{1}{3\xi\Sigma_s\Sigma_{tr}} \frac{dE'}{E'} \quad (II6)$$

and,

$$h(E_{th}) = \int_{E_{th}}^{E_{max}} \frac{\Sigma_a(E')}{\Sigma_s(E')\xi(E')} \frac{dE'}{E'} \quad (II7)$$

where Σ_a , Σ_s , Σ_{tr} are the absorption, scattering, and transport cross sections, respectively, and ξ is the mean logarithmic energy decrement. The radial dependence of ϕ_{th} is the same as that of ϕ_f in Equation (II2). The validity of this result is supported by the experimental data which show that the thermal and epithermal fluxes have approximately the same radial slopes (e.g., Fig. 10 and 11).

The variation of ϕ_{th} with Σ_a in Equation (III) through (II4) also suggests that the effect of thermal neutron absorption of parasitic contaminants can be "subtracted out" from the thermal neutron flux expression if the presence of these contaminants does not affect the transport of neutrons while slowing down to thermal. Thus, consider medium M that contains poison p and define a poison-free medium m such that:

$$\Sigma_a(M) = \Sigma_a(p) + \Sigma_a(m) \quad (II8)$$

Then, using Equation (III) through (II4), we can write:

$$\phi_{th(m)} = \frac{\Sigma_a(p) + \Sigma_a(m)}{\Sigma_a(m)} \phi_{th(M)} \quad (II9)$$

This procedure can be applied for comparing thermal neutron fluxes of two media that have similar neutron slowing-down properties but dissimilar absorption cross sections (e.g., sandstone and concrete).