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**Neutron Interrogator Assay System for the
Idaho Chemical Processing Plant
Waste Canisters and Spent Fuel:
Preliminary Description and
Operating Procedures Manual**

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NEUTRON INTERROGATOR ASSAY SYSTEM
FOR THE
IDAHO CHEMICAL PROCESSING PLANT
WASTE CANISTERS AND SPENT FUEL

PRELIMINARY DESCRIPTION AND OPERATING PROCEDURES MANUAL

by

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ABSTRACT

A neutron interrogation assay system is being designed for the measurement of waste canisters and spent fuel packages at the new Idaho Chemical Processing Plant to be operated by Allied Chemical Corp. The assay samples consist of both waste canisters from the fluorinel dissolution process and spent fuel assemblies. The assay system is a ^{252}Cf "Shuffler" that employs a cyclic sequence of fast-neutron interrogation with a ^{252}Cf source followed by delayed-neutron counting to determine the ^{235}U content.

I. INTRODUCTION

A. Assay Problem

In striving for improved nuclear material control and accounting, Allied Chemical Corp. (ACC) has identified an accountancy problem in the waste canisters from the Idaho Chemical Processing Plant (ICPP) fluorinel dissolver. The sludge is nonuniform and contains large quantities of zirconium, fission products, and thermal neutron poisons such

as boron and cadmium. This material is generated in the dissolution of high-enrichment spent fuel elements. No proven method or instrumentation exists for measuring this type of material.

The heterogeneities and high gamma-ray background ($\sim 10^3$ R/h) from the fission products preclude the use of passive gamma-ray assay techniques. Also, the large quantities of boron and cadmium neutron poisons preclude the use of thermal-neutron interrogation techniques. For these reasons, it was concluded that an assay system employing fast-neutron interrogation would be most appropriate.

During the nondestructive assay phase, the waste will be contained in a metal canister that is roughly 12 cm in diameter and 61 cm long, normally containing less than 300 g of ^{235}U . Those samples containing more uranium than 300 g will be recycled back through the dissolution process after measurement.

In addition to the waste canister measurement, the same assay station will be used to measure the ^{235}U content of spent fuel assemblies before dissolution. This fuel package analysis serves as an additional criticality check and it is not a requirement for facility operations.

B. Alternative Assay Methods

During the initial feasibility studies for the assay system, several alternative fast-neutron interrogation techniques were considered as follows:

- (1) Cf-252 neutron interrogation combined with delayed-neutron counting following interrogation,
- (2) ^{238}Pu -Li neutron interrogation combined with prompt fission neutron counting,
- (3) Photoneutron (e.g., ^{124}Sb -Be) interrogation combined with prompt fission neutron counting, and
- (4) 14-MeV neutron generator interrogation combined with delayed-neutron counting following interrogation.

Method (1) was chosen because Cf-252 has high reliability, convenient half-life (2.6 yr), high-neutron yield (10^{10} n/s for a 5-mg source), and relatively low gamma-ray output per source neutron. Method (2) suffered from low-neutron interrogation intensity with a maximum practical yield of $\sim 5 \times 10^6$ n/s, which would have difficulty overriding the expected neutron background from fission products in the sample. In general, the methods employing prompt fission neutron counting have a signal/background advantage over the delayed-neutron counting methods by a factor of 200 because of the prompt/delayed neutron ratio. However, this advantage is lost in method (2) from the low (α, n) source yields and for method (3) by the short half-life (60 days) for ^{124}Sb . The commercial neutron generators considered for method (4) have yields of 10^{10} - 10^{11} n/s, but more neutron tailoring material is required to shift the 14-MeV neutron energy below the fission threshold for ^{238}U and the resulting number of fissions per source neutron is then less than for ^{252}Cf of similar yield. Also, the 14-MeV neutron generators are not as reliable as radioactive sources.

If the neutron backgrounds from the waste canister are larger than expected, an ^{124}Sb -Be based system could be used to gain a more favorable signal/background ratio. For example a 100-Ci ^{124}Sb source would produce a neutron yield of $\sim 10^9$ n/s and when this is combined with the prompt/delayed-neutron factor of 200, it gives the same signal/background ratio as a 2×10^{11} -n/s ^{252}Cf source (100 mg). However, the operational difficulties in exchanging a 100-Ci ^{124}Sb source every 60-120 days are formidable.

II. OPERATING PRINCIPLE AND THEORY

A. Cf-252 Shuffler Principle

The "shuffler" is a modulated ^{252}Cf source assay system that repetitively transfers the neutron source from the inter-

rogation position to a shielded storage position while the delayed neutrons are counted.¹⁻⁹ The assay system includes a source shield tank, a decoupling CH₂ shield, and an irradiation area. The assay sample is irradiated in a chamber surrounded by ³He neutron detectors that are used to count delayed neutrons produced following each irradiation. The shuffler system has been evaluated at the Los Alamos Scientific Laboratory (LASL) for the assay of fuel rods, inventory samples, scrap and waste, uranium ore, irradiated fuel, and plutonium mixed-oxide fuel.¹⁻⁸

The cyclic transfer of the ²⁵²Cf source back and forth between the storage position and the irradiation position is accomplished by means of a Teleflex helix-wound cable driven by a stepping motor.¹⁰ A transfer time of ~ 0.45 s is achieved by using an electronic ramp-integrator for the stepping motor control. Each sample is exposed to multi-irradiation and count cycles until a sufficient number of counts have been accumulated for the required assay precision. An encoder on the stepping motor is used to digitally determine the position of the source in its irradiate and storage positions. For high-precision measurements both the source-transfer speed and the storage times are controlled by a crystal oscillator. A more detailed description of the shuffler's components is given in Sec. II.

One of the shielding considerations for a shuffler system is the distance between the storage position and the irradiation position, which partially determines the source-related neutron background in the detectors.¹¹ Figure 1 shows the background count rate in the ³He counter as a function of source position in the polyethylene shield. The sharp rise at ~ 0.90 m is caused by the transfer tube's line-of-sight path (tube i.d. = 16 mm) between source and detector for source positions closer to the sample than ~ 1 m. A shielded transfer distance of ~ 1.3 m reduces the source-related background to roughly the equivalent signal from 1-g ²³⁵U for fast-neutron interrogation.

B. Cycle Optimization and Scanning

Fig. 1. ^3He background count rate vs source position in the Modulated ^{252}Cf Assay System "Shuffler."

The evaluation of the shuffler has included measurements¹² and calculations¹³ to determine the optimum cycle timing for the delayed-neutron measurements. The cycle period is the sum of the irradiation time, count time, and two transfer intervals. The transfer time is ~ 0.45 s; therefore, the minimum cycle period is 0.90 s. To determine the optimum period for a 50% duty cycle (irradiation time equal to count time), a series of 200-s measurements were made using the present shuffler¹⁰ where the period ranged from 2 to 50 s. The sample was allowed to cool for > 3 min between runs to allow the delayed neutrons to decay to an insignificant level. The results of the delayed-neutron measurements and calculations are shown in Fig. 2 (Curve A), and the accumulated number of counts is maximum for a period of ~ 9 s. However, a longer period might be more de-

Fig. 2. Calculated and measured relative delayed-neutron yields for various periods and duty cycles from fast fission of ^{235}U .

desirable to increase the lifetime of the transfer system. For example, a 20-s period will reduce the number of accumulated counts by only 10%.

For samples with a low signal rate or a high-neutron background, such as plutonium and irradiated fuel, it is desirable to decrease the count time relative to the irradiation time to reduce the background contribution. Curves C and E in Fig. 2 show the effect of using other than a 50% duty cycle. Curve E demonstrates that a 25% count time (a factor of 2 reduction in background) reduces the accumulated counts by only ~ 30%.

Because the sample lengths range up to 3.0 m (10 ft), it was decided to operate the shuffler in a scanning mode. The

shuffler will be turned on when the leading edge of the sample enters the active region of the neutron interrogator, and the measurement will continue until the back end of the sample container has passed through the interrogation region. The interrogation zone is expected to have a full-width half-maximum (FWHM) of ~ 30 cm. The scan will include the full container length even though the canister or package is only partially filled. The computer will record and display the uranium content as a function of length for both the waste canisters and the spent fuel packages. This information can be used to flag any irregularities in the expected sample length or fissile distribution.

III. NEUTRON INTERROGATOR COMPONENTS

The neutron interrogator system for the Idaho flourinel facility consists of three major categories: (1) the interrogation chambers and surrounding components, (2) the measurement electronics, and (3) the data processor and communications terminals. The system has two interrogation chambers to allow assay of both waste canisters (15-cm-diam hole) and fuel elements (50.8-cm-diam hole). The components surrounding each assay chamber serve identical functions but have differing locations and configurations to enhance the measurement of the particular sample. A discussion of these components and of the interrogator system is presented below.

A plan view showing the interrogation chambers and surrounding elements is contained in Fig. 3. The major components for each assay chamber are:

1. Stainless steel tube,
2. Cadmium sleeve,
3. Flux monitor (^{235}U fission chamber),
4. Nickel and iron,
5. Lead and tungsten,
6. Source transfer tube, drive motor and Teleflex cable,

Fig. 3. Plan and sectioned view of the proposed Idaho Chemical Processing Plant (ICPP) Neutron Interrogator.

7. Polyethylene--CH₂,
8. Delayed-neutron detectors (³He), and
9. Sample position sensors.

The operating principle for the shuffler system is based upon fast-neutron interrogation of the sample followed by delayed-neutron counting, after the source has been removed to a storage position. The locations and types of materials used in the construction of the shuffler are important to achieve an assay system based on fast-neutron interrogation.

Presently, a two-dimensional discrete-ordinates transport code¹⁴ (TWOTRAN-II) and a three-dimensional Monte Carlo Code¹⁵ (MCNP) are being used to study the effects of differing materials and their locations on flux distribution for the interrogation of both fuel and waste canisters to enhance delayed-neutron assay counting. The results of these design analyses will aid in the choice of material type and their locations to improve the capability to consistently and accurately perform material assays. In addition, the location and response of detectors is being studied, using the codes, to reduce the effects of matrix composition, fuel element skewness, and sample multiplication.

A description of the location and the function of interrogator components follows:

1. Stainless Steel Tube. The tube provides physical separation between the dissolution cell and the interrogator cubicle. The purpose is to prevent contamination of the cubicle when contaminated samples are lowered into the interrogator chambers for assay. This tube is to be specified by R. M. Parsons Co. in their Title II design.

2. Cadmium Sleeve. The cadmium surrounds the outside surface of the stainless liner. Its function is to capture low-energy (thermal) neutrons that would otherwise enter the sample chamber. The purpose is to aid in criticality safety through neutron removal and also to ensure that sample interrogation is accomplished using fast-(high-) energy neutrons, which have better penetrability.

3. Flux Monitor (²³⁵U fission chamber). The ²³⁵U fission chamber is located between the stainless liner and the cadmium sleeve. This detector provides a measure of neutron spectral characteristics during irradiation of samples. The purpose is to detect the presence of hydrogen-bearing substances, such as water, that decrease the energy of the interrogation neutrons. The effect of water is to increase the fission rate (fission cross section of ²³⁵U increases as

neutron energy decreases), producing a larger delayed-neutron signal than would otherwise have been measured. Over a limited range the flux monitor can be used to correct for the effects of neutron spectral shift.

4. Nickel and Iron. These materials are placed between the ^{252}Cf source and interrogation chambers. Californium produces neutrons through spontaneous fission. A significant fraction of the spectrum exists above the fission threshold of fertile isotopes, such as ^{238}U . Nickel and iron serve to reduce the source spectrum through elastic and inelastic scattering without producing many thermal neutrons. These materials enable a high-energy penetrating flux to be maintained while reducing the response from fission of fertile isotopes in the sample.

5. Lead and Tungsten. Neutron-assay measurements in the presence of strong gamma fields suffer from gamma-ray-induced deadtime and pile-up effects. Lead and tungsten shielding, placed between the sample and the delayed-neutron detectors, are used to reduce these effects to acceptable levels. In addition, these materials serve to reduce the dose produced in the vicinity of the interrogator cubicle.

6. Source Transfer Tube, Drive Motor, Teleflex Cable. The source transfer tube provides a method to move the ^{252}Cf capsule between a storage position and sample interrogation location. Waste canisters are small in diameter, compared to the neutron penetrability, and the source can be held in one location during interrogation. However, the larger diameter of the fuel elements requires the source to be moved around the assay chamber to ensure more uniform interrogation of the fuel. This is accomplished using the drive motor, controlled by the data processor computer, to transfer the source, which is attached to one end of the Teleflex cable, around the fuel package tube at a specified rate.

7. Polyethylene (CH_2). Polyethylene is located on the outer regions of the interrogator and surrounds the He-3 detectors on all sides. The hydrogen content of the polyethylene moderates and thermalizes neutrons. This is a desirable effect because the efficiency of the He-3 detectors increases rapidly ($1/V$ cross section) with decreasing neutron energy. Placing polyethylene on the outside surfaces of the detectors returns some neutrons that would otherwise be lost. In addition, background neutrons that enter the interrogator from an exterior surface will be thermalized and have a large probability of capture in the polyethylene shield before reaching a detector.

8. Helium-3 Detectors. These detectors are cylindrical chambers, filled to 4 atm with He-3, and operated in the ionization region (~ 1500 V). The detection principle is based on the reaction ${}^3\text{He} (n,p) {}^3\text{H}$, which is an exothermic reaction with an energy yield of 764 keV.

The detector banks for fuel element assay are located in a symmetrical arrangement, at approximately 120° spacing, around the interrogation chamber. The large fissile loading of fuel elements requires fewer detectors for assay compared to the waste chamber detectors. The symmetrical arrangement aids in mitigating effects of skewness and shifts in sample location during assay.

The waste canister chamber is surrounded on three sides by detectors to provide maximum signal and to enable low fissile content measurements to be obtained. The waste canisters closely fit the dimensions of the chamber and special detector arrangements to balance skewness effects are not required.

9. Sample Position Sensors. Two sensors, based on monitoring a transmitted gamma beam, are required to detect when a sample has entered or left the interrogation scanning region. The sensors enable the assay to be started and stopped automatically. In addition, the transfer time of samples can be measured to check and correct for variations in

transfer speed, to monitor the length of samples, and to indicate if a sample is lowered into the wrong chamber.

The second category of the interrogation system consists of the measurement electronics. Details of the electronic modules for measurement are presented below. The various electronic modules consist of the following:

1. Detector high-voltage supply,
2. Preamplifiers,
3. Amplifiers,
4. Discriminators,
5. Scalers, and
6. Gating and control modules.

These units will be composed of both CAMAC and Nuclear Instrumentation Modules (NIM) connected to and controlled by the data processor. Each bank of detectors in the interrogator system will be operated separately to limit effects of counting rate deadtime and to allow comparisons between banks for consistency checks.

All electronics with the exception of the amplifier, Teleflex drive motor and power supply, and the detector preamplifiers will be housed outside the interrogator cubicle. The preamplifiers will provide impedance matching and drive capabilities to supply detector pulses to the amplifier modules. The amplifiers will shape the preamp pulses and provide pole-zero compensation and baseline restoration to improve count-rate capabilities. Small pulse heights, characteristic of gamma events, will be blocked by the discriminators, and only large events, induced by neutrons, will be passed to the scalars.

The measurement process will consist of enabling the flux-monitor scalars and disabling the delayed-neutron scalars during sample irradiation. Following movement of the source to the storage position, the flux monitors will be disabled

and the delayed-neutron counters enabled. Module gating, scaler readout and all assay functions will be controlled by the computer.

The third category of the shuffler system consists of the data-processing computer and communications terminals. Appendix B outlines the operational sequence and software interface for sample assays. The processor will be a Digital Equipment Corporation LSI-11 microcomputer containing up to 32-K memory of 16-bit words. Communications to equipment modules will use standard interface cards connected to the Q-bus. Scalers and measurement electronics may be connected and controlled through the CAMAC dataway or packaged within the processor's mainframe.

IV. EFFECTS OF VARIABLES ON ASSAY

A. Moisture Effects on the Assay of Waste Canisters

One of the questions raised concerning the measurement of waste from the fluorinel process has been the effect of moisture in the waste canister on the assay. Monte Carlo calculations were performed to study this effect on measurements with the ^{252}Cf shuffler.

The geometry was modeled using concentric cylindrical shells. The waste was contained in a cylindrical canister, 12.7 cm in diameter and 60.9 cm in height. The canister was surrounded by a thin layer (5×10^{-6} cm) of ^{235}U , simulating a fission monitor containing 100 mg of ^{235}U , followed by a 0.8-mm-thick layer of cadmium. Next there was a 15.2-cm-thick cylindrical shell of lead and another 0.8-mm-thick layer of cadmium, followed by ^3He imbedded in polyethylene. The inner layer of polyethylene was 2.54 cm thick; the shell of ^3He was 0.84 cm thick with 4 atm pressure; and the outer layer of polyethylene was 6.78 cm thick. This arrangement gave the same amount of ^3He as would be in the actual ICPP case of 27 ^3He tubes 2.54 cm in diameter and 30.48 cm in

length filled to a pressure of 4 atm. For the neutron interrogation, a point ^{252}Cf source was located 15.32 cm from the center of the waste canister.

The waste from the ICPP resulted from dissolving spent fuel elements in a fluorinel process. The highly radioactive material contained ZrO_2 , iron, and uranium, as well as some fluorine. Additionally there was some boron and cadmium to act as neutron poisons for criticality control. For the calculations, the mass of ZrO_2 plus the mass of iron was assumed to be 45 kg; 97% of this was ZrO_2 and 3% of this was iron. There were 100 g of ^{235}U and 5.26 g of ^{238}U in the waste canister.

Two series of calculations were made. In the first series, there were always 176 g fluorine, 13.9 g boron, and 34.8 g cadmium in the waste. This corresponds to four molar fluorine, 6 g/l boron, and 15 g/l cadmium in a canister having 30% water by volume. The only variation from one calculation to another was the amount of water in the canister. The first calculation used 30% water by volume; subsequent calculations used less water, simulating the drying of the waste. The events that were tallied by the Monte Carlo code were the number of fissions occurring in the waste and in the ^{235}U fission monitor. The number of fissions occurring in the monitor was applied as a correction factor to the number of fissions occurring in the waste. The relative number of fissions in the waste as a function of the amount of water in the waste canister is shown in Fig. 4. Also shown in the figure are the fission counts in the waste corrected by the fission monitor events.

The number of fissions in the waste differs by more than a factor of 10 as the amount of water in the waste canister varies from 0% to 30%. When these counts are corrected by the fission monitor counts, this factor reduces to 1.8. It is seen that the fission monitor corrects for some of the effect of the water, but there is still a noticeable effect on the

Fig. 4. The relative number of fissions in the waste canister (solid curve) and the relative number of fissions in the waste canister corrected by the fission monitor (dashed curve) as a function of the amount of water in the canister. The canister had a four-molar fluorine concentration and no boron or cadmium.

corrected response of the assay instrument when there is more than 5% water by volume in the waste canister. As long as the total amount of water in the waste canister is less than 5% by volume, the fission monitor can be used to correct for the effect of water.

The second series of calculations studied the effect of poisons, boron, and cadmium, on an assay. For this series of calculations there was no boron and cadmium in the waste, and the fluorine concentration was four molar. This series of calculations, when compared to the preceding series with poison, indicates the maximum effect of boron and cadmium.

The results of these calculations are shown in Fig. 5. The relative number of fissions in the waste and the relative number of fissions in the waste corrected by the fission monitor events are shown. The number of fissions in the waste increases by more than a factor of 20 as the amount

Fig. 5. The relative number of fissions in the waste canister (solid curve) and the relative number of fissions in the waste canister corrected by the fission monitor (dashed curve) as a function of the amount of water in the canister. In all cases there were 176 g fluorine, 13.9 g boron, and 34.8 g cadmium in the canister.

of water in the waste canister increases from 0% to 30%. When these counts are corrected by the fission monitor, this factor becomes 2.5. These numbers are to be compared to 10 and 1.8, respectively, from the case with neutron poison. For moisture contents less than 5% by volume, the boron and cadmium seem to have very little effect on the assay when corrected by the fission monitor. But in general over the range of water concentrations considered here, the fission monitor does not do as well correcting for the water when there is no boron and cadmium. The poisons, besides their use for criticality reasons, also help to reduce the sensitivity of the assay instrument to water.

B. Skewness Effects

Problems with skewness are limited to the fuel elements. Waste canisters have a fixed geometry and the interrogation chamber has been sized to the canister diameter. However, fuel

elements may vary in both size and shape enabling various off-center positions to be realized when lowering samples through the chamber for measurement. Placing a sample in various positions within the chamber, relative to the detectors and interrogation source, could produce a range of counting values. The net effect could be to induce a bias in a calculated assay.

Mitigation of skewness effects is being accomplished by two methods. First, the ^{252}Cf source will be cycled around the circumference of the interrogation chamber. This movement will provide a sample irradiation from all sides of the chamber and decrease matrix effects in addition to nonuniform irradiation effects for samples located along any side of the chamber. Second, the three delayed-neutron detector banks are located symmetrically around the chamber, spaced approximately 120° apart. The movement of a sample in a direction away from one detector bank will produce a lower count rate in that bank. However, the other detector banks will have a compensating effect through increased count rates. The overall purpose is to produce a measured assay independent of sample location or movement within the interrogation chamber.

Presently, transport codes are being used to study flux distributions for various positions of a sample in the interrogation chamber. These calculations are being used to quantify and mitigate the effects of skewness on an assay.

C. Thermal-Neutron Poisons

The presence of thermal-neutron poisons such as cadmium and boron in the waste canisters dictated the use of a fast-neutron interrogation system. In addition, the walls of the interrogator chambers are lined with cadmium and possibly boron to absorb scattered neutrons with energies low enough to be affected by these poisons. The fast-neutron flux is not significantly affected by the thermal-neutron poisons and this flux is required to give good penetration for the high fissile loading in the fuel packages.

If there is hydrogen present in the fuel packages or waste canisters (e.g., moisture), a fraction of the fast-neutron flux is slowed down from scattering reactions with the hydrogen. The neutrons with these lower energies then have a much higher probability for causing fission reactions in the ^{235}U or for being absorbed by the Cd and B. The fission chamber monitor is used to measure and correct for this problem as described in Sec. III.A.

D. Curium and Other Neutron Backgrounds

There are three sources of neutron backgrounds, (1) the ^{252}Cf source in its storage position, (2) the external room background, produced primarily from the dissolution cell, and (3) the neutron background from the assay sample positioned inside the interrogation tube. The first of these background sources is reduced to a negligible level by the location of the shuffler shielding material as shown in Fig. 3. Note that if the ^{252}Cf source strength increases by a factor of 10, both the signal and background from this source increase by the same factor and their ratio is not altered.

The second source of room neutron background is expected from fuel packages that are not in the interrogator tube and also from cosmic-ray interactions in the lead shielding. This background level is reduced by the thick walls of the interrogator cubicle and exterior CH_2 shielding behind the ^3He detectors.

By far, the most serious neutron background source is expected from the assay sample itself. These neutrons originate from the spontaneous fission of ^{242}Cm and ^{244}Cm as well as possible (α, n) reactions. It is not possible to shield against these neutrons without eliminating the induced delayed neutrons that are used for the assay. However, for a 3-m-long fuel package, only the 0.3- to 0.6-m section of the package in the active region of the interrogator will significantly contribute to the background. The remaining sections of the

fuel package are shielded from the ^3He detectors by CH_2 and Cd. The signal/background ratio from this background source is independent of the ^3He detector efficiency, but it is directly proportional to the ^{252}Cf source strength and the ^{235}U content. Thus, it is necessary to choose a ^{252}Cf intensity that is high enough to override this background. The source being considered for the interrogator is 5 mg of ^{252}Cf , which gives a neutron yield of 10^{10} n/s. Signal/background ratios as low as 1/20 can be used because it is possible to accurately measure the background before the neutron interrogation is started. However, it is desirable to have a higher signal/background ratio to improve the accuracy of the measurement.

The neutron background levels from the spent fuel are not well known at this time. Measurements are planned, in the near future, to better establish the neutron and gamma-ray backgrounds from actual spent fuel assemblies.

E. Multiplication

During the moisture studies for the waste canisters, the multiplication was calculated as a function of H_2O content. The waste canister was assumed to contain 100 g ^{235}U , 176 g fluorine, 13.9 g boron, and 34.8 g of cadmium. The remainder of the mass was ZrO_2 and iron giving a total mass of 45 kg. The only material that was varied from one calculation to another was the amount of water in the canister.

The results of the calculations are given in Table I. The neutron leakage multiplication goes from about 0.2% to 0.5% as the moisture increases from 0 to 30% by volume. However, this multiplication effect is more than offset by the reduction in signal counting efficiency. This reduction is caused by the shielding effect of the additional H_2O in the canister that reduces the number of neutrons making it to the ^3He detectors.

TABLE I

ICPP WASTE CANISTER MULTIPLICATION

<u>% H₂O</u> <u>by Volume</u>	<u>Multiplication</u>	<u>He captures/ Source Neutron</u>
0	1.0019	0.0821
1	1.0019	0.0822
3	1.0020	0.0818
5	1.0020	0.0816
10	1.0024	0.0814
20	1.0037	0.0802
30	1.0052	0.0785

The calculations for the multiplication of fuel packages have not yet been completed; however, similar studies have been performed for a shuffler to be evaluated at the Savannah River Plant (SRP). The shuffler in this case also has a fast-neutron interrogation flux, and the delayed neutrons are counted using the same type ³He detectors. The samples at SRP have a diameter of 14 to 18 cm and a maximum length of 30 cm. The mass of these samples ranged from 0.3 to 3.1 kg of ²³⁵U and the maximum detector response due to multiplication was calculated using Monte Carlo to be 1.064. For the ³He detector response, this multiplication effect was more than offset by the increase in neutron self-shielding, calculated

to be 1.090, from the ratio of fissions in outer to inner regions of the sample. These calculations indicate that self-shielding more than compensates for increases in neutron response due to multiplication for ^{235}U loadings up to 3.0 kg in the shapes described. The ^{235}U loading per unit length of the SRP samples is about 10 kg/m, which is higher than the typical loading of a spent ICPP fuel package. Also, the diameter of the SRP sample is smaller than that for the ICPP fuel package, which increases the multiplication for a given amount of ^{235}U .

These are indirect arguments to help establish the bounds on the expected multiplication. The actual multiplication for the ICPP fuel packages will be calculated in the near future. In addition, the ICPP neutron interrogator will be calibrated using spent fuel with the same nominal multiplication as the unknown fuel packages. Thus, the multiplication perturbation will be included in the calibration function for each type fuel package.

V. CALIBRATION AND STANDARDS

A. Waste Canisters

The calibration standards and procedure will be considerably different for the waste canister as compared to the spent fuel packages. For the waste canisters, no chemical analysis of the actual waste material is possible, and thus standards will be prepared that approximate the waste material. Appropriate amounts of ZrO_2 , Fe, F, B, Cd, etc., will be prepared by ACC in typical waste canisters. Three of these type canisters will be prepared containing 100, 200, and 400 g of ^{235}U to establish a calibration curve for the interrogator. After the standards have been carefully measured to determine the calibration function, it is only necessary to routinely measure one of the standards on a day-to-day basis to normalize the performance of the system.

A few test canisters will be prepared that vary the hydrogen content, poison content, and geometric profile of the uranium loading to check the sensitivity of the assay system to these variables. As far as possible these tests will be performed before the installation of the interrogator at the ICPP location. Because of the fast-neutron interrogation, the complete scan of the canister, and the fission chamber flux monitor, it is not expected that the assay system will be very sensitive to these variables (see Sec. III.A on moisture effects and Sec. III.C on neutron poisons).

B. Fuel Packages

The standards and calibration problem is complicated because of the wide variety of fuel package geometric configurations. It is expected that the interrogator's data processor will contain a separate calibration function for each major type of fuel package. If possible, a standard made out of unirradiated uranium will be fabricated for the most prevalent type fuel package. This standard would contain 10-11 kg of ^{235}U and would be used to help establish a preliminary calibration curve when combined with calculations of response as a function of loading.

After the ICPP is operational, more accurate calibration parameters will be obtained from feedback from the dissolution accountability system. For example, when a group of a particular fuel package type is to be processed through the facility, two or more of the samples would be carefully measured relative to each other in the neutron interrogator. Then one of these spent fuel samples would be saved as a standard and the others would be sent through the dissolution process. The subsequent chemical analyses would establish the uranium content in the destroyed fuel packages and thus determine the calibration parameters for the NDA system. The fuel sample that was saved would then serve as a normalization standard for any future measurements of that type fuel package.

This type of calibration procedure has the advantage of including the appropriate neutron absorption, self-shielding and multiplication effects in the calibration function. After obtaining some operational experience with the NDA system, it is expected to give a consistency check on the chemical analyses as well as an initial ^{235}U value before the fuel assembly is placed in the dissolution tank.

APPENDIX A

ASSEMBLY AND MAINTENANCE

I. INTERROGATOR COMPONENTS AND ASSEMBLY

The interrogator unit will be shipped to ICPP in an un-assembled condition. This will facilitate transfer to and assembly in the assigned cubicle. All components are designed so two workmen can manually carry the individual pieces from wheeled carts located outside the cubicle and put them in place inside the cubicle. It is anticipated that no piece will be heavier than 100 pounds.

As a basis for this report consider the following sequence of assembly:

1. Install the main base plate on the anchor foundation.
2. Install cadmium around the 5-in and 20-in sample tubes.
3. Install the lead shielding around the 5-in and 20-in sample tubes.
4. Attach the main structural framework to the base plate.
5. Install the source-transfer tube around the 20-in sample tube.

6. Put the tungsten source-shield over the transfer tube.
 7. Install the nickel reflector over the transfer tube.
 8. Install the three lead shields around the 20-in sample tube.
 9. Install the 6-in-thick lead shielding around the 5-in diam sample tube
 10. Put the six detector banks in place. (These have previously been covered with cadmium.)
 11. Install the lithium-loaded polyethylene around the tungsten source shield.
 12. Install the polyethylene shielding using coded pieces as a guide.
 13. Install the sample position sensors.
 14. Run the signal cabling to the detector banks, fission chambers, and position sensors.
 15. Install the fission chambers.
 16. Attach outer metal skin around sides and on top of the unit.
 17. Transfer Cf-252 source into the unit using Standard Operating Procedure--Appendix A, Sec. B.
 18. Run all power and signal cables for the source-drive motor and position sensors.
- Major components and associated weights are listed below.

<u>Component</u>	<u>Weight</u>
Lead surrounding fuel analysis tube	9840
Lead protecting fuel analysis detectors	880
Lead surrounding waste canister analysis tube	3000
Additional lead shielding on waste canister tube	1112
Polyethylene	6620
Nickel reflector	160
Fuel detectors--9 @ 3 lbs each	30
Waste canister detectors--27 @ 3 lbs each	80
Tungsten surrounding source	100
Outer 1/8" aluminum skin (probably should be S.S.)	250 +
Aluminum baseplate (probably should be S.S.)	300 +
<hr/>	
Total interrogator weight	22 372 + lbs

II. PROCEDURE FOR TRANSFER OF Cf-252 SOURCE FROM SHIPPING CASE TO ICPP NEUTRON INTERROGATOR

The source cask is shipped with a removable support stand that keeps the wheels clear of the floor and provides protection to the source drive cable at the bottom of the cask. The cask is designed to be lifted by fork-lift from the bottom. The support may be removed while the cask is lifted. This allows the cask to be lowered onto its wheels for transport.

Refer to Fig. A-1 for part numbers called out in the procedure.

Loading Procedure

1. Transport the source cask by fork-lift to the ICPP load-out elevator at the 0-0 level. Remove the support stand (1) before lowering the cask onto the elevator.
2. Lower the elevator to the 13 ft -0 level and push the cask by hand to a predetermined position adjacent to the Neutron Interrogator.
3. Unlock and remove the cask shield-plug (2).

Fig. A-1. $^{252}\text{Source}$ Transfer Schematic.

4. Attach the source guide tube (3) between the top of the cask and the front of the Neutron Interrogator using the bolted flanges.
5. Remove the bolted cover (4) from the cask source drive-shaft and engage the drive crank (5).
6. Remove the cable protector (6) at the bottom of the cask and allow the Teleflex cable (7) to uncoil and straighten on the floor.

7. Clear the immediate area of all personnel except for one operator for the crank drive (5). The operator will assume a low position keeping the shield cask between himself and the source guide tube (3). This will provide shielding while the source is in transit from the cask to the interrogator unit.
8. The operator will turn the crank handle (5) until the source cable is completely disengaged from the cask Teleflex drive mechanism. The source at this time will be safely inside the interrogator unit. The operator may now assume a standing position and continue the procedure.
9. Unbolt the guide tube (3) and manually secure the Teleflex cable at position (8). Carefully remove the guide tube (3) while allowing the source cable to slide from inside it.
10. Place keeper-plug (8) over the end of the source cable and slide the plug into the end of the source tube (9).
11. Engage the interrogator source drive motor (10) with the end of the source cable (7) and manually move it along the cable until it is in place for bolting to the interrogator. The source is now in place and engaged by the drive motor mechanism.

During normal operation, personnel are excluded from the interrogator cubicle. The interrogator will be installed with no source present. When the shuffler system is ready for checkout, LASL personnel will transfer the Cf source from its transportation cask into the interrogator shield. After the system is operational, the interrogator cubicle will only be entered for maintenance and repair.

The frequency of this activity is estimated to be less than four times per year. All parts that might require repair such as the detectors and the shuffler drive mechanism are designed to be easily removed from the interrogator for repair

or maintenance outside the cubicle. A typical repair sequence for a malfunctioning detector would be to enter the cubicle and walk past the Cf storage position to the detector end of the interrogator, remove a small amount of CH shielding, and lift the detector out of the interrogator. Then a new detector unit would be inserted in place of the original unit. The person would then leave the cubicle walking past the Cf shield position. The minimum time in the cubicle for this operation is estimated to be 10 min and the maximum time 60 min. The time in the region of maximum dose (i.e., 25 inches from the Cf source) would be less than 1 min. Repair or maintenance of the shuffler drive mechanism would take a minimum of 30 min and a maximum of 2 h. Any massive repair or change in the shuffler drive system would require the removal of the Cf source to its transfer cask outside the interrogator cubicle, and thus would not influence the shuffler shield thickness.

APPENDIX B

OPERATIONAL SEQUENCE, DATA HANDLING, AND SOFTWARE INTERFACE

The purpose of this appendix is to (1) outline the operational sequence for interrogation of ICPP fuel or waste, (2) describe the identification and handling of assay data, and (3) propose the software interface to facilitate communications between the interrogator processor and main ICPP computer.

Software Operating System

The software operating system presently under development for the ICPP neutron interrogator is being designed to reduce required operator communications and enable easy and concise

operation. The operating system will collect, from the interrogator electronics, all measurement data and perform the computations to determine the fissile content of measured samples. In addition, the software will be used to monitor system performance and help provide detection and identification of failing components. Errors in sample identification for assay purposes will be reduced by requiring redundant identification of samples prior to assay.

I. OPERATIONAL SEQUENCE

Communication and operation of the interrogator will be accomplished using the CONTROL or the OPERATIONS terminals. The OPERATIONS terminal is to be located in the crane room to allow the interrogator operator a direct view, via a shielding window, of sample location and movement during assay operations.

The interrogator CONTROL terminal will be used to enable or disable the OPERATIONS terminal. Before assay operations, the identification numbers for fuel packages and waste canisters can be input to the interrogator processor at the CONTROL terminal in the sequence they are to be assayed. Any changes in the assay sequence or the identification number for samples can be restricted to the CONTROL terminal. The output of assay results will be sent to this terminal, which can also be used to monitor the operational state of the interrogator. The CONTROL terminal will not be capable of initiating an assay.

The OPERATIONS terminal will be used to perform all sample assays. Commands through this terminal will be capable of performing assays, calibrations, and routine checks of the interrogator system, when enabled by the CONTROL terminal. The crane (or assay) operator might be required to correctly identify each sample in a sequence previously specified at the CONTROL terminal. This identification procedure is proposed to limit the possibility of an assay being associated with AN INCORRECT SAMPLE.

II. DATA HANDLING

Calibrations, system control, and determinations of fissile content will be performed solely by the neutron interrogator computer and associated equipment. During or following a measurement, if the processor detects malfunction in any components, the operator will be notified by terminal messages of the problem. All data corresponding to a malfunction measurement will be stored on the interrogator floppy disk for later retrieval and analysis. Following successful measurements, collected data will also be stored on floppy disk to enable a system performance to be continually evaluated and to provide a base for comparing and classifying any system problems. Identification numbers of fuel packages and waste canisters will not be retained by the interrogator processor and will not be stored on floppy disk but will be transferred to the main ICPP computer.

Transfer of assay data to the ICPP computer will occur only following successful measurements. The data to be submitted will consist of (1) the sample identification number, (2) the fissile content of the sample, and (3) the uncertainty in the fissile content.

Following the completion of each measurement, the operator will be prompted to proceed with operations or to repeat the measurement.

III. SOFTWARE INTERFACE

One interface is required to permit the interrogator processor to communicate with the main ICPP computer. The communications line will operate through a serial 20 milliamp current loop interface at 4800 baud. Information will be transferred with ASCII code. Communications will be initiated by a software interrupt request from the interrogator processor to the main computer. Three pieces of data will be sent in the following order:

SAMPLE IDENTIFICATION	(CARRIAGE RETURN)
FISSILE CONTENT	(CARRIAGE RETURN)
FISSILE UNCERTAINTY	(END OF TRANSMISSION)

Following each piece of data a carriage return symbol (ASCII code--octal 004) will be sent. An end-of-transmission (EOT) character (ASCII code--octal 004) will be sent following the last piece of data. Characters received by the main computer will be echoed and checked by the interrogator for accuracy in transmission. Detection of incorrect characters will result in a cancel (ASCII code--octal 030) character being transmitted followed by the correct character again being transmitted. In the event of inability to transmit information, the control room personnel will be notified by messages at the control terminal.

APPENDIX C

REFERENCES TO OTHER SHUFFLER SYSTEMS

Prior to the design of the ICPP neutron interrogator, considerable operating experience has been obtained on closely related shuffler systems. The original ^{252}Cf shuffler assay system was developed in 1974 for the assay of samples ranging in size up to 1-gal cans.¹⁰ This system, shown in Fig. C-1, has the option of a fast-neutron or thermal-neutron interrogation by changing the neutron flux tailoring materials around the neutron source. The sensitivity limit of this shuffler is 40 mg ^{235}U for fast-neutron interrogation, which is much better than the requirements for the ICPP shuffler system. Detailed information concerning this system is given in references 8 and 10.

The above shuffler interrogator was modified in 1976 for the assay of 55-gal barrels containing plutonium mixed with fission product contaminated waste.¹⁶ The system was designed for the measurement of barrels with a high gamma-ray background (10^3 R/h), and thus the sample cavity is lined

Fig. C-1. Plan and front views of the modulated ^{252}Cf assay system "shuffler."

with 15 cm of lead to protect the ^3He detectors from the high gamma dose. This prototype shuffler system is being evaluated for the assay of small amounts of plutonium in scrap and waste so, to obtain high sensitivity (~ 50 mg Pu), thermal-neutron interrogation is used. A schematic diagram of this unit is shown in Fig. C-2.

Fig. C-2. Barrel scanning ^{252}Cf assay system "shuffler."

Starting in July 1978, a LASL-designed shuffler-type assay system will be evaluated at the SRP for measurement of highly enriched uranium materials.¹⁷ The unit employs cyclic fast-neutron interrogation and delayed-neutron counting to assay the ^{235}U content of fuel alloys with up to 2.4 kg ^{235}U per sample. The accuracy goal for the measurements is 1-3%. A schematic diagram of the system is given in Fig. C-3.

This SRP unit will employ a ^{252}Cf source, source drive system, and electronic control system that is very similar to the planned ICPP interrogator system. Thus, the results of the SRP evaluation will be valuable in the development of the interrogator for ICPP.

Fig. C-3. Savannah River Plant (SRP) Shuffler.

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