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AUTHOR(S) Kenneth L. Coop
Paul E. Fehlau
Henry F. Atwater

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A NEUTRON PORTAL MONITOR FOR VEHICLES*

Kenneth L. Coop, Paul E. Fehlau, and Henry F. Atwater
Los Alamos National Laboratory
Los Alamos, NM 87545 USA

BSTRACT

'e have designed and built a portal vehicle monitor-ng system for detecting neutron-emitting special uclear material (SNM) such as plutonium. Monte arlo calculations were used to optimize the esign of the 15-cm-deep x 122-cm-high x 244-cm-ong detector chambers, which utilize ^{3}He proportion-l counters inside a hollow polyethylene box. esults for a variety of parametric studies, ncluding polyethylene thickness and detector umber, are described. Our experimental measure-ents are in good agreement with the computer alculations. The monitor's decision logic uses he Sequential Probability Ratio Test (SPRT) on oisson distributed counting data, which is superior o other statistical tests in many applications. e performed computer simulations of the SPRT ogic to determine expected false-positive decision ates. A controller unit of our design that uses his SPRT was built commercially. The cost of the omplete monitoring system is similar to that of ehicle portal monitors that detect gamma rays. his new neutron monitor can serve as an addition o standard gamma-ray vehicle portals or as a tand-alone portal monitor in particular safeguards onitoring situations. The monitor is being ested at Los Alamos and is scheduled for in-plant valuation at another DOE facility in 1987.

NTRODUCTION

he use of neutron detectors in safeguards portal onitors has been very limited compared with gamma-ay detectors. There are several reasons for his. In the case of uranium metal, essentially o neutrons are emitted, so neutron detectors are f no value. And plutonium emits many more gamma ays than neutrons, thus making gamma-ray detection onsiderably more sensitive in typical monitoring ituations. Another problem with neutron portal onitors has been their high cost compared with amma systems. This higher cost is generally ssociated with the large number of expensive eutron proportional counters typically used in uch systems.

However, if costs could be reduced, there are situations where a neutron portal would be the instrument of choice. For example, a few centimeters of lead or other high-Z materials can attenuate the gamma rays from plutonium so effectively that detection of the neutrons may become the more sensitive technique. The presence of such shielding is a potential safeguards problem when it cannot be detected readily by other methods such as visual inspection or the use of metal detectors, and a neutron portal monitor might solve the problem. It could be used in conjunction with a gamma-ray monitor, or as a stand-alone system for particular applications where moderate sensitivity to unshielded plutonium would be sufficient. Other monitoring conditions, such as high and variable gamma-ray backgrounds, might also favor the use of neutron detection systems that are insensitive to gamma rays.

With these considerations in mind, we have designed and built a vehicle portal monitor specifically to detect only neutrons. Our goal was to design a monitor that could be built at a price comparable to that of gamma-ray monitoring systems, with the best neutron sensitivity possible given the cost constraint, and to transfer the technology to the commercial instrument manufacturing sector. We concentrated our design efforts on the two critical components of the system: the neutron detector chamber and the monitor logic controller. The other electronic and mechanical components of the system are standard off-the-shelf items, similar or identical to those used in gamma-ray monitors we've previously designed.[1]

MONTE CARLO NEUTRON TRANSPORT CALCULATIONS

The starting point for our neutron detector chamber calculations was a rectangular box 122 cm high and 244 cm wide (4 ft x 8 ft), a standard size for commercially available polyethylene sheets, with only a few neutron-sensitive proportional counters located in a void inside. We used the MCNP Monte Carlo neutron transport code[2] to investigate the sensitivity of detection efficiency to variations in the depth of the inside void, the thickness of polyethylene on the front and back faces, the number of counters, the counter type (^{3}He or $^{10}BF_3$), and the amount of polyethylene moderator

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surrounding the fission-neutron source. Generally, enough neutrons were started to provide results with statistical standard errors in the range from 3 to 7%. The detector chamber geometry was set up with the long dimension placed horizontally on a large (effectively infinite, below the ground plane) concrete pad. The fission neutron source was centered on the front face at a distance of 229 cm (Fig. 1). Thus, the neutrons detected included those coming directly from the source and surrounding source moderator, and those reflected (after single or multiple collisions) from the concrete pad. The solid angle subtended by the detector chamber face at this distance was 0.49 sr. All proportional counters were 183-cm (6-ft) active length and 5 cm (2 in.) in diameter. They were symmetrically positioned in the void, with the distance between the top and bottom of the void and the nearest counter set at half the distance separating neighboring counters. The counter fill pressure used was 2 atm for ^3He and 1 atm for $^{10}BF_3$.

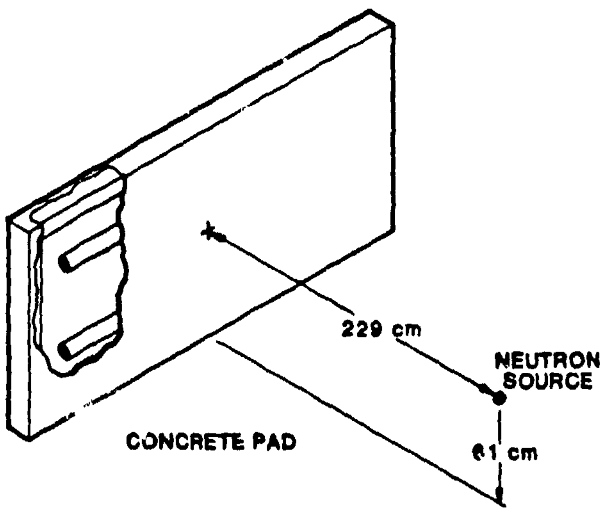


Fig. 1. Schematic of the geometry used for the MCNP calculations. The case shown is for two proportional counters. The chamber sits on a concrete pad that is effectively of infinite extent below the ground plane.

Selected results of the MCNP calculations are shown in Table I and Fig. 2. Table I contains calculated detection efficiencies using four ^3He proportional counters for a bare fission source or with the source surrounded by a polyethylene sphere with either a 5- or 15-cm radius, while varying front- and back-face thicknesses of polyethylene on the detector chamber. Increasing the back thickness from 5 cm to 10 cm should generally increase efficiency. This is demonstrated in the case of a bare source and thin front face (1.3 cm), where the effect is about 15%. We would expect fractional increases in efficiency due to this back-face thickness to be less for either a moderated source or a thicker front face. For a bare source, the thickest front face examined (5 cm) provides the best efficiency, whereas for the 5- and 15-cm moderated sources, the thinner faces are better. However, even with a source moderator 15 cm thick, the 1.3-cm-thick face provides better detection than having no front face.

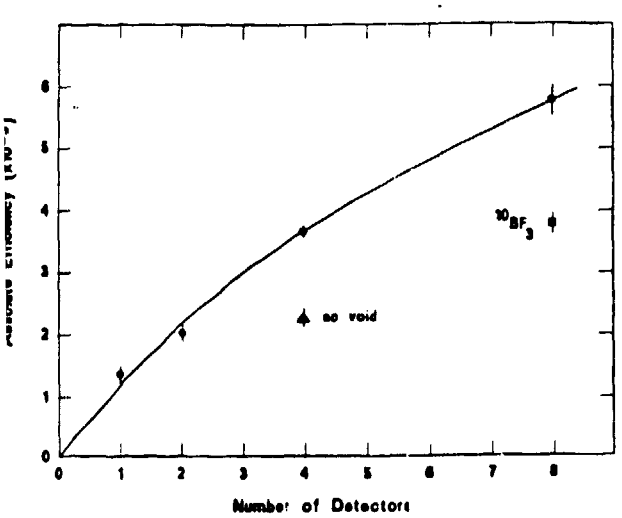
The above results were obtained with a void depth of 6.3 cm, just enough to accommodate the counters and mounting brackets. For comparison, we made

TABLE I

MONTE CARLO EFFICIENCY CALCULATIONS

Polyethylene Source Moderation Thickness (cm)	Chamber Back Face Thickness (cm)	Chamber Front Face Thickness (cm)	Summed Chamber Efficiency (counts per 10^3 neutrons)
0	5	3.7	3.8
0	5	2.5	3.4
0	5	1.3	2.4
0	10	1.3	2.8
5	10	5.0	2.9
5	10	2.5	3.7
5	10	1.3	3.7
15	10	2.5	0.6
15	10	1.3	1.0
15	10	0.0	0.7

Calculational setup: Four ^3He counters, each 183 cm long and 5 cm in diameter, with a fill pressure of 2 atm. The chamber sides (other than front and back) were 1.3 cm thick. The fission source was located 229 cm from the front face, 61 cm off the concrete pad. Inside void spacing between the front and back faces was 6.3 cm



'ig. 2. *MCNP calculated values for absolute fficiency. The circles are for ^3He counters ocated in a 6.3 cm x 230 cm x 109 cm void. A mooth curve has been drawn by eye through the ata points. The square is for $^{10}BF_3$ counters in he same void, and the triangle is for ^3He counters ith the void completely filled with polyethylene. he vertical bars represent the statistical standard rrors of the MCNP calculations.*

dditional MCNP calculations with four ^3He counters nd a bare fission source using void depths of 6.3 cm and 36.3 cm. The counters were centered ront-to-back in the void. The calculated efficieny for the 16.3-cm void depth was the same as for he comparable 6.3-cm case shown in the table, hile the efficiency for the 36.3-cm depth was 20% ess. Because increased depths result in increased osts and bulk (but don't improve efficiency based n these calculations), the 6.3-cm depth was used or all other calculations and was the approximate epth selected for the chambers we constructed.

igure 2 shows the effect of changing the number f ^3He counters when the source is moderated by cm of polyethylene and the front face thickness s 1.3 cm. In the range of 1 to 8 counters, doubling he number of counters increases efficiency by bout 50-70%. Also shown by the triangle is the ncreased efficiency for the 4-counter case that esults when the void is filled with polyethylene. e expect that filling the void with polyethylene i.e., embedding the counters in solid polyethylene) ould be less detrimental when more counters are sed and vice versa. The square point is for eight $^{10}BF_3$ counters, and it shows they give the same fficiency as four ^3He counters for this particular ase. Because the cost of each $^{10}BF_3$ counter is nly about one-third the commercial cost of a ^3He ounter* (about \$750 versus \$2300), it would be ore cost effective to use $^{10}BF_3$. However, we are ble to purchase the ^3He gas directly from the proucer at a reduced (government) price, which makes the cost of one ^3He counter the same as two BF_3 counters. Furthermore, the ^3He counters can be operated at much lower voltages (900 V versus 2100 V), resulting in fewer problems with spurious noise. For these reasons, we chose to use ^3He counters in our vehicle neutron monitor rather than BF_3 counters.

After deciding on the design of the neutron detector chamber (described in the following section), we set up the MCNP code to more closely approximate that configuration, including the exact wall thicknesses and compositions. With a ^{252}Cf fission source in a 5-cm-radius polyethylene sphere located at 229 cm from the front face of the detector chamber, we calculated an efficiency of 3.5 x 10^{-3} counts per emitted neutron, which we compared with the experiment below.

DETECTOR CHAMBER MEASUREMENTS

Based on chamber fabrication costs at Los Alamos versus proportional counter costs and efficiencies, we chose to use four ^3He counters per detector chamber (Fig. 3). We picked a front face thickness of 1.3 cm to optimize the chambers for moderately shielded (5-15 cm of polyethylene) neutron sources. The polyethylene thickness on the back and sides was fixed at 5 cm, with an additional 2.5 cm of borated (5% B) polyethylene on the outside of the back and sides to reduce background count rates. Based on measurements we made on a similar neutron detection system, we estimate that the borated polyethylene reduced background count rates by about 20%.

To provide weatherproofing, the entire chamber was enclosed in a 0.3-cm-thick aluminum frame (which is attached to a heavier framework around the sides of the chamber). A small ventilation fan was later added when corrosion was found on the ^3He counters after the detector chamber had been

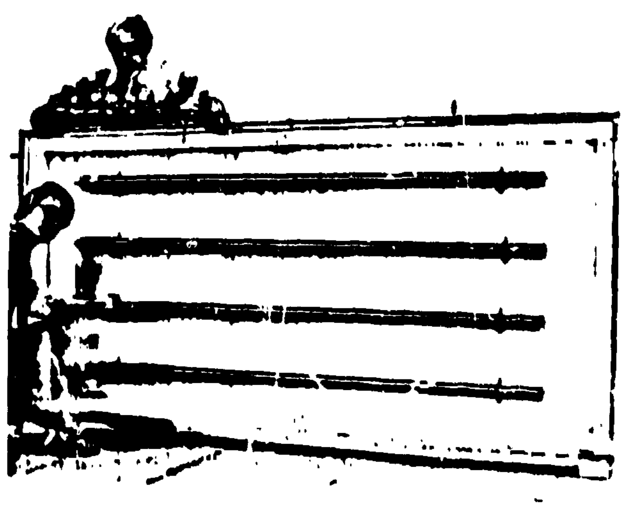


Fig. 3. Photograph of one of the two identical neutron detector chambers, with the front aluminum housing and polyethylene face removed.

*Reuter Stokes, Twinsburg, OH 44087.

outdoors for several months. The four proportional counters are connected in series to a standard preamplifier that is housed in a steel electrical junction box attached to the back of the detector chamber. The high-voltage, power, and signal cables from both chambers are routed to an all-weather electronics cabinet, containing a combined amplifier and single-channel analyzer, a high-voltage power supply, and the controller logic unit with its teletype. More details on these can be found in References 3 and 4.

To compare the Monte Carlo result described at the end of the preceding section with the experiment, we set one of the detector chambers on a concrete floor in our laboratory. Using a ^{252}Cf source in a polyethylene sphere with a 5-cm wall thickness and set at a distance of 229-cm, we measured a detection efficiency of 3.2×10^{-3} counts per emitted neutron. This is in excellent agreement with the calculated value of 3.5×10^{-3}, considering the potential sources of error, and supports our reliance on the MCNP calculations.

For test purposes, the two chambers were then set facing each other with a separation of 7.3 m (Fig. 4). The combined background count rate for the two chambers was approximately 60 c/s when measured outside our laboratory at Los Alamos, at an elevation of approximately 2100 m. We used a ^{252}Cf fission-neutron source to characterize the positional response of the detector chambers (Figs. 5-7). For the source centered between the detector chambers, 1.0 m from the road surface, the measured efficiency for the summed chambers was 2.15×10^{-3} counts per emitted neutron.

Based on a one-second count at this position and using four times the square root of the background, in this case $4 \times \sqrt{60}$, as the lower limit of detection, we calculated a source strength of about 1.4×10^{4} n/s as the minimum detectable. A vehicle passing through the center of the monitor at

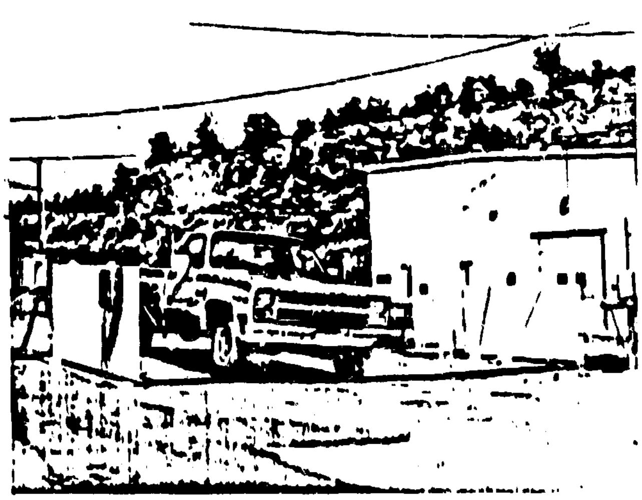


Fig. 4. Test setup of the neutron monitoring system at Los Alamos showing the two neutron detector chambers and, near the back chamber, the all-weather electronics cabinet.

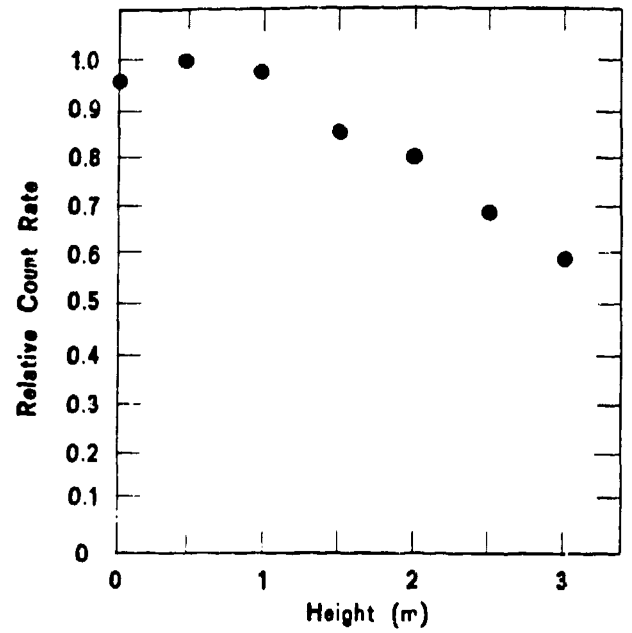


Fig. 5. Relative count rate in the center of the experimental setup (see text) as a function of height for a bare ^{252}Cf source. The statistical standard error bars are smaller than the circles.

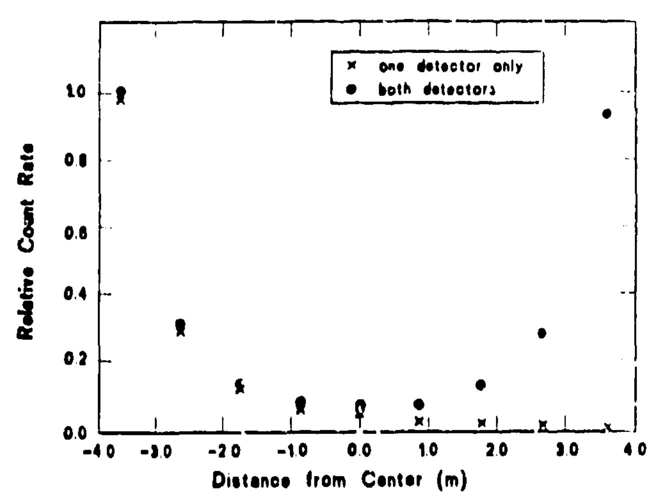
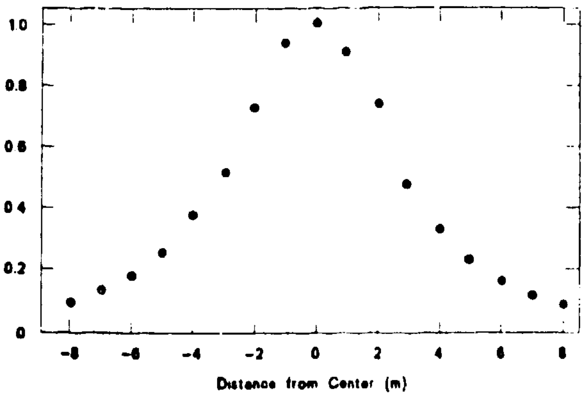


Fig. 6. Relative count rate for a bare ^{252}Cf as the source is moved along a line connecting the detector chamber centers. The bare ^{252}Cf source was 0.61 m from the ground. Statistical standard error bars are smaller than the circles.



g. 7. *Relative count rate for the detector amber pair as the source is moved along a line rallel to the detector faces and equal distance om them. The bare ^{252}Cf source was 0.61 m from e ground. Statistical standard error bars are aller than the circles.*

! m/s (5 mph) with such an unshielded source ild give approximately this signal while in the iter of the monitor. Integrating the signal ring the entire passage (Fig. 7) would decrease size of the minimum source that could be ected. However, a more substantial improvement ild be made by forcing the vehicle to pass ser to one of the detectors or by simply moving detector chambers closer together. For example, the spacing between the detector chambers was luced to 3.6 m, we deduce from Fig. 6 that the counts from the same size source at the center ild increase by about a factor of 4. Requiring vehicle to wait in the monitor would also prove detectability significantly. For example, 16-second long counting interval would also crease the lower limit of detection by another ctor of 4. (In such a wait-in monitor, additional tector chambers might be needed to adequately serve the entire length of a long vehicle, and might be desirable to test the output of each tector chamber separately, instead of summing em.)

erefore, with both these changes in geometry and unting times, a source emitting about 10^3 n/s ild be reliably detected in vehicles monitored these detectors. Of course, if the sources re heavily shielded with low-Z moderators, orer detectability would result, but even 15-cm-ick polyethylene on all sides of the source ild only cause an increase of a factor of 3 in minimum detectable amount (Table I). Several ntimeters of lead will have little shielding fect, as will the steel frames of ordinary hicles.

also expect to improve detectability when the nitor is used at facilities near sea level, cause cosmic radiation, a major source of back-ound neutrons, will be much less. We have yet measure the magnitude of this effect, but we pect the backgrounds to decrease by a factor of, perhaps, 4 or 5 at sea level. If the decrease were a factor of 4, the minimum detectable amount would be reduced by about 50%.

In gamma-ray portal monitors the presence of vehicles often suppresses the background significantly. In some cases, we have observed background suppression of 20% or more.[3] Because the gamma background rates can be relatively high, these percentages represent several standard deviations of background. This suppression significantly increases (by a factor of 2 or more) the signal required to trigger an alarm. For our neutron monitor, however, vehicle suppression of the background is relatively small ($\leq$ 3%) and the background count rate is much less, so there is no significant effect on detection due to the presence of the vehicle.

PORTAL CONTROLLER

The decision logic used in our monitor controller is the Sequential Probability Ratio Test (SPRT) for Poisson counting statistics.[5] We have shown that the SPRT makes decisions faster with fewer errors in many applications than the more familiar single interval test or sliding interval decision logic.[5,6] Our previous SPRT controllers[6,7] were designed for use in gamma-ray monitors, where background count rates were sufficiently high to approximate the counting statistics with a normal distribution. However, with neutron detectors, the background count rates may be so low that the normal distribution would not provide an adequate approximation. As a result, we modified our controller to use the SPRT on Poisson distributed background counts.[5] To our knowledge, this is the first time a Poisson SPRT controller has been used for nuclear counting applications.

The test parameter used in the Poisson SPRT is the logarithm of the probability ratio

$$Z_i = (M0 - M1)/N + [\ln (M1/M0)]\, C_i \quad , \qquad (1)$$

where M0 is the mean of the background distribution, N is a user selected parameter, and C_i is the gross counts obtained at step i of the sequential counting procedure. M0 is determined by the controller software from the background count rate, which is periodically updated. M1 is the mean of a nominal counting distribution corresponding to background plus a radiation source. Its value is dependent on M0, and α_0 and β_0, the nominal false-positive and false-negative detection probabilities, respectively.

The alarm and background decision levels, A and B, respectively, are determined by the equations

$$A = \ln [(1 - \beta_0)/\alpha_0] \text{ and} \qquad (2)$$

$$B = \ln [\beta_0/(1 - \alpha_0)] \quad . \qquad (3)$$

Details on the use of equations 1-3 in the Poisson SPRT are contained in Reference 5.

ince M0 changes with the background, it is frequently necessary to determine a new M1. To calculate M1 accurately after each background update for an arbitrary α_0 and β_0 would require an inordinate amount of time in our controller, using any methods of which we are aware. We have, therefore, restricted the number of (α_0,β_0) pairs permitted and have devised a procedure for determining the approximate value of M1 for integer M0 values as described below.

n our controller logic design, we initially selected $M0 = 64$, $\alpha_0 = 10^{-6}$, and $\beta_0 = 0.5$, typical values for our neutron monitor. By calculation, we determined that the value of M1 corresponding to these parameters was 101 (the nearest integer). We then picked 1 values of α_0, more or less evenly covering the range from 0.01 to 0.000004, and determined for each the corresponding β_0 that would give $M1 \approx 101$ for $M0 = 64$. This table of (α_0,β_0) values is incorporated in the controller's memory (PROM). When the user types in a desired value for the false-alarm probability on the controller's teletype, the program goes to the table and selects the nearest value of α_0 smaller than the input value; it then uses that and the corresponding tabulated value for β_0 to calculate the decision levels A and B.

nother table in memory contains (M0, M1) integer pairs that we calculated for M0 = 1,2,3,...,256, corresponding to $\alpha_0 = 10^{-6}$ and $\beta_0 = 0.5$. Each time the background of the controller is updated, a value is determined for M0 (exp), the experimentally determined background mean, which is normally non-integer. The controller selects the integer) value from the table nearest M0 (exp) and obtains the corresponding tabulated M1 value; the two table values are then used for subsequent calculations of Z_i. However, if M0 (exp) is greater than 256, it is used directly as M0, and a value is calculated for M1 based on the normal distribution, using the equation

$$M1 = 4.26\ (M0)^{1/2} + M0 \quad . \tag{4}$$

hus, this controller can be used for very low to very high count rates; the only restriction is $M0 \geq 1$. We can show that the equation for Z_i used here is equivalent to that used in our gamma-ray controllers, when the background count rates are large and, consequently, $M0/M1 \approx 1$.

should emphasize that the M1 values in the (M0,M1) tables are only approximations of the correct M1 values for particular (α_0,β_0) pairs. For $\alpha_0 = 10^{-6}$ and $\beta_0 = 0.5$, the approximations are very close to the true values, as they are at $M0 = 64$ for all (α_0,β_0) pairs. For M0 values far from 64, especially smaller values, and α_0 values considerably larger than 10^{-6}, the approximation may be poorer. To obtain some idea of what effect this might have on the SPRT, we performed a computer simulation of the Poisson SPRT for $\alpha_0 = 0.00125$, $= 0.125$, and N= 10, while varying M0 from 1 to 6 (in increments of 2 x M0). Over the entire range, the calculated false-alarm probability ranged from 4×10^{-4} to 7×10^{-4}, remaining reasonably constant and well below the nominal input α_0. Thus, it appears from this limited examination that the approximations used here do not have a significant detrimental effect on the SPRT.

The controller can be used in both the Singles and Continuous modes. The Singles mode is most appropriate for a monitor where vehicles stop and wait for a period of time.* The vehicle must wait until a single SPRT is completed, resulting in either an alarm or an all-clear signal. The Continuous mode is more appropriate for a drive-through portal and consists of repeated sequential tests until the vehicle clears the monitor or an alarm occurs. Several tests may be completed during the vehicle passage; a minimum of one test is always performed for each vehicle. Various audio and visual indicators inform the monitor's attendants (guards) of the status of the monitor and the results of the test (as in our previously described portal monitors[3,4]).

CONCLUSIONS

A complete neutron vehicle monitoring system has been built and assembled, and has operated continuously for about a year outdoors at Los Alamos. No significant problems have been noted, except for the corrosion problem, which was alleviated by ventilating the chamber. We expect the system to be easier to maintain and calibrate than gamma-ray portal monitors. Standard proportional counter electronics can be used in the system, and the neutron controller** and ^3He counters*** can be purchased commercially.

Although we built the detector chambers in-house their simple design can be easily reproduced by commercial vendors. In fact, if a production-model, weatherproof cabinet of the appropriate size can be obtained to replace the custom-built aluminum housing we used, we think fabrication costs might be reduced significantly. We estimate that the cost to duplicate our entire system would be about $30K plus the cost of installing it and fabricating the detector chambers.

Thus, we consider that it is quite feasible to use this neutron monitoring system as a stand-alone portal monitor for particular applications or as an additional safeguards portal monitor used in conjunction with gamma-ray monitoring equipment. We have scheduled an in-plant evaluation of our system at another Department of Energy facility later this year.

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*Note that if the user selects N=1 in the Singles mode and forces a decision after this step, the same result as the single interval test (SIT) is given. Thus, this test is also an option with the controller, even though it isn't explicitly indicated in the software menu.

**Jomar Systems, Inc., Los Alamos, NM 87544.

***Reuter Stokes, Twinsburg, OH 44087.

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REFERENCES

1. P. E. Fehlau, K. Coop, C. Garcia, Jr., and J. Martinez, "The Pajarito SNM Monitor: A High Sensitivity Monitoring System for Highly Enriched Uranium," Proceedings of the 25th Annual Meeting of the Institute of Nuclear Materials Management, Nuclear Materials Management, Vol. XIII, pp. 66-72, Columbus, Ohio (July 1984).

2. J. F. Briesmeister, Ed., "MCNP - A General Monte Carlo Code for Neutron and Photon Transport," Version 3A, Los Alamos National Laboratory report LA-7396-M, Rev. 2 (September 1986).

3. P. E. Fehlau, "An Applications Guide to Vehicle SNM Monitors," Los Alamos National Laboratory report LA-10912-MS (March 1987).

4. P. E. Fehlau, "A Low-Cost Safeguards Pedestrian Portal Monitor Using Chamber Neutron Detectors," 9th ESARDA Symposium on Safeguards and Nuclear Material Management, London, England (May 1987).

5. K. L. Coop, "Monte Carlo Simulation of the Sequential Probability Ratio Test for Radiation Monitoring," IEEE Trans. on Nucl. Sci., NS-32 (1), pp. 934-938 (February 1985).

6. P. E. Fehlau, K. L. Coop, and J. T. Markin, "Application of Wald's Sequential Probability Ratio Test to Nuclear Materials Control," Proceedings of the ESARDA/INMM Joint Specialists Meeting on NDA Statistical Problems, ESARDA 18, pp. 141-147, Ispra, Italy (September 1984).

7. P. E. Fehlau, K. L. Coop, and K. V. Nixon, "Sequential Probability Ratio Controllers for Safeguards Radiation Monitors," Proceedings of the Sixth Annual ESARDA Symposium on Safeguards and Nuclear Materials Management, ESARDA 17, pp. 155-157, Venice, Italy (May 1984).