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NEUTRON AND GAMMA-RAY NONDESTRUCTIVE EXAMINATION OF CONTACT-HANDLED TRANSURANIC WASTE AT THE ORNL TRU WASTE DRUM ASSAY FACILITY

F. J. Schultz
D. E. Coffey
L. B. Norris
K. W. Haff

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NEUTRON AND GAMMA-RAY NONDESTRUCTIVE EXAMINATION OF CONTACT-HANDLED TRANSURANIC WASTE AT THE ORNL TRU WASTE DRUM ASSAY FACILITY

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ABSTRACT

A nondestructive assay system, which includes the Neutron Assay System (NAS) and the Segmented Gamma Scanner (SGS), for the quantification of contact-handled (<200 mrem/h total radiation dose rate at contact with container) transuranic elements (CH-TRU) in bulk solid waste contained in 208-L and 114-L drums has been in operation at the Oak Ridge National Laboratory since April 1982. The NAS has been developed and demonstrated by Los Alamos National Laboratory (LANL) and the Oak Ridge National Laboratory (ORNL) for use by most U.S. Department of Energy Defense Plant (DOE-DP) sites. More research and development is required, however, before the NAS can provide complete assay results for other than routine defense waste.

To date, 525 ORNL waste drums have been assayed, with varying degrees of success. The isotopic complexity of the ORNL waste creates a correspondingly complex assay problem.

The NAS and SGS assay data are presented and discussed. Neutron matrix effects, the destructive examination facility, and enriched uranium fuel-element assays are also discussed.

INTRODUCTION

Contact-handled transuranic (CH-TRU) waste, packaged in stainless-steel and mild-steel drums, has been retrievably stored in concrete bunkers at ORNL since 1970. Contact-handled TRU wastes are those as defined in DOE Order 5820.2. In addition, ^{233}U and ^{226}Ra contaminated wastes are handled by ORNL as TRU wastes. The majority of TRU waste at ORNL is generated by research laboratories; and, typically, only small quantities (a fraction of a gram to several grams) of the TRU isotopes are contained in large quantities (kilograms) of discarded material.[1]

The CH-TRU waste is stored in 208-L and 114-L drums and has a measured radiation dose rate at the container surface of less than 200 mrem/h. Consequently, the drums do not require shielding. Approximately 1% by volume of the stored TRU waste is categorized as remote-handled (RH-TRU).[1] This waste, which contains high neutron and/or beta-gamma radiation levels — up to 30,000 rem/h — is stored in concrete casks of varying thicknesses. Remote-handled TRU waste will not be addressed in this document.

A nondestructive assay system for the quantification of contact-handled transuranic elements in bulk solid waste contained in 208-L and 114-L drums has been in operation at the ORNL since April 1982. The major objectives of the ORNL-Los Alamos National Laboratory (LANL) cooperative program are the following:

1. demonstrate, evaluate, and field test the nondestructive neutron assay technique developed by the LANL Advanced Nuclear Technology Group;
2. provide a training facility for those parties interested in the new technology;
3. reduce the volume of CH-TRU waste retrievably stored on site by reclassification of that material which is improperly classified; and
4. identify the radionuclide content of ORNL CH-TRU waste.

Each of these objectives will be discussed in more detail in the body of this report.

The ORNL CH-TRU waste is complex in isotopic composition, with more than 20 alpha-emitting isotopes having been identified. This isotopic complexity creates a correspondingly complex assay problem. It must be recognized that for a percentage of waste drums it will not be possible to quantify all TRU isotopes present. A contributing factor to the assay problem is the large population of drums possessing a high passive-neutron source strength. Thirty-nine percent of the drums assayed to date contain a net passive neutron source strength greater than

1.0×10^4 n/s. Passive neutron data analysis becomes more difficult under these conditions. In most cases, however, the neutron assay system provides an upper-limit estimate of the total TRU activity based on the passive and active neutron measurements.

ORNL GAMMA-RAY AND NEUTRON ASSAY INSTRUMENTATION

A two-tier system, which consists of a Segmented Gamma Scanner (SGS) and a Neutron Assay System (NAS), has been employed at the TRU Waste Drum Assay Facility (TWDAF) for the examination of CH-TRU waste generated at ORNL. A detailed description of both instruments can be found in the literature.[2,3,4]

The SGS provides the facility operators with a list of the gamma-emitting isotopes, present in sufficient quantities, contained in the waste drum. The NAS provides the total fissile mass, expressed in milligram ^{239}Pu equivalent, contained in the drum. Estimates of the upper-limit quantities of the spontaneous fission emitters and (α,n) nuclide contributions to the total TRU content of the drum are also supplied by the NAS assay. Thus, the data obtained from both the SGS and the NAS allow the facility operators to determine the upper limit TRU content of a waste drum and the identities of the gamma-emitting isotopes present. This information is then used to classify a drum as low-level waste (LLW) or TRU waste. However, there are complications involved, and these are discussed later in this report.

SEGMENTED GAMMA SCANNER

The SGS (see Figs. 1 and 2) identifies gamma-ray-emitting isotopes which are present in sufficient quantities in the waste. It has been demonstrated[5] that a detection limit better than 100 nCi/g of waste exists for the transuranic isotopes ^{237}Np/^{233}Pa, ^{239}Pu, ^{241}Am, and ^{243}Am/^{239}Np, for the long-lived fission products ^{125}Sb, ^{134,137}Cs, ^{154}Eu and ^{60}Co.

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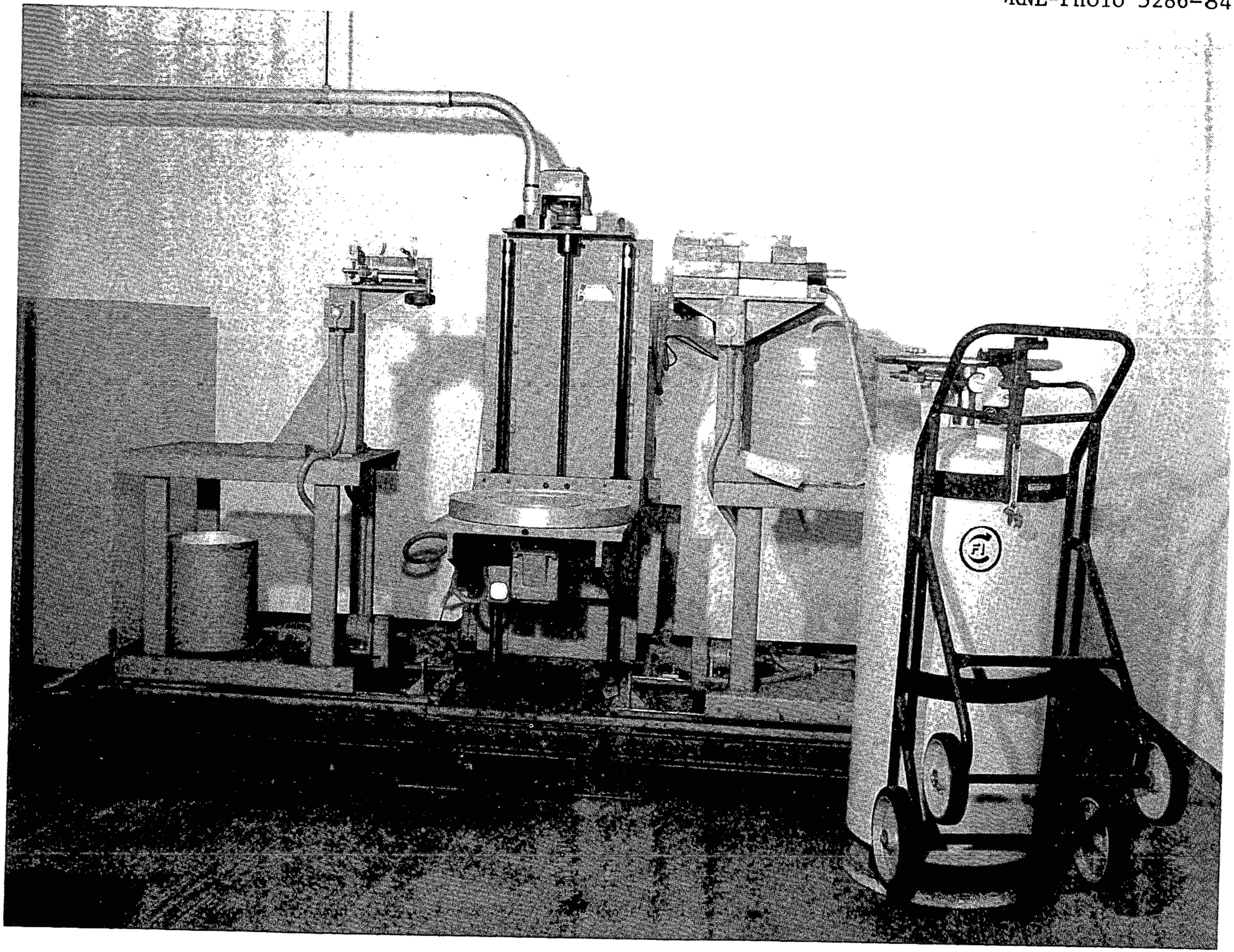

Fig. 1. Segmented Gamma Scanner

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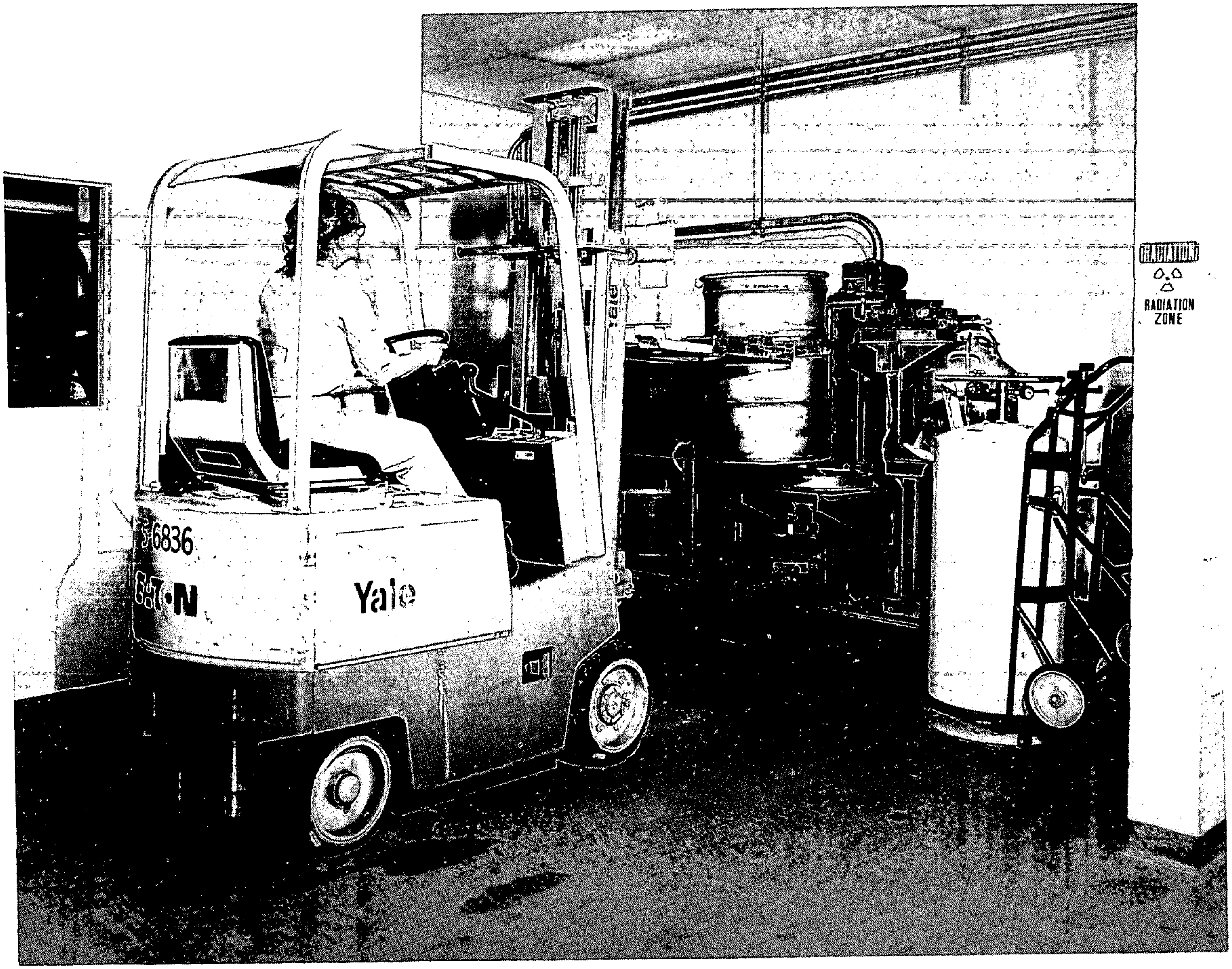

Fig. 2. Technician Positioning Waste Drum onto SGS Rotating Table

Before the cooperative program with LANL began, ORNL had purchased a Canberra Segmented Gamma Scanner.* However, the system was limited to assaying only two isotopes, ^{235}U and ^{239}Pu. Under the cooperative program, the capability of this instrument has been greatly expanded.

For simplicity in data-handling and computer software compatibility, it was decided to process the gamma-ray data through the LeCroy 3500 multichannel analyzer system, thereby resulting in a single data-acquisition and analysis system for both the neutron and gamma-ray assay measurements. All existing electronics for the SGS, except for the scan table control circuitry, have been replaced with CAMAC-compatible modules interfaced to the LeCroy 3500. The Canberra hardware for mechanically moving the waste drum and the solid-state detector and its associated electronics remain intact (Figs. 3 and 4).

A second LeCroy 3500 analyzer has recently been added to the assay facility (see Figs. 3 and 4). It replaces the sequential neutron and gamma-ray data acquisition and analyses procedures used previously with a simultaneous or concurrent capability. Also, a new computer peak-search program has been introduced which identifies photon energy peaks and correlates each with its parent radionuclide.

Each drum is scanned in three segments with data-collection time, as well as live-time determination, for each segment controlled by the computer software. See Appendix A for the SGS system software printout. If the dead-time is greater than 20%, a message informing the operator is printed on the line printer. In these cases, the drum is transferred from the SGS rotating table and suspended by a forklift truck approximately 1 m from the detector. Each segment's spectrum, along with a summation spectrum, is stored on floppy magnetic disks.

The SGS is also capable of quantitative scanning in the 100-nCi/g range using a $^{152}Eu/^{154}Eu$ transmission source.[5] Work in this area will continue in the next fiscal year.

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*Canberra Industries, Inc., 45 Gracey Avenue, Meriden, CT 06450

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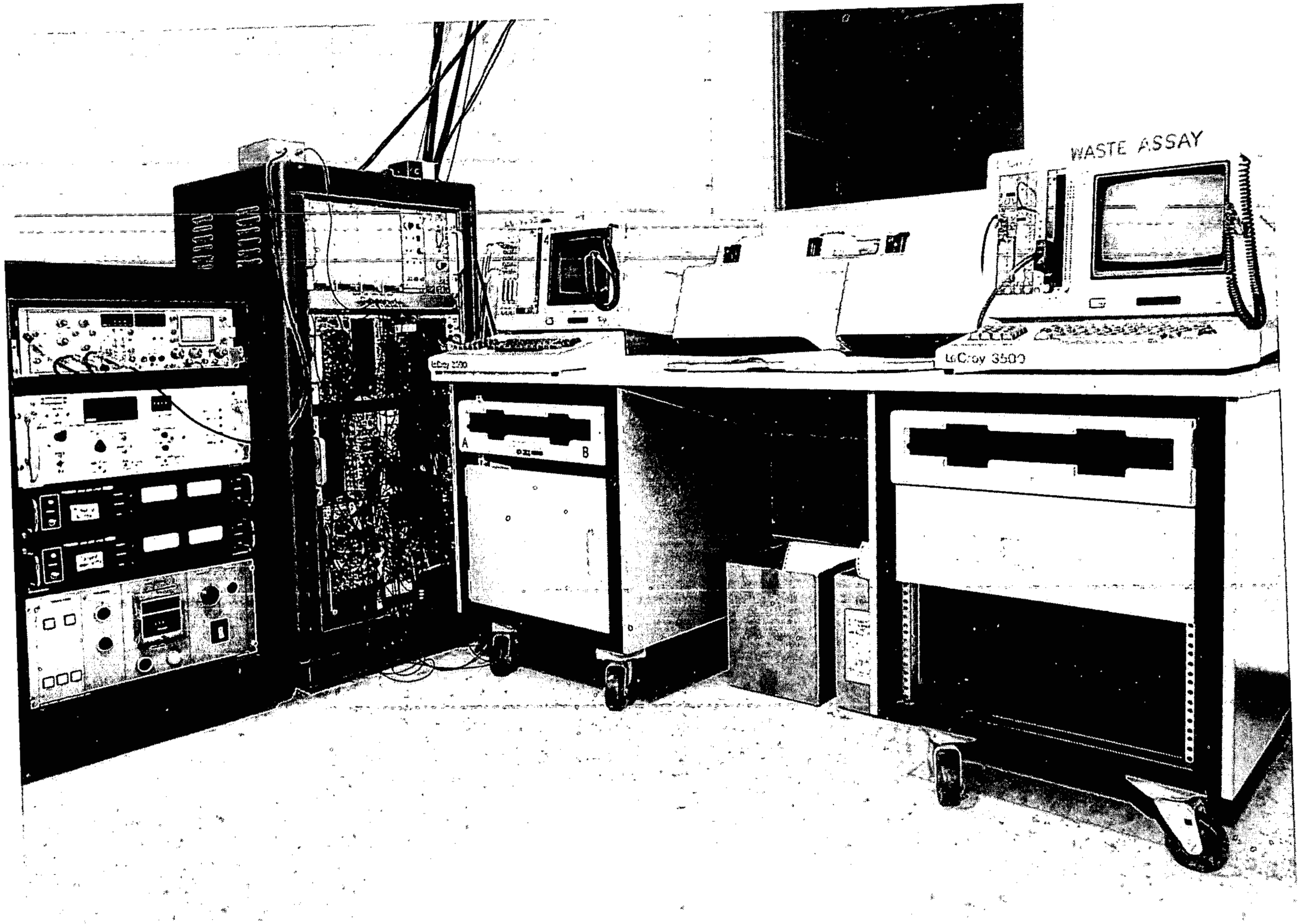

Fig. 3. TRU Waste Drum Assay Facility Control Room

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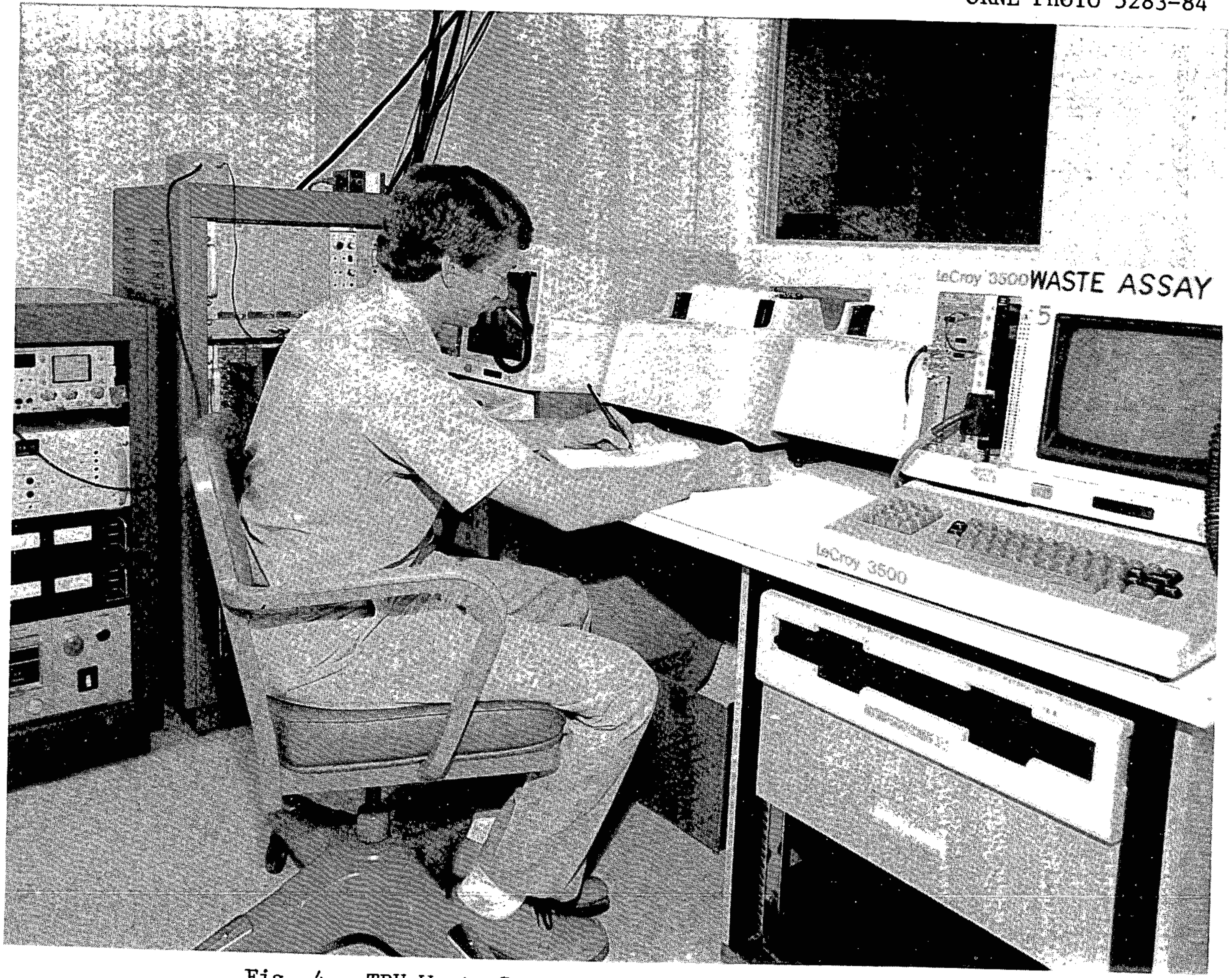

Fig. 4. TRU Waste Drum Assay Facility Data Recording

NEUTRON ASSAY SYSTEM

The NAS (see Figs. 5-8), which was developed by the LANL Advanced Nuclear Technology Group, is an active and passive 4π neutron-detection system. The active assay [differential dieaway technique (DDT)][2] refers to interrogating the waste container with an external source of neutrons, such as a portable neutron generator[6] or an electron linear accelerator (LINAC). The fast neutrons (14 MeV) produced by the neutron generator are then moderated by the graphite walls of the assay chamber. The resulting interrogating thermal neutrons induce fission reactions in the fissile material contained in the waste. The induced radioactivity (prompt and delayed fission neutrons) is proportional to the fissile TRU content (e.g., ^{235}U and ^{239}Pu). The passive assay refers to monitoring the radioactivity already occurring naturally in the sample [e.g., spontaneous fission neutrons from ^{252}Cf atoms, (α,n) reaction neutrons, etc.]. The neutron-detection packages consist of moderated (cadmium-shielded) and bare ^{3}He proportional counters embedded in the assay chamber walls.

Three neutron time-histories [counts per channel vs detector dwell-time (10 μs/ch)] are acquired for each drum during the NAS active scan. A total of 1023 channels, having a total dwell-time of 10.23 ms, are scanned. The three time-histories are listed below.

1. Shielded detectors total count from time interval 0.71 ms to 4.70 ms and 5.71 ms to 9.70 ms (background).
2. Thermal-neutron flux monitor count from time interval 0.71 ms to 4.70 ms.
3. 14-MeV monitor count (outside the assay chamber opposite the neutron generator) from time interval 0.10 ms to 0.20 ms.

The ratio of the net shielded totals (background corrected) to the flux monitor totals taken in the same time interval is proportional to

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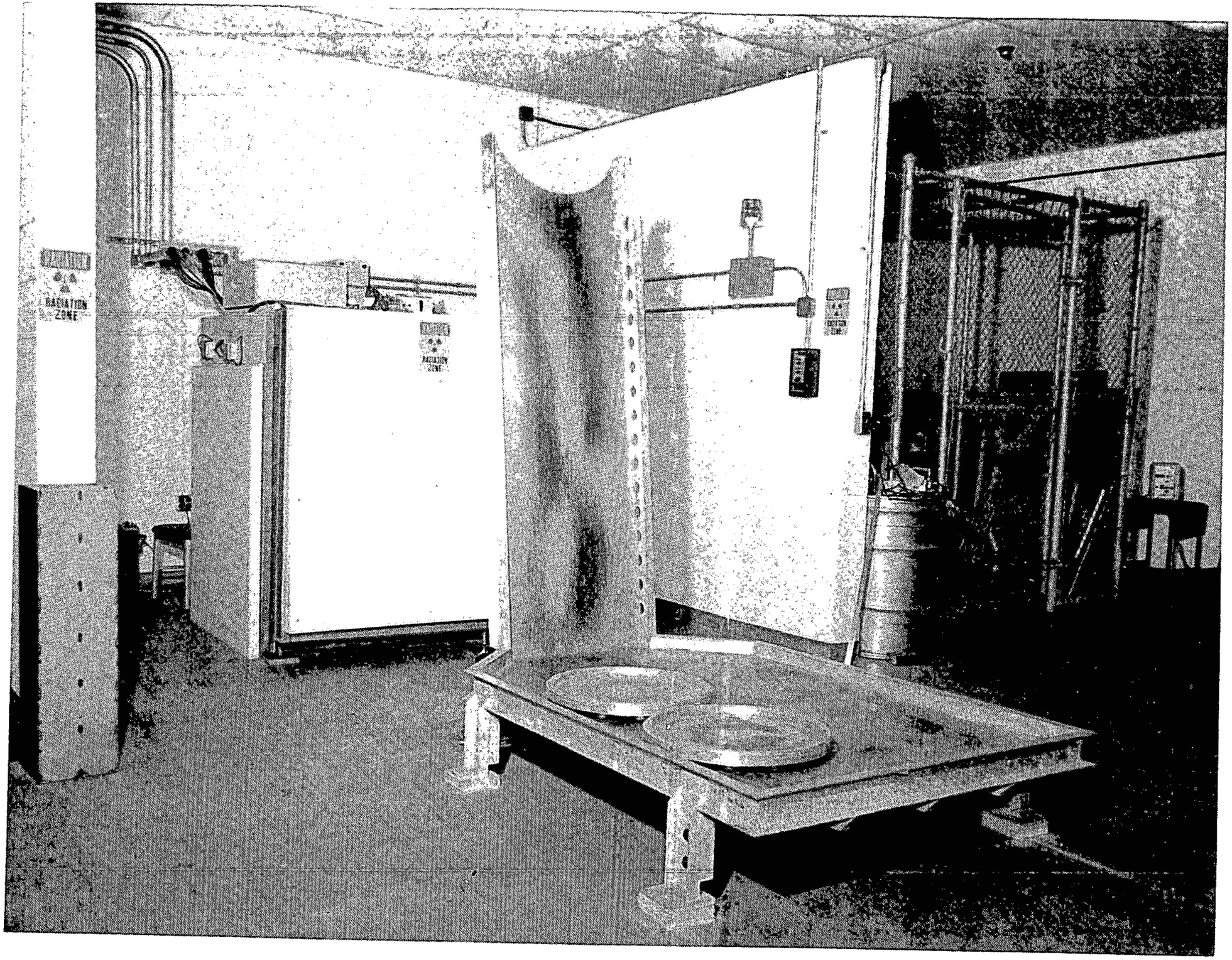

Fig. 5. Neutron Assay System

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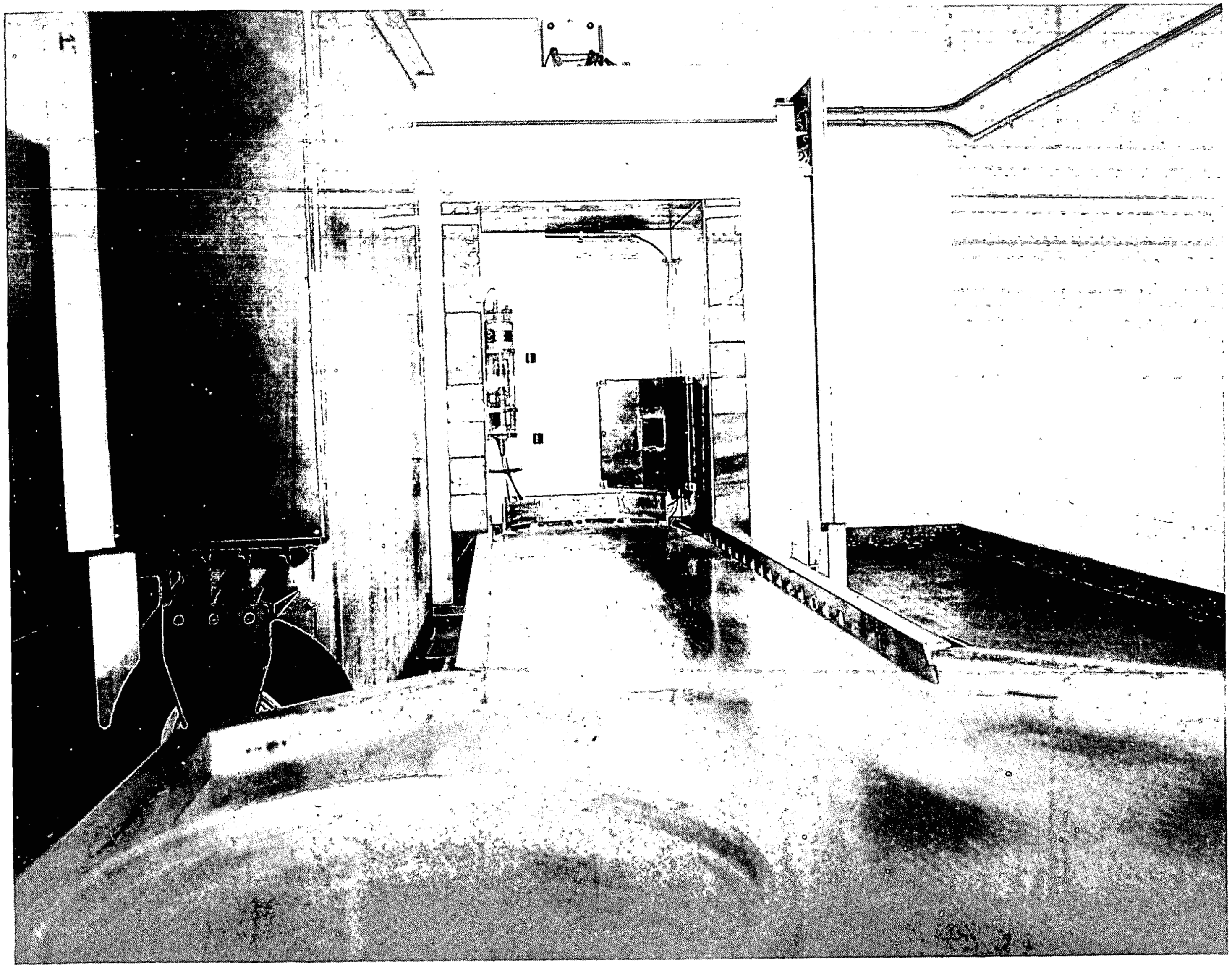

Fig. 6. Neutron Assay System Assay Chamber

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Fig. 7. Operator Transferring Waste Drum onto NAS Loading Platform

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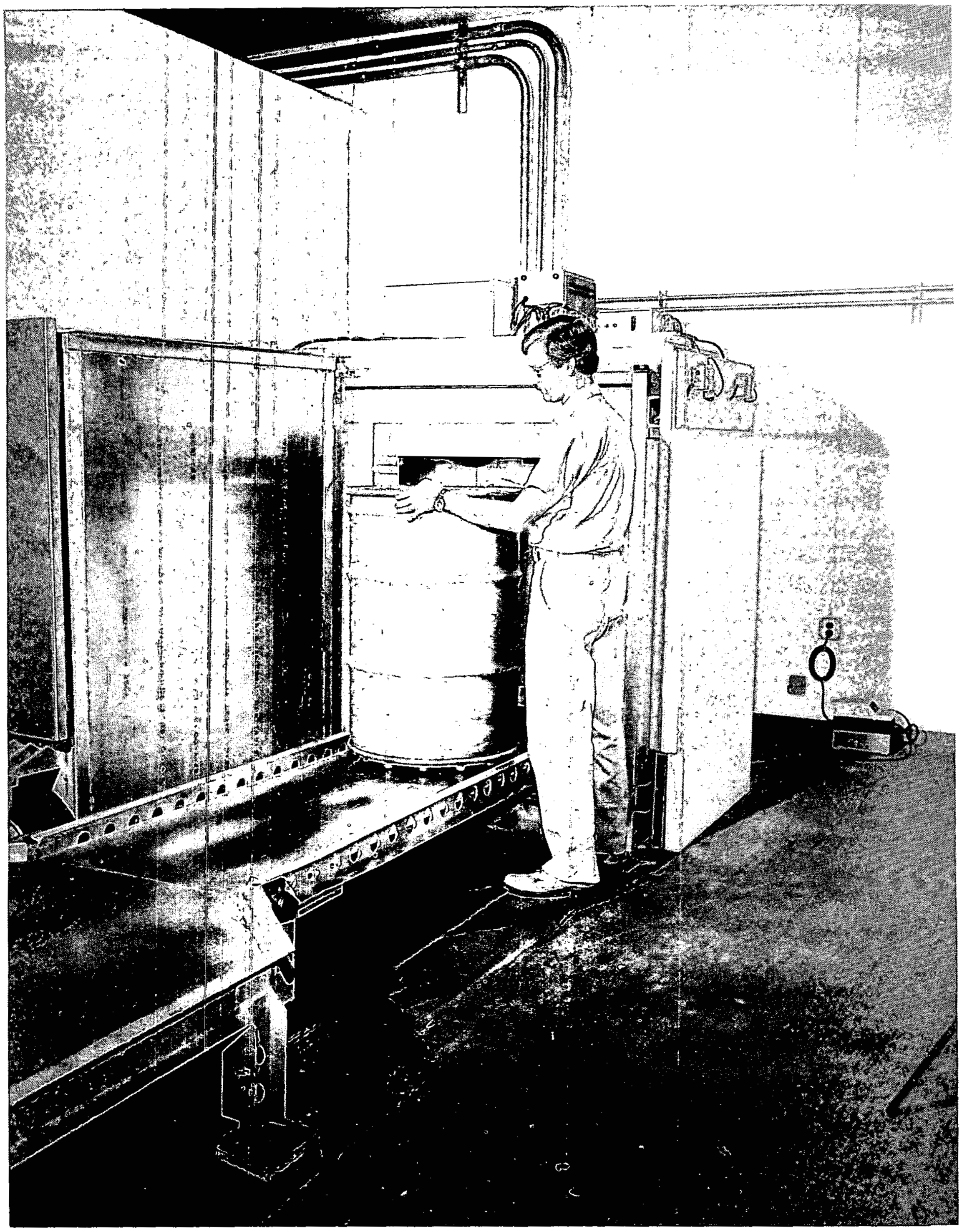

Fig. 8. Operator Positioning Waste Drum in NAS Chamber

the fissile content of the waste container. The 14-MeV monitor count rate is an indication of the neutron generator performance. The neutron generator output varies less than 10%.

An exponential least-squares fit to the thermal-neutron flux monitor time-history is performed on the interval 1.5 ms to 3.0 ms. Two variables are obtained from the fit to the data: A_o and $T_{1/2}$. A_o is the neutron amplitude normalized to time = 0 and is related to the matrix-moderating characteristics. $T_{1/2}$ is the thermal-neutron flux half-life obtained from the slope of the fitted line and is related to the matrix absorption characteristics. These two variables are then used to determine the degree of matrix correction, if any is required, for a waste drum.[3] Approximately 30% of the ORNL waste drums assayed require a matrix correction.

Since starting the operation of the NAS in April 1982, a considerable number of modifications to the instrument have occurred. These modifications are listed below.

1. Additional stress-relieved, high-density, polyethylene shielding has been affixed to the assay chamber walls and ceiling (3 in. to each of four walls and 5 in. to the ceiling). This was done in response to the higher-than-expected neutron background encountered in the facility. The background due to effects other than cosmic radiation was reduced by a factor of four.

2. The detector packages' linear amplifiers were transferred from the control room (approximately 50 ft from the NAS assay chamber) to a position atop the assay chamber. The power cables had increased the electronic noise level in the preamplifier signal cables, which had been placed in an adjacent conduit. After the transfer, the electronic noise was reduced to a minimum.

3. A second-generation neutron-generator control chassis was installed. The major advantage of the second unit was that the pulse-forming network (PFN) circuitry and power supply have been removed from the control chassis and placed adjacent to the assay chamber. This

eliminated an intense magnetic field in the control chassis, which had caused intermittent problems with the reservoir control circuitry and feedback loop. Another advantage of the second unit was that it allowed for more precise control of the reservoir circuitry voltage.

4. Three inches of stress-relieved, high-density, polyethylene shielding was added to the 1-ft-thick concrete walls surrounding the drum storage areas in the TWDAF (see Fig. 9). Neutron background levels were reduced to a minimum in the facility, consistent with operator safety and the ALARA (as low as reasonably achievable) principle.
5. A program of updating NAS computer software continued throughout the reporting period. The latest NAS system software version is called SNEUT (see Appendix B). This program was written by LANL personnel and was used originally at the Rocky Flats Plant for the neutron interrogation of waste crates (4 ft × 4 ft × 7 ft). The program contains subroutines which are extraneous to ORNL operations.
6. A magnetic-streamer tape-drive unit has been added to the data acquisition and analysis subsystem. It provides a more permanent archival storage medium than do the magnetic floppy disks.

ASSAY RESULTS

As of August 1984, 525 ORNL waste drums have been analyzed for their TRU content by the SGS and NAS. Of those drums assayed, 383 have been categorized TRU (i.e., >100 nCi/g TRU concentration) and returned to retrievable storage in anticipation of further characterization and eventual shipment to the Waste Isolation Pilot Plant (WIPP). Twenty-nine drums have been categorized low-level waste (LLW), suitable for shallow-land burial at ORNL. The LLW category represents 5.5% of the drum population. This percentage could change since the drums assayed were not selected as representative of the general drum population. Presently, approximately 120 waste drums are assayed each month at the TWDAF.

ORNL-PHOTO 5288-84

Fig. 9. Technician Transferring Waste Drums to the Shielded Drum Storage Area

Table 1 presents the basic active and passive neutron data set obtained from the assayed waste drums. Column 1 lists the drum identification number (the number assigned by the burial-ground operators). The second column lists the results, before matrix corrections are applied, as the weight of ^{239}Pu in milligrams equivalent to the fissile mass in the drum. The third column lists the net total passive neutron source strength in neutrons per second. The fourth and fifth columns list the results of the least-squares fit to the flux monitor time-history data. A_o is the normalized neutron amplitude, and $T_{1/2}$ is the thermal-neutron lifetime. The sixth column lists the ratios of the passive neutron-shielded total count rates to the system total count rates (an indication of the amount of moderator contained in the waste drum influencing the passive neutron source signal). Column 7 lists the matrix-corrected fissile mass in units of milligrams of ^{239}Pu content.

The total TRU content of each assayed drum has not been determined in most cases. However, if the net passive neutron source strength is sufficiently low (20 neutrons/s or lower) and if the fissile inventory is less than 100 mg, these data suffice to qualify the drum for the LLW category. Thirty-six waste drums fall into this category. However, if the gamma-ray assay data indicate the presence of ^{241}Am, the drum is removed from the LLW category and reclassified as unknown.

There exists a sizable drum population wherein resides a fissile inventory of less than 100 mg, but each drum contains a large passive neutron source (>20 n/s, $<10^5$ n/s). One hundred two waste drums are in this category. A more refined and detailed interpretation of the complicated passive neutron data is required before the quantities of spontaneous fission emitters and those contributions from (α,n) reactions due to such isotopes as ^{252}Cf, ^{240}Pu, and ^{241}Am can be assessed. The task is further complicated by the number of isotopes typically present in each waste drum. Los Alamos National Laboratory and ORNL will begin a program in early FY 1985 to unravel the complicated passive neutron data and provide the total TRU content of each ORNL CH-TRU waste drum.

Table 1. ORNL waste drum active and passive neutron data

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_o, cts/10 μs	Thermal neutron flux monitor, $T_{1/2}$, μs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
2291	<10	1.31×10^6	3355	374	0.287	1.87	<10
2357	<10	6.73×10^5	4015	398	0.251	2.66	<10
2098	<10	3.03×10^6	4488	484	0.260	1.00	<10
2080	<10	1.71×10^5	4748	540	0.236	1.00	<10
2327	16	9.08×10^4	2913	441	0.278	1.16	19
2156	8,853	3.09×10^3	4218	461	0.246	1.00	8,853
2157	2,700	2.51×10^3	2870	405	0.270	1.14	3,078
2158	41,433	6.35×10^4	4096	405	0.273	1.75	72,508
2290	<10	5.08×10^4	6588	466	0.217	1.00	<10
2327	<10	8.92×10^4	3417	423	0.279	1.41	<10
2402	1,089	4.29×10^2	4452	538	0.250	1.00	2,402
2289	<10	2.65×10^4	5347	471	0.219	1.00	<10
2344	6,003	3.47×10^3	2898	558	0.276	1.00	6,003
2287	2,554	1.02×10^3	4838	507	0.238	1.00	2,554
2331	13,244	1.80×10^4	2898	546	0.273	1.00	13,244
2332	1,344	5.66×10^2	4502	493	0.239	1.00	1,344
2335	24,157	1.31×10^4	2970	528	0.274	1.00	24,157
2409	1	4.25×10^4	4718	475	0.246	1.00	1
2337	12	2.37×10^4	3904	392	0.256	2.52	30
2334	7,083	2.21×10^4	6524	458	0.191	1.00	7,083

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_o, cts/10 μs	Thermal neutron flux monitor, $T_{1/2}$, μs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
2328	<10	3.11×10^3	3534	412	0.330	1.47	<10
2329	1,132	1.11×10^3	5105	506	0.439	1.00	1,132
2338	1,811	1.04×10^5	3373	386	0.261	1.89	3,423
2404	<10	4.32×10^6	5010	538	0.280	1.00	<10
2288	725	8.94×10^5	3656	439	0.271	1.53	1,109
2084	79	1.68×10^5	6512	469	0.204	1.00	79
2401	<10	5.76×10^6	2266	394	0.329	1.20	<10
2293	15	1.61×10^6	6075	479	0.215	1.00	15
2405	41	1.24×10^5	5492	497	0.224	1.00	41
2283	<10	6.90×10^5	4733	531	0.238	1.00	<10
2294	<10	1.43×10^5	6304	491	0.202	1.00	<10
2345	<10	5.54×10^6	3684	435	0.315	1.54	<10
2340	213	1.52×10^6	3766	425	0.227	1.58	337
2095	<10	9.36×10^4	5121	475	0.229	1.00	<10
2092	<10	1.13×10^6	3242	383	0.277	1.73	<10
2297	5	1.91×10^5	5862	495	0.213	1.00	5
2407	12	3.16×10^5	4514	491	0.242	1.00	12
2400	<10	6.08×10^6	4501	501	0.293	1.00	<10
2292	<10	2.50×10^6	4818	546	0.248	1.00	<10
2382	161	1.11×10^2	2769	460	0.313	1.00	161

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_o, cts/10 μs	Thermal neutron flux monitor, $T_{1/2}$, μs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
2385	479	1.87×10^3	4281	392	0.263	2.98	1,427
2159	119	82	3968	506	0.463	1.00	119
2295	962	5.94×10^2	4613	484	0.276	1.00	962
2162	1,167	3.25×10^2	4585	519	0.287	1.00	1,167
2160	259	<2	4101	531	-	1.00	259
2330	17,913	1.30×10^4	2856	524	0.274	1.00	17,913
2348	2,439	2.34×10^3	2853	549	0.285	1.00	2,439
2096	33	1.19×10^2	4464	544	0.271	1.00	33
2163	663	2.79×10^2	4679	524	0.254	1.00	663
2342	5,195	2.21×10^3	2835	565	0.277	1.00	5,195
2160	259	85	4101	531	0.275	1.00	259
2346	147	75	3450	477	0.340	1.00	147
2298	32	9.86×10^2	4621	539	0.238	1.00	32
2408	<10	5.70×10^3	4359	503	0.233	1.00	<10
2299	35	8.59×10^3	4531	541	0.237	1.00	35
2403	1,644	2.86×10^2	3834	408	0.293	1.62	2,663
2333	2,661	1.29×10^4	3619	492	0.256	1.00	2,661
2341	362	1.49×10^2	3162	387	0.320	1.63	590
2356	438	1.64×10^2	3804	381	0.317	2.40	1,051
2343	10,037	4.34×10^3	2960	546	0.278	1.00	10,037

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_o, cts/10 μs	Thermal neutron flux monitor, $T_{1/2}$, μs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
2354	19	<2	5136	446	-	2.27	43
2161	718	2.95×10^2	4768	517	0.249	1.00	718
2286	20	4.85×10^4	6197	472	0.215	1.00	20
2336	17,469	1.73×10^4	2987	480	0.283	1.00	17,469
2326	<10	3.40×10^5	3527	423	0.279	1.46	<10
2410	9	2.84×10^5	5805	494	0.213	1.00	9
2355	27	4.93×10^5	5247	432	0.242	2.32	63
2374	3,314	6.85×10^4	3532	370	0.258	2.08	6,893
2325	5	9.17×10^4	3200	430	0.271	1.30	7
2097	<10	6.60×10^4	4400	487	0.245	1.00	<10
2439	5	4.79×10^4	4758	518	0.229	1.00	5
2376	9	1.76×10^4	4586	418	0.242	1.99	18
2390	356	3.58×10^2	4301	512	0.249	1.00	356
2398	3,860	1.71×10^4	2944	599	0.268	1.00	3,860
2393	718	3.34×10^2	3821	464	0.258	1.00	718
2445	2,455	1.77×10^4	3293	377	0.263	1.79	4,394
2383	56	1.62×10^2	3086	445	0.294	1.24	69
2458	2	4.99×10^4	5461	487	0.216	1.00	2
2444	9	8.89×10^4	5806	489	0.212	1.00	9
2396	2,242	1.03×10^4	3672	416	0.267	1.54	3,453

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_0, cts/10 µs	Thermal neutron flux monitor, $T_{1/2}$, µs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
2507	5,505	1.35×10^3	4069	454	0.251	1.00	5,505
2453	<10	8.61×10^5	4659	521	0.239	1.00	<10
2502	1,333	4.68×10^5	2812	392	0.262	1.21	1,613
2397	1,770	5.64×10^6	2873	399	0.307	1.29	2,283
2459	<10	5.97×10^6	2201	375	0.331	1.20	<10
2456	11	8.93×10^5	5688	489	0.217	1.00	11
2454	<10	5.33×10^6	3911	489	0.285	1.00	<10
2451	1	1.24×10^6	5888	479	0.215	1.00	1
2389	585	1.72×10^5	4296	476	0.239	1.00	585
2449	<10	6.02×10^6	2413	379	0.319	1.20	<10
2452	19	1.94×10^5	4674	517	0.238	1.00	19
2442	390	2.56×10^5	2839	397	0.262	1.25	488
2437	<10	6.15×10^6	4423	509	0.285	1.00	<10
2441	1,212	1.33×10^6	4763	514	0.234	1.00	1,212
2447	<10	3.20×10^5	4189	539	0.244	1.00	<10
2443	<10	4.46×10^5	4835	529	0.245	1.00	<10
2448	<10	1.92×10^5	5890	487	0.214	1.00	<10
2435	421	2.64×10^6	2880	402	0.279	1.14	480
2365	7	2.41×10^4	4166	403	0.242	1.78	12
2380	3,513	6.59×10^2	4083	415	0.277	1.74	6,113

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_0, cts/10 μs	Thermal neutron flux monitor, $T_{1/2}$, μs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
2399	5,155	4.52×10^3	2877	551	0.271	1.00	5,155
2099	23	9.48×10^4	5317	468	0.207	1.00	23
2367	<10	9.23×10^4	3435	395	0.249	1.96	<10
2413	116	2.97×10^6	2777	467	0.286	1.00	116
2412	1,301	1.61×10^5	2764	408	0.264	1.10	1,431
2202	121	7.20×10^4	5116	477	0.218	1.00	121
2220	<10	1.36×10^6	4827	495	0.215	1.00	<10
2436	<10	6.15×10^6	2102	443	0.322	1.10	<10
2415	<10	3.26×10^6	2019	368	0.313	1.20	<10
2364	37	3.98×10^5	3813	432	0.241	1.61	60
2363	47	2.19×10^5	3265	413	0.263	1.33	63
2360	1,932	3.28×10^3	3553	416	0.258	1.48	2,859
2361	131	6.84×10^1	3118	464	0.285	1.00	131
2421	0	<2	2569	460	-	1.00	<10
2428	<10	<2	3014	478	-	1.00	<10
2424	<10	<2	3086	470	-	1.00	<10
2422	<10	3.12×10^2	3230	517	0.242	1.00	<10
2433	<10	<2	3415	491	-	1.00	<10
2373	239	1.79×10^4	3127	437	0.244	1.26	301
2358	1,067	9.83×10^3	3289	424	0.261	1.34	1,430

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_o, cts/10 μs	Thermal neutron flux monitor, $T_{1/2}$, μs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
2430	<10	<2	3062	468	-	1.00	<10
2420	<10	1.56×10^1	3587	426	0.327	1.49	<10
2418	4	7.81×10^3	3771	452	0.229	1.00	4
2378	239	2.07×10^2	3547	434	0.276	1.47	351
2379	3	<2	3137	411	-	1.27	4
2386	1,170	2.96×10^2	4266	446	0.264	1.83	2,141
2387	1,355	2.62×10^2	4344	471	0.264	1.00	1,355
2375	165	1.15×10^3	3362	440	0.261	1.38	228
2377	9	1.44×10^4	4023	378	0.249	2.67	24
2501	2,586	2.73×10^3	2743	573	0.273	1.00	2,586
2388	289	2.08×10^3	4192	414	0.265	1.80	520
2381	2,891	2.37×10^3	3829	413	0.249	1.61	4,655
0904	4,431	2.28×10^2	3075	483	0.256	1.00	4,431
0974-3	3,075	7.63×10^2	2946	480	0.273	1.00	3,075
1076	20,223	1.03×10^3	3822	435	0.233	1.61	32,559
1139	2,974	1.88×10^2	2731	495	0.263	1.00	2,974
1140	24,043	1.27×10^3	3611	486	0.246	1.00	24,043
1115	1,183	6.14×10^1	3045	477	0.293	1.00	1,183
0976-4	3,199	1.01×10^2	2957	491	0.282	1.00	3,199
1142	6,715	3.45×10^2	2708	510	0.260	1.00	6,715

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_o, cts/10 μs	Thermal neutron flux monitor, $T_{1/2}$, μs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
0974-1	1,825	6.73×10^1	3209	484	0.261	1.00	1,825
2368	4	3.78×10^3	3210	421	0.240	1.31	5
2431	<10	8.94×10^1	2923	534	0.297	1.00	<10
2432	<10	<2	3351	451	–	1.00	<10
2362	1,783	4.63×10^3	3154	499	0.244	1.00	1,783
2419	0	6	3754	407	0.555	1.58	<10
2429	<10	6	3184	473	0.466	1.00	<10
2359	<10	2.73×10^4	2871	396	0.253	1.29	<10
2427	<10	6.35×10^2	3506	489	0.237	1.00	<10
2425	1	<2	3226	477	–	1.00	1
2423	0	1.04×10^2	3078	460	0.275	1.00	<10
2426	0	4.34×10^1	3756	444	0.955	1.58	<10
2411	27	1.97×10^4	4438	482	0.216	1.00	27
2165	553	2.28×10^2	3864	509	0.250	1.00	553
2164	337	1.17×10^2	3329	540	0.262	1.00	337
2169	315	8.94×10^1	3525	473	0.273	1.00	315
2167	1,381	5.36×10^2	3746	516	0.245	1.00	1,381
2166	3,645	1.96×10^3	3429	521	0.240	1.00	3,645
2168	1,417	5.37×10^2	3474	522	0.251	1.00	1,417
2440	41	2.26×10^5	5751	473	0.205	1.00	41

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_o, cts/10 μs	Thermal neutron flux monitor, $T_{1/2}$, μs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
2416	69	3.27×10^4	5946	458	0.206	1.00	69
2207	170	1.75×10^5	2877	381	0.275	1.29	219
2214	13	1.69×10^6	4238	535	0.243	1.00	13
1080	1,234	1.35×10^2	2605	518	0.176	1.00	1,234
1082	2,149	3.61×10^2	3574	456	0.213	1.00	2,149
1267	645	3.30×10^2	4024	364	0.231	2.67	1,722
0798	19	2.09×10^4	3029	411	0.245	1.21	23
1268	3,131	1.55×10^3	3211	408	0.248	1.31	4,102
0977-2	8,820	4.67×10^2	4002	475	0.235	1.00	8,820
0976-1	2,188	5.13×10^1	3458	475	0.262	1.00	2,188
0977-3	731	1.46×10^1	2760	495	0.271	1.00	731
1111	3,942	1.00×10^2	3361	460	0.250	1.00	3,942
1061	2,753	8.08×10^1	3102	443	0.259	1.25	3,441
1064	16,375	7.41×10^2	4238	488	0.227	1.00	16,375
1065	20,041	1.02×10^3	3998	465	0.227	1.00	20,041
1063	3,150	1.09×10^2	3080	475	0.261	1.00	3,150
1157	1,160	6.44×10^1	3167	440	0.274	1.28	1,485
1164	1,095	2.26×10^1	3232	469	0.274	1.00	1,095
0998	2,095	5.32×10^1	3075	467	0.287	1.00	2,095
1056	1,369	3.66×10^1	2879	482	0.239	1.00	1,369

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_o, cts/10 µs	Thermal neutron flux monitor, $T_{1/2}$, µs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
1059	1,457	1.49×10^2	2598	488	0.244	1.00	1,457
1051	12,025	9.67×10^2	3355	388	0.250	1.87	22,487
0961-1	479	1.16×10^1	2592	502	0.454	1.00	479
0969-2	11,647	3.36×10^2	2356	537	0.268	1.00	11,647
1159	875	2.72×10^1	3190	470	0.368	1.00	875
1162	778	1.83×10^1	3073	470	0.314	1.00	778
0970-1	451	1.00×10^1	2988	483	0.228	1.00	451
0959-4	740	8.46×10^1	3344	455	0.377	1.00	740
0977-4	895	1.08×10^1	3074	479	0.404	1.00	895
1074	22,801	1.22×10^3	2887	526	0.243	1.00	22,801
0977-1	6,725	6.53×10^2	2634	411	0.267	1.10	7,398
0989	1,789	3.65×10^1	3260	484	0.275	1.00	1,789
1146	5,757	1.96×10^2	3289	483	0.255	1.00	5,757
1119	1,770	1.49×10^3	2781	471	0.241	1.00	1,770
0667	10,630	2.10×10^4	3479	390	0.244	2.01	21,366
0736	448	2.93×10^5	2891	452	0.257	1.00	448
0815	123	1.24×10^6	3135	389	0.268	1.60	197
0703	3	7.21×10^5	3201	403	0.273	1.30	4
0734	<10	4.73×10^6	3106	415	0.304	1.25	<10
0869	<10	1.13×10^5	3189	430	0.256	1.29	<10

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_o, cts/10 µs	Thermal neutron flux monitor, $T_{1/2}$, µs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
0733	39	3.46×10^4	3082	399	0.266	1.54	60
0403	349	2.68×10^4	2955	478	0.240	1.00	349
0870	<10	7.26×10^5	3186	433	0.262	1.29	<10
0704	<10	1.52×10^7	917	727	0.508	1.00	<10
0723	<10	1.63×10^6	3257	413	0.276	1.33	<10
0813	63	5.93×10^5	3781	401	0.253	1.59	100
1280	21	1.56×10^4	2907	425	0.260	1.15	24
1235	10	2.58×10^4	3224	412	0.248	1.31	13
0888	<10	<2	6002	385	-	5.04	<10
0887	<10	<2	4276	481	-	1.00	<10
0580	48	5	3115	386	<0.001	1.58	76
0587	22	1.20×10^1	3520	356	0.536	2.06	45
0699	4	9.03×10^2	3766	416	0.260	1.58	6
0586	385	3.74×10^2	4094	343	0.238	4.44	1,709
0895	10,195	4.44×10^2	3866	510	0.239	1.00	10,195
0577	71	1.20×10^2	4547	346	0.260	5.12	364
0573	121	1.33×10^2	3636	408	0.241	1.52	184
0898	14	<2	3801	482	-	1.00	14
0583	55	1.39×10^1	3769	357	0.432	3.95	217
0740	54	5.56×10^2	2774	365	0.257	1.20	65

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_o, cts/10 µs	Thermal neutron flux monitor, $T_{1/2}$, µs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
0741	255	4.56×10^3	3305	347	0.237	3.26	831
0578	28	2.22×10^1	3411	340	0.433	3.42	96
0588	11	2.54×10^1	4270	358	0.433	4.71	52
0735	173	9.43×10^3	2954	383	0.244	1.38	239
0688	87	7.64×10^3	3162	355	0.238	3.04	264
1243	1	1.90×10^4	4290	436	0.227	1.85	2
2525	352	4.38×10^3	3089	417	0.260	1.53	539
2526	78	1.90×10^2	3715	478	0.247	1.00	78
2522	1,226	5.48×10^2	3083	421	0.274	1.24	1,520
2527	149	1.20×10^4	3795	405	0.255	1.60	238
1238	<10	<2	4222	438	–	1.81	<10
0385	<10	1.92×10^4	3367	456	0.250	1.00	<10
0383	69	4.63×10^2	3434	470	0.251	1.00	69
2500	550	2.21×10^4	2617	550	0.275	1.00	550
1244	273	9.28×10^2	3401	468	0.253	1.00	273
2524	6,797	5.62×10^1	3654	492	0.246	1.00	6,797
2391	26	5.48×10^3	2659	508	0.270	1.00	26
2469	17	3.48×10^5	4088	525	0.231	1.00	17
0792	619	1.90×10^2	3913	461	0.180	1.00	619
0929	386	1.44×10^2	3935	482	0.177	1.00	386

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_0, cts/10 µs	Thermal neutron flux monitor, $T_{1/2}$, µs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
0930	751	2.61×10^1	3683	491	0.212	1.00	751
0926	8,069	3.83×10^2	4690	503	0.232	1.00	8,069
0917	5,105	1.16×10^3	3956	418	0.263	1.68	8,576
1019	1,554	4.59×10^1	3736	464	0.227	1.00	1,554
1020	927	1.69×10^1	3186	497	0.177	1.00	927
0984	1,638	3.04×10^1	3833	463	0.251	1.00	1,638
1024	1,189	2.46×10^1	3697	476	0.255	1.00	1,189
1036	2,560	6.08×10^1	3625	476	0.239	1.00	2,560
1403	2,400	1.25×10^2	4094	427	0.251	1.75	4,200
1015	12,813	5.41×10^2	4194	497	0.245	1.00	12,813
1014	13,272	5.27×10^2	3818	526	0.250	1.00	13,272
0997	1,345	2.89×10^1	3518	488	0.228	1.00	1,345
1022	2,013	5.11×10^1	3574	480	0.267	1.00	2,013
1021	1,296	3.81×10^1	3499	503	0.228	1.00	1,296
1011	15,681	7.40×10^2	3334	534	0.256	1.00	15,681
0990	1,727	2.96×10^1	4029	474	0.284	1.00	1,727
1043-1	55	1.30×10^4	2898	512	0.253	1.00	55
0232	16	8.57×10^3	2694	568	0.242	1.00	16
0175	6,227	3.32×10^3	2362	589	0.272	1.00	6,227
1043-2	<10	2.93×10^3	2778	534	0.246	1.00	<10

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_o, cts/10 µs	Thermal neutron flux monitor, $T_{1/2}$, µs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
2533	627	8.10×10^4	3183	474	0.246	1.00	627
2532	425	6.38×10^2	3832	457	0.226	1.00	425
2529	1,683	4.36×10^2	3020	457	0.252	1.00	1,683
2530	950	1.61×10^3	3139	432	0.257	1.27	1,207
2531	719	1.69×10^2	3992	431	0.241	1.70	1,222
1008	401	7.90×10^6	2138	593	0.352	1.00	401
0620	29	2.00×10^5	2650	497	0.265	1.00	29
1070-2	<10	5.05×10^6	2205	546	0.308	1.00	<10
1070-1	163	7.65×10^6	2119	578	0.336	1.00	163
0448	8,661	6.62×10^4	2784	540	0.239	1.00	8,661
0022-3	19	1.74×10^6	2949	498	0.267	1.00	19
2479	62	1.10×10^5	5252	458	0.205	1.00	62
0950-1	10	6.34×10^5	2680	508	0.275	1.00	10
0021	1,302	2.29×10^4	3342	422	0.236	1.37	1,784
0386	37	7.45×10^4	3394	383	0.267	1.91	71
0948-1	47	1.60×10^5	3186	440	0.257	1.29	61
0016-2	11,955	3.22×10^4	2609	549	0.261	1.00	11,955
0265	1,790	3.33×10^4	2763	381	0.229	1.20	2,148
2476	100	1.27×10^5	4541	436	0.238	3.37	337
0592	<10	1.18×10^7	1881	564	0.411	1.00	<10

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_o, cts/10 μs	Thermal neutron flux monitor, $T_{1/2}$, μs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
2478	8	7.68×10^6	3666	527	0.219	1.00	8
0209	1,987	1.24×10^3	2827	572	0.257	1.00	1,987
0463	5	3.25×10^2	2848	491	0.253	1.00	5
0024-2	693	9.60×10^3	2837	518	0.255	1.00	693
0979-1	<10	2.29×10^2	2649	515	0.272	1.00	<10
0979-2	<10	5.93×10^1	3192	509	0.294	1.00	<10
0948-3	1	<2	2748	500	-	1.00	1
0027-1	5,983	5.86×10^3	2996	524	0.249	1.00	5,983
0027-3	2,727	3.17×10^3	2837	580	0.256	1.00	2,727
0976-3	23,841	8.38×10^2	3219	550	0.239	1.00	23,841
0014-3	<10	2.85×10^1	3227	495	0.268	1.00	<10
0242	552	6.87×10^2	2794	366	0.255	1.20	662
0264	215	4.02×10^3	2869	386	0.245	1.28	275
0944	22,176	1.10×10^3	3216	507	0.247	1.00	22,176
0287	<10	1.48×10^1	3191	482	0.317	1.00	<10
2477	7	1.03×10^4	4038	518	0.227	1.00	7
2480	393	1.08×10^4	3426	408	0.244	1.41	554
0950-2	<10	7.99×10^2	3126	530	0.236	1.00	<10
0950-4	1	6.76×10^1	5052	481	0.216	1.00	1
0243	1,133	9.25×10^2	3354	370	0.262	1.86	2,107

Table 1. *ORNL waste drum active and passive neutron data (cont.)*

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_o, cts/10 μs	Thermal neutron flux monitor, $T_{1/2}$, μs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
0261	209	2.17×10^3	2941	356	0.275	1.37	286
0397	43	7.18×10^2	3458	419	0.257	1.43	61
2471	56	2.59×10^5	4927	485	0.204	1.00	56
2464	5	5.10×10^6	3351	549	0.283	1.00	5
0429	<10	3.16×10^5	2762	504	0.248	1.00	<10
2511	947	3.73×10^2	2727	505	0.228	1.00	947
2462	<10	6.98×10^5	4036	524	0.244	1.00	<10
2465	1	5.46×10^4	5389	473	0.206	1.00	1
2470	<10	1.30×10^5	4162	525	0.229	1.00	<10
2468	46	1.26×10^5	5293	476	0.205	1.00	46
2466	<10	1.33×10^5	3671	548	0.230	1.00	<10
2472	271	3.21×10^5	2590	370	0.267	1.20	325
0553	<10	1.48×10^5	3638	399	0.272	1.20	<10
2463	59	5.74×10^5	5567	467	0.212	1.00	59
2467	49	2.22×10^6	2393	374	0.280	1.20	59
2517	3,146	3.07×10^3	3529	451	0.241	1.00	3,146
1261	3,011	4.94×10^3	3148	472	0.248	1.00	3,011
2510	133	7.80×10^1	3579	416	0.280	1.49	198
0910	7	1.84×10^4	3487	401	0.258	1.44	10
0928	367	<2	2787	495	-	1.00	367

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_o, cts/10 µs	Thermal neutron flux monitor, $T_{1/2}$, µs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
0918	98	<2	2771	488	–	1.00	98
0916	767	<2	3056	493	–	1.00	767
0961-2	839	1.04×10^1	3269	462	0.481	1.00	839
0970-2	1,187	1.75×10^1	3182	461	0.328	1.00	1,187
0896	13,516	5.27×10^2	3910	499	0.243	1.00	13,516
0961-3	692	8.4	3032	483	0.753	1.00	692
0911	4,236	1.69×10^2	4278	461	0.246	1.00	4,236
0914	953	1.29×10^1	3702	456	0.439	1.00	953
1053	5,051	2.01×10^2	3354	483	0.274	1.00	5,051
1054	8,316	3.47×10^2	3627	438	0.260	1.51	12,557
0992	915	1.82×10^1	2758	493	0.311	1.00	915
0902	1,368	2.07×10^1	2661	508	0.332	1.00	1,368
0991	15,955	7.66×10^2	3678	496	0.232	1.00	15,955
0974-2	1,175	4.53×10^1	3259	454	0.272	1.00	1,175
0999	926	1.30×10^1	2610	496	0.391	1.00	926
1045	54	7.97×10^3	3639	375	0.258	2.21	119
1042	1,408	1.19×10^5	3351	430	0.257	1.38	1,943
1154	2,598	2.87×10^3	3843	456	0.240	1.00	2,598
1029	21	2.17×10^3	3361	480	0.248	1.00	21
0982	51	9.59×10^3	3269	388	0.253	1.76	90

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_o, cts/10 µs	Thermal neutron flux monitor, $T_{1/2}$, µs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
1266	2,074	1.76×10^3	4697	492	0.214	1.00	2,074
1205	<10	5.24×10^2	3656	475	0.209	1.00	<10
1151	207	5.82×10^5	3223	433	0.262	1.31	271
0903	485	1.88×10^2	2969	488	0.187	1.00	485
0913	6,916	4.53×10^2	3286	468	0.266	1.00	6,916
0892	6,861	5.78×10^2	3501	465	0.231	1.00	6,861
0912	3,703	5.03×10^2	4022	423	0.232	1.71	6,332
1068	4	1.13×10^3	3993	481	0.233	1.00	4
1083	11,962	7.93×10^2	3370	484	0.218	1.00	11,962
0974-4	999	1.87×10^1	3891	484	0.262	1.00	999
1116	722	1.09×10^1	4668	464	0.166	1.00	722
1112	2,242	5.86×10^1	4293	447	0.246	1.85	4,148
1035	506	2.13×10^1	4203	440	0.195	1.80	911
1034	1,013	2.26×10^1	3918	474	0.207	1.00	1,013
1016	15,873	6.29×10^2	4680	500	0.238	1.00	15,873
1010	16,051	8.14×10^2	3663	536	0.228	1.00	16,051
1079	4,347	1.11×10^2	5354	445	0.254	2.38	10,346
1057	18,425	1.33×10^3	3806	382	0.254	2.41	44,404
0970-3	1,252	2.09×10^1	4153	473	0.183	1.00	1,252
1138	1,520	1.01×10^2	3542	516	0.247	1.00	1,520

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_o, cts/10 µs	Thermal neutron flux monitor, $T_{1/2}$, µs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
0537	144	2.63×10^3	3145	388	0.246	1.61	232
0615	31	4.90×10^2	4143	418	0.254	1.77	55
0622	1,667	2.89×10^4	4167	361	0.240	2.84	4,734
1409	3,549	1.47×10^2	4562	518	0.232	1.00	3,549
0616	79	3.21×10^2	4225	405	0.252	1.81	143
1407	11,659	5.64×10^2	4407	502	0.238	1.00	11,659
1408	7,993	3.47×10^2	4753	491	0.237	1.00	7,993
1410	5,046	2.98×10^2	4792	486	0.228	1.00	5,046
0794	8,317	1.45×10^3	4191	391	0.255	2.87	23,870
0795	3,779	1.08×10^3	4090	405	0.256	1.75	6,613
0791	6,059	5.26×10^2	3964	505	0.242	1.00	6,059
0790	3,141	3.13×10^2	2879	534	0.220	1.00	3,141
0923	8,439	6.35×10^2	3918	456	0.233	1.00	8,439
0793	11,861	1.50×10^3	3620	486	0.224	1.00	11,861
0810	<10	1.42×10^6	3036	382	0.266	1.48	<10
1269	195	7.29×10^1	2816	460	0.337	1.00	195
1239	1	<2	4530	419	-	1.97	2
0924	403	<2	3131	488	-	1.00	403
0927	537	<2	3404	460	-	1.00	537
0925	6,105	2.64×10^2	3811	473	0.248	1.00	6,105

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_0, cts/10 µs	Thermal neutron flux monitor, $T_{1/2}$, µs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
0818	2,483	8.92×10^1	3314	477	0.281	1.00	2,483
0874	802	4.79×10^1	3772	467	0.316	1.00	802
1422	9,460	4.65×10^2	3693	507	0.240	1.00	9,460
0808	0	1.82×10^3	2750	418	0.260	1.10	<10
1423	725	1.76×10^2	2479	463	0.286	1.00	725
0758	6,603	3.67×10^2	2611	532	0.265	1.00	6,603
0757	4,529	3.16×10^2	3888	463	0.245	1.00	4,529
0875	29	6.86×10^3	3647	440	0.239	1.52	44
0411	<10	<2	3912	452	-	1.00	<10
0824	15,461	1.08×10^3	3225	518	0.233	1.00	15,461
0806	129	1.21×10^3	2945	489	0.254	1.00	129
0821	5,265	2.31×10^2	3020	538	0.260	1.00	5,265
0807	6	3.94×10^3	2538	447	0.268	1.10	7
2484	<10	3.11×10^2	4100	447	0.229	1.75	<10
2483	180	3.90×10^5	3550	382	0.251	2.10	378
2485	<10	1.20×10^7	1270	569	0.425	1.00	<10
0249	7	<2	4131	344	-	4.50	32
0948-2	3	4.75×10^1	3197	501	0.285	1.00	3
0541	151	4.07×10^3	3600	348	0.252	3.70	559
0188	1,087	3.76×10^3	2378	423	0.252	1.10	1,196

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_o, cts/10 μs	Thermal neutron flux monitor, $T_{1/2}$, μs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
2473	3	6.18×10^4	3949	540	0.227	1.00	3
0945	10,770	5.39×10^2	3262	481	0.258	1.00	10,770
0398	10	6.98×10^1	4486	432	0.280	1.94	19
0595	4	1.70×10^3	3721	403	0.252	1.56	6
0946	671	9.3	3091	432	0.464	1.25	839
0435	10,851	6.89×10^3	2970	541	0.246	1.00	10,851
0975-3	7,955	3.33×10^2	2688	581	0.255	1.00	7,955
0022-4	<10	5.28×10^2	3015	537	0.255	1.00	<10
0161	<10	<2	2668	569	-	1.00	<10
2051	7	3.6×10^5	5378	518	-	1.00	7
2057	24	1.2×10^5	6115	485	-	1.00	24
1940	11	1.2×10^5	-	-	0.246	-	11
2081	28	4.0×10^5	5599	496	0.230	1.00	28
2075	9	1.9×10^5	4665	548	0.267	1.00	9
1934	11	6.5×10^5	5500	524	0.222	1.00	11
1928	15	1.8×10^5	4825	510	0.237	1.00	15
2039	7	1.0×10^5	4775	472	0.247	1.00	7
2005	25	1.5×10^5	5994	483	0.234	1.00	25
2112	12	3.1×10^5	5220	472	0.265	1.00	12
1929	22	2.1×10^5	6636	483	0.197	1.00	22

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_o, cts/10 μs	Thermal neutron flux monitor, $T_{1/2}$, μs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
2013	5	2.3×10^5	5064	525	-	1.00	5
2001	8	1.1×10^5	4708	542	0.246	1.00	8
2010	7	1.3×10^5	5215	532	0.228	1.00	7
1926	-	2.7×10^5	-	-	0.197	-	-
2003	52	5.9×10^5	2575	354	0.286	2.50	130
1951	75	3.6×10^5	5840	490	-	1.00	75
2052	83	5.5×10^5	6389	480	-	1.00	83
1950	449	4.0×10^5	4891	534	-	1.00	449
2109	3,200	8.8×10^5	3587	402	0.276	1.49	4,768
0749	30	2.5×10^6	2910	551	-	1.00	30
2044	0	4.4×10^6	4088	535	0.267	1.00	<10
1998	0	7.6×10^6	3478	459	-	1.00	<10
2094	4	2.0×10^6	3318	377	0.270	1.81	7
1933	56	2.2×10^6	4770	526	0.271	1.00	56
2110	0	2.5×10^6	4305	476	0.200	1.00	0
1959	-	3.0×10^6	-	-	-	-	-
1936	63	2.1×10^6	3323	372	0.260	1.82	115
1938	50	5.7×10^6	2617	463	-	1.00	50
1948	71	1.4×10^6	3152	376	0.293	1.62	115
2019	41	1.3×10^6	2821	365	0.270	1.85	76

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_o, cts/10 µs	Thermal neutron flux monitor, $T_{1/2}$, µs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
1939	467	6.0×10^6	3869	545	0.133	1.00	467
1945	2	0	4518	503	-	1.00	2
1921	1	0	3593	484	-	1.00	1
1924	1	0	4119	481	-	1.00	1
1922	3	0	4337	469	-	1.00	3
1919	4	0	4403	452	-	1.00	4
1208	1	0	4678	466	-	1.00	1
1788	49	3	3449	551	-	1.00	49
2101	11	0	-	-	-	-	11
1923	1	0	5075	458	-	1.00	1
2321	7	6.6×10^3	3670	410	0.266	1.54	11
2318	13	1.0×10^3	3784	360	0.251	3.18	41
2323	337	1.1×10^3	4319	417	0.256	1.90	640
2281	32	2.3×10^3	4955	527	0.229	1.00	32
2324	9	3.7×10^3	4534	417	0.252	1.99	18
2282	3	9.9×10^4	4725	423	0.252	2.06	6
2312	1	4.2×10^2	5253	438	-	2.33	2
2280	32	2.5×10^4	6411	476	0.206	1.00	32
2027	9	2.3×10^4	4810	542	-	1.00	9
2201	7	1.9×10^4	5540	418	-	2.57	18

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_o, cts/10 µs	Thermal neutron flux monitor, $T_{1/2}$, µs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
1946	12	1.4×10^3	4012	501	-	1.00	12
1374	19	1.3×10^4	4511	387	-	3.22	61
2018	5	5.4×10^4	4960	507	0.238	1.00	5
2025	23	1.8×10^4	6021	493	0.199	1.00	23
2030	11	1.4×10^4	6052	489	0.211	1.00	11
2016	21	1.2×10^4	5586	497	0.223	1.00	21
2035	31	2.3×10^4	5330	503	0.218	1.00	31
2047	25	3.1×10^4	5608	510	0.216	1.00	25
1935	15	4.8×10^4	5764	496	0.219	1.00	15
2011	27	8.5×10^4	5788	446	0.221	2.69	73
1944	1	1.5×10^4	4208	437	0.253	1.80	2
2042	7	3.0×10^4	5290	522	0.186	1.00	7
2083	47	2.2×10^4	6374	487	0.204	1.00	47
2004	45	1.5×10^4	4926	538	0.254	1.00	45
2017	55	2.5×10^3	5141	525	0.227	1.00	55
1917	39	8×10^3	3381	372	0.159	1.89	74
1925	35	1.3×10^2	3540	352	-	3.61	126
2006	43	3.5×10^4	6500	491	0.210	1.00	43
1956	37	1.8×10^4	3650	442	0.258	1.52	56
1955	73	2.2×10^4	3938	469	0.256	1.00	73

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_0, cts/10 µs	Thermal neutron flux monitor, $T_{1/2}$, µs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
2014	79	7.2×10^4	5077	530	0.228	1.00	79
2151	205	3.6×10^1	4259	532	-	1.00	205
2029	176	3.2×10^4	4866	529	0.192	1.00	176
2009	186	1.9×10^4	4893	530	0.232	1.00	186
1918	133	1.9×10^4	3419	367	0.260	1.93	257
2031	166	1.0×10^2	2577	345	-	2.50	415
1932	121	1.2×10^4	3728	366	0.248	2.30	278
1676	327	4.7×10^1	5178	528	-	1.00	327
1996	123	5.4×10^2	4806	445	-	2.10	258
2026	627	3.4×10^2	4706	534	-	1.00	627
1817	476	2.1×10^3	4635	495	-	1.00	476
1947	360	2.0×10^2	4028	528	-	1.00	360
2107	427	0	4781	505	-	1.00	427
1949	495	4.7×10^2	3491	418	-	1.89	936
2015	394	5.7×10^2	4193	527	-	1.00	394
2045	653	3.2×10^4	4089	454	0.258	1.00	653
1927	367	1.6×10^2	4778	494	-	1.00	367
2008	606	3.9×10^2	4975	492	-	1.00	606
2021	380	1.7×10^3	4574	534	0.256	1.00	380
1060	606	1.4×10^3	4165	458	0.278	1.00	606

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_o, cts/10 μs	Thermal neutron flux monitor, $T_{1/2}$, μs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
1954	318	1.5×10^2	4300	457	0.266	1.00	318
1920	44,019	1.4×10^4	3727	435	0.257	1.56	68,670
1952	272	3.6×10^3	4201	439	0.254	1.80	490
2053	936	3.2×10^3	2881	475	0.285	1.00	936
1772	11,200	4.9×10^2	4613	514	0.227	1.00	11,200
1771	9,133	4.3×10^2	4630	517	0.227	1.00	9,133
1957	1,400	3.4×10^4	3881	345	0.255	4.12	5,768
1930	1,200	1.2×10^3	3753	428	0.284	1.58	1,896
1977	853	5.6×10^2	4558	481	0.243	1.00	853
1776	2,367	2.4×10^2	4329	417	-	1.86	4,403
2153	13,533	7.4×10^3	3864	459	0.245	1.00	13,533
2111	733	1.4×10^4	-	-	0.249	-	733
1770	9,267	3.5×10^2	4938	524	-	1.00	9,267
1961	4,247	2.5×10^4	4527	432	0.251	1.96	8,324
1777	2,000	8.0×10^1	4493	428	-	1.95	3,900
2046	943	1.2×10^4	8186	495	-	1.00	943
1631	3,209	1.5×10^4	3693	380	-	2.25	7,220
1798	1,800	1.7×10^3	4315	477	0.242	1.00	1,800
1789	27,600	1.1×10^3	3907	527	-	1.00	27,600
2059	1,733	1.4×10^3	4487	510	0.246	1.00	1,733

Table 1. ORNL waste drum active and passive neutron data (cont.)

Drum identification	Uncorrected fissile mass ^{239}Pu equivalent, mg	Passive net neutron source strength, n/s	Thermal neutron flux monitor, A_0, cts/10 μs	Thermal neutron flux monitor, $T_{1/2}$, μs	(Passive neutron shielded totals) ÷ (system totals)	Matrix response factor	Matrix corrected fissile mass ^{239}Pu equivalent, mg
2108	847	9.0×10^2	4609	451	0.239	1.50	1,271
1783	867	5.3×10^2	5365	464	-	1.00	867
1941	4,267	6.7×10^3	3430	509	0.282	1.00	4,267
1791	27,200	3.4×10^4	4771	501	0.211	1.00	27,200

As mentioned above, the interpretation of the passive neutron data can be a difficult task. However, in some cases the task is simplified by the availability of information detailing the elemental and isotopic content of a particular waste drum. The following examples will illustrate the methods employed in the interpretation of these passive neutron data. The elemental and isotopic data for these drums were obtained from ORNL Nuclear Materials Intra-Laboratory Transfer sheets (UCN-2681). These sheets contain a compilation by the generator of the amounts of accountable isotopes contained in the drum, such as uranium and plutonium isotopes.

Example 1 - Drum ATN-2396

Net passive neutron source strength: 9.94×10^3 n/s,
SHTOT/SYSTOT = 0.267,
SYSCOINC = 127.7 cps,
SYSTOT/SYSCOIN = 10.9,
P3/P2 = 0.071 ± 0.010,
P1/P2 = 10.395 ± 0.190,

where

SHTOT = net shielded detectors total count rate (cps),
SYSTOT = combined net shielded and net unshielded detectors (ie., system) total count rate (cps),
SYSCOINC = net system coincidence count rate (cps),
P3/P2 = neutron multiplicity ratio, triples to doubles,
P1/P2 = neutron multiplicity ratio, singles to doubles.

The ratio SHTOT/SYSTOT is an indicator of the amount of moderating material contained in the waste drum. If the calculated ratio is 0.260 or greater, the matrix contains little moderator; and a "bare model" can be used as an aid in data interpretation.

The ratio SYSTOT/SYSCOINC is an indicator of the contribution by uncorrelated neutron sources [(α,n)reactions] to the total passive neutron source strength. A ratio of 13 or greater indicates the presence of a significant amount of (α,n) reactions, such as those caused by the alpha decay of ^{241}Am.

The gamma-ray data acquired from drum ATN-2396 indicate the presence of the following isotopes: ^{241}Am, ^{239}Pu, ^{241}Pu, ^{235}U, and ^{235}U daughters.

The gamma-ray data are used in conjunction with the passive neutron data to identify the fissile material and spontaneous fission emitters contained in the waste drum.

The nuclear material transfer sheet data are given below.

Element Weight	Isotope
6 g Pu	11.5% ^{240}Pu

For 6 g of 11.5% ^{240}Pu (or 0.7 g of ^{240}Pu), one would expect a SYSCOINC rate of 8.8 cps. From the data, SYSCOINC = 127.7 cps. Therefore, 93% of the spontaneous fissions can be attributed to some isotope(s) other than ^{240}Pu. The neutron multiplicity ratio P3/P2 is very close to that obtained for a pure ^{244}Cm source. Also, the neutron multiplicity ratio P1/P2 indicates the presence of a mixture of ^{240}Pu and ^{244}Cm. Consequently, the passive neutron signal is dominated by the radioisotope ^{244}Cm. From calibration data previously obtained, the net passive neutron signal, after accounting for the small ^{240}Pu contribution, indicates the presence of 1 mg of ^{244}Cm.

The assay report for drum ATN-2396 can be found in Appendix C.

Example 2 - Drum ATN-2399

Net passive neutron source strength: 4.38×10^3 n/s,
SHTOT/SYSTOT = 0.271,
SYSCOINC = 31.99 cps,

SHCOINC = 3.184 cps,
SYSTOT/SYSCOINC = 19.2,
P3/P2 = 0.053 ± 0.010,
SHTOT/SHCOINC = 52.2,

where SHCOINC = net shielded coincidence count rate (cps).

The gamma-ray data indicate the presence of the following isotopes: ^{241}Am, ^{239}Pu, and ^{235}U.

The nuclear material transfer sheet data are given below.

Element Weight	Isotope
22 g Pu	11.50% ^{240}Pu
10 g U	93.3% ^{235}U

For 2.5 g of ^{240}Pu, one would expect a net passive neutron source strength of 4.35×10^2 n/s. The data show approximately ten times that amount (i.e., 4.38×10^3 n/s). The SHTOT/SYSTOT ratio allows one to use the "bare" model in the calculations. The high SHTOT/SHCOINC ratio implies a large source of (α,n) neutrons. This implication is corroborated by the strong 59.5 keV ^{241}Am line in the gamma-ray spectrum.

The neutron multiplicity ratio P3/P2 is very close to that obtained from a calibration source containing 1.24 g of ^{240}Pu (i.e., 0.054 ± 0.004). For 1.24 g of ^{240}Pu, one would expect a shielded coincidence rate (SHCOINC) equal to 1.30 cps. From the data, SHCOINC = 3.18 cps. Therefore, a shielded coincidence rate of 3.18 cps implies the presence of approximately 3 g of ^{240}Pu. This conclusion is consistent with the information supplied by the nuclear material transfer sheets.

Example 3 - Drum ATN-2500

Net passive neutron source strength: 2.21×10^4 n/s,

SHTOT/SYSTOT = 0.268,

SYSTOT/SYCOINC = 11.6,

P3/P2 (system) = 0.118 ± 0.038,

P3/P2 (shielded) = 0.0338 ± 0.049,

SHCOINC = 29.56 cps,

SHTOT/SHCOINC = 28.

The gamma-ray data indicate the presence of the following isotopes: ^{241}Am, ^{239}Pu, ^{241}Pu, and ^{238}Pu.

The nuclear material transfer sheet data are given below.

Element Weight	Isotope
38 g Pu	41.87% ^{240}Pu

For 15.9 g of ^{240}Pu, one would expect a net passive neutron source strength of 2.77×10^3 n/s. The passive neutron data show approximately ten times that amount (i.e., 2.21×10^4 n/s).

The SHTOT/SYSTOT ratio allows the use of the "bare" model in the calculations. The high system multiplicity ratio P3/P2 indicates the presence of ^{252}Cf. This indication is confirmed by the high shielded multiplicity ratio P3/P2. Consequently, the passive neutron source signal is dominated by the ^{252}Cf spontaneous fission neutrons. The low SHTOT/SHCOINC ratio indicates few excess (α,n) neutrons.

Example 4 - Drum ATN-2501

Net passive neutron source strength: 2.64×10^3 n/s

SHTOT/SYSTOT = 0.273

SYSTOT/SYSCOINC = 19.7

P3/P2 = 0.070 ± 0.009

The gamma-ray data indicate the presence of the following isotopes: ^{241}Am, ^{239}Pu, ^{241}Pu, ^{235}U, and ^{237}U.

The nuclear material transfer sheet data are given below.

Element Weight	Isotope
16 g Pu	8.4% ^{240}Pu

For 1.34 g of ^{240}Pu, one would expect a net passive neutron source strength of approximately 2 × 10^3 n/s. Since the net passive neutron source strength is 2.64 × 10^3 n/s, one can assume that the overwhelming majority of the detected neutrons originate from ^{240}Pu. This assumption is confirmed by the neutron multiplicity ratio P3/P2 and the system neutron coincidence rate SYSCOINC of 18.83 cps. This rate corresponds to approximately 1.5 g of ^{240}Pu. Therefore, the passive neutron data verifies the presence of approximately 1.5 g of ^{240}Pu.

In each of the above cases, however, the active neutron data did not accurately reflect the fissile content (^{239}Pu and/or ^{235}U) as reported in the nuclear material transfer sheets. This trend is shown in Table 2.

Table 2. NAS active scan results for selected drums

Drum ID	NAS results (mg ^{239}Pu equivalent)	SNM transfer sheet data (mg ^{239}Pu equivalent)
ATN-2396	3.4 × 10^3	21.0 × 10^3
ATN-2399	5.1 × 10^3	26.0 × 10^3
ATN-2500	5.5 × 10^2	20.0 × 10^3
ATN-2501	2.6 × 10^3	17.7 × 10^3

The NAS-DDT was originally designed to assay very low levels of TRU-burdened waste (i.e., <10 nCi/g TRU content). Gram quantities of fissile material can cause difficulties, especially if the material is in dense form, such as a radioisotope source. Interrogation of the waste in the NAS is accomplished by thermal neutrons. The mean free path of a thermal neutron in, for example, plutonium oxide is approximately 0.001 m. Consequently, fissile material that is more than 0.001-m thick is not accurately assayed.

In the case of the drums listed in Table 2, however, there is another method available for obtaining the fissile mass content, namely, the SGS. Drum ATN-2501 contains an essentially pure ^{240}Pu passive neutron source signal and, consequently, can be used as a calibration source. The passive neutron data confirmed the presence of 15 g of ^{239}Pu in drum ATN-2501. Based on this information, the assumption that the gamma-ray attenuation is approximately the same for each drum (a good approximation for high-energy gamma rays), one could use the ^{239}Pu gamma-ray peak areas (325.04 and 413.71 keV) obtained from drum ATN-2501's spectrum as being proportional to 15 g of ^{239}Pu. The plutonium gamma rays obtained from the spectra of other waste drums (if available) would then yield the quantity of ^{239}Pu contained in that drum. This synergistic relationship between the NAS and SGS illustrates the important role each instrument plays in the characterization of ORNL waste.

Table 3 presents the data and the results obtained from the assay of SRP ^{238}Pu-contaminated glove box waste contained in four 208-L galvanized steel drums. These assays represent the case where the elemental and isotopic content of the waste is firmly established and does not vary from drum to drum. Only the element plutonium in its 238, 239, and 240 isotopes is present to any appreciable extent in a waste drum. The plutonium isotopic ratios are also known. The assay results are, therefore, easier to interpret, and meaningful results can be obtained.

Table 3. Passive and active neutron assay measurements on SRP ^{238}Pu waste drums

Drum ID	Av net passive systems totals rate, cps	Av net passive neutron coincidence rate, cps	Active mg ^{239}Pu equivalent	Drum weight, kg	^{238}Pu activity, nCi/g
S/N-139	8.30 ± 0.32	0.53 ±0.06	0 ± 1	72.3	410 ± 16
S/N-140	281.2 ± 0.9	0.93 ± 0.31	15.5 ± 5[a]	109.5	9168 ± 29
S/N-141	16.9 ± 0.35	1.12 ± 0.07	-0.5 ± 1.0	54.3	640 ± 13
S/N-138	-0.50 ± 0.28	0.03 ± 0.04	-1.5 ± 1.0	70.5	-25 ± 14
Background	25.00 ± 0.16	0.30 ± 0.02	-	-	-

[a] The increase in the total error is the result of uncertainties in the matrix correction factor.

The final column depicts the calculated ^{238}Pu activity contained in each drum in units of nanocuries per gram, using the approximate experimental relationship that each millicurie of alpha activity should generate a total neutron source of approximately 2 n/s from the reaction $^{17,18}O(\alpha,n)$. Each drum was weighed, and this total drum weight was used in the ^{238}Pu activity calculation.

The active measurements indicated less than 1 mg of ^{239}Pu in three of the four drums. Drum S/N-140 was assayed to contain (after matrix correction) 15 ± 5 g of ^{239}Pu. From data obtained from the Savannah River Laboratory (SRL), it was indicated that the ^{238}Pu glove box waste would be expected to contain the following plutonium isotopes: ^{238}Pu (80% to 84%), ^{240}Pu (2%), and ^{239}Pu (14% to 18%). If midrange values are assumed, then 15 ± 5 mg of ^{239}Pu (^{239}Pu assumed to be 16% of total plutonium and ^{238}Pu assumed to be 82% of total) implies the presence of 77 ± 9 mg of ^{238}Pu.

Using a measured ^{238}Pu half-life of 87 years, one calculates that 1 mg of ^{238}Pu has a total alpha activity of 17.2 mCi. Thus, 77 ± 9 mg of ^{238}Pu corresponds to a total drum alpha activity of

12,000 ± 1500 nCi/g. This agrees within error limits with the direct passive neutron value of 9200 ± 2000 nCi/g, allowing for an overall systematic error in the passive measurements of ±20%. This example illustrates the importance of both the active and passive neutron assays in obtaining meaningful results and how each can serve as a check for the other.

It should be noted that for the other three drums, the fissile masses of ^{239}Pu expected, based on the indicated ^{238}Pu activities, all correspond to well under 1 mg. Thus, the measured "zero" ^{239}Pu values are consistent with the measured ^{238}Pu activities from the passive neutron data.

It should also be noted that all data obtained were "routine" in the sense that all passive neutron assays on the four drums were 400 s in duration, and all active assays used 2000 pulses of the neutron generator (40 s elapsed time).

WASTE MATRIX EFFECTS ON NEUTRON DATA

Waste matrix compositions can have a profound effect on the active and passive neutron experimental data. For the NAS, the moderation and absorption of the interrogating thermal-neutron flux and moderation of the signal prompt-fission neutrons are important factors. As briefly discussed in another section of this report, two parameters (A_o and $T_{1/2}$) are presently used to determine the effects of absorber and moderator material contained in the waste drum. These variables are then used to effect a matrix correction to the assay data. Over 20 208-L drums were filled with varous matrices to simulate a wide range of absorber and moderator loadings. A calibration source was placed in 30 different locations in each drum and were assayed by the NAS. A set of matrix response factors, K, was determined for a range of $T_{1/2}$. Approximately 70% of the waste drums assayed do not require a matrix correction. Matrix corrections have been discussed in more detail in previous reports (see References 3 and 4).

In order to achieve a better understanding of the effects of the matrix on the neutron response of the system — in particular, the response to the prompt-fission neutrons — a new in-cavity neutron detector [drum flux monitor (DFM)] has been developed by LANL. The DFM is positioned in the right-hand corner of the assay chamber and is surrounded by cadmium metal on all sides except that directly facing the drum. The cadmium shields the detector (a low-pressure ^{3}He proportional counter) from the interrogating thermal-neutron flux.

Mock-up matrices containing calibration sources have been assayed at LANL by an NAS equipped with a DFM. The mock-up matrices are listed below:

1. 17.5 kg polyethylene and 20 kg vermiculite,
2. 29 kg peat moss,
3. 450 lb iron scrap,
4. 29 kg peat moss and 0.25 kg borax,
5. 52.5 kg polyethylene and 12.2 kg vermiculite,
6. 17.5 kg polyethylene, 22.5 kg vermiculite, and 2.0 kg borax,
7. 29 kg peat moss and 0.5 kg borax,
8. 900 lb iron scrap,
9. 29 kg peat moss and 1.0 kg borax, and
10. 29 kg peat moss and 2.0 kg borax.

All matrices were in 208-L stainless steel drums. The polyethylene beads and peat moss were used as moderator matrices, while the borax and iron were used as absorber matrices. The vermiculite was used as a substrate.

Three matrix correction equations have been developed for computing the corrected fissile content of a waste drum. Each equation is used for a defined range of the ratio FM/DFM. These equations are given below in Table 4.

Table 4. NAS fissile mass matrix-correction equations

FM/DFM	Matrix-corrected fissile mass (mg ^{235}U equivalent)
<2.0	$[(A - B)/C - D]/0.187$
>2.0 and <15.0	$\{[(A - B)/C - D]/0.187\}[0.133 + 1.2507 \ln(C/E)]$
>15.0	$[(A - B)/E - F]/0.833$

where,

A = SHTOT (0.71 ms to 4.70 ms),
B = SHTOT (5.71 ms to 9.70 ms),
C = FM (0.71 ms to 4.70 ms),
D = [SHTOT (0.71 ms to 4.70 ms) - SHTOT (5.71 ms to 9.70 ms)]/ FM (0.71 ms to 4.70 ms),
E = DFM (0.71 ms to 4.70 ms), and
F = [SHTOT (0.71 ms to 4.70 ms) - SHTOT (5.71 ms to 4.70 ms)]/ DMF (0.71 ms to 4.70 ms).

The matrix correction for the assay data is thus obtained. The DFM allows the assay data to be corrected by smaller increments, 15 vs 4, with just the flux monitor tube alone. Matrix corrections will be more precise and sensitive.

The DFM is to be installed in the Oak Ridge system in early FY 1985.

TECHNOLOGY TRANSFER

One of the major objectives of the cooperative program between LANL and ORNL is to provide a demonstration and training facility for personnel from DOE and its contractors. The high interest in the NAS and SGS continued this year as evidenced by Table 5, which lists visitors to the TWDAF.

Table 5. Visits to TRU Waste Drum Assay Facility, September 1983 - August 1984

Date	Representative	Affiliation
09/13/83	J. S. Sanchez, F. C. Ruesgas, A. C. Gomez, A.G. de la Huebra Gordo	Spanish nuclear agency
11/01/83	B. Schappel, R. Taylor	Oak Ridge Gaseous Diffusion Plant (ORGDP)
11/08/83	B. R. Barré	French Embassy, Washington, D.C.
11/08/83	F. Brés	Commissariat Energie Atomique (CEA), Paris, France
11/09/83	L. F. Hary	GAT, Portsmouth, Ohio
11/09/83	D. Ziegler	Rockwell International/ Rocky Flats Plant
11/09/83	J. L. Williams	Paducah Gaseous Diffusion Plant (PGDP)
11/09/83	R. Salazar	Lawrence Livermore National Laboratory (LLNL)
11/09/83	B. D. Helton	SRP
11/09/83	J. D. Wells	EG&G
11/09/83	R. B. Weidner	FMPC, Cincinnati, Ohio
11/09/83	J. Couch	Fermilab
11/11/83	O. Towler	Savannah River Laboratory (SRL)
11/11/83	J. L. Warren	LANL
11/11/83	J. Dugger	DOE - Headquarters
11/11/83	F. J. Homan	ORNL

Table 5. Visits to TRU Waste Drum Assay Facility,
September 1983 - August 1984 (Cont.)

Date	Representative	Affiliation
11/11/83	J. Cortella, J. Ohmann, D. Mangin	CEA, Valduc
11/11/83	R. Gros, F. Delobeau, H. Kerviler	CEA, Paris, France
02/14/84	J. DuBois	General Accounting Office
02/14/84	L. Jones	Inspector General's Office
02/17/84	K. E. Mersman, T. R. Herold	SRP
02/23/84	A. Stubbs	New England states coalition for radioactive waste management
02/24/84	Major General W. W. Hoover	Office of Military Application, DOE
03/27/84	G. D. Baudin, P. E. Pottier	CEA, France
04/04/84	L. Stewart	WATE-TV
04/11/84	T. Fabian	*Nuclear Waste News*
04/26/84 - 04/27/84	L. Shope, M. O'Neall	Sandia National Laboratory (SNL)
06/01/84	S. Neuhauser	SNL-TTC
06/01/84	G. Daer	Waste Isolation Pilot Plant (WIPP)
06/01/84	M. H. McFadden	DOE/ALO
06/01/84	S. W. Woolfolk	WIPP/TSC
06/12/84	H. Witte	Waste Chemicals, Ramsey, New Jersey

Table 5. Visits to TRU Waste Drum Assay Facility, September 1983 - August 1984 (Cont.)

Date	Representative	Affiliation
06/12/84	E. W. McDaniel, J. O. Blomeke	ORNL
07/13/84	R. J. Greer, B. J. Campbell, R. E. Green	DOE - Oak Ridge Operations (DOE-ORO)
08/24/84	W. R. Morris, C. W. Mallory, J. W. Peel, D. A. Wilkie	Westinghouse
08/24/84	D. E. Large	DOE-ORO

Further activities included:

1. in October 1983, an ORNL Public Relations Department News Release on the TRU Waste Drum Assay Facility was completed;
2. on February 21, 1984, K. W. Haff presented a history and current status of the TRU assay program at ORNL to J. DuBois of the U.S. General Accounting Office and L. Jones of the U.S. Inspector General's Office;
3. F. J. Schultz presented a review of the ORNL TRU Waste Assay Program to attendees of the TRU Waste Management Seminar held at ORNL May 17;
4. on August 17, 1984, F. J. Schultz gave a presentation on nondestructive assay of transuranic wastes to Dr. T. Okada, Power Reactor and Nuclear Development Corporation, Tokai-mura, Japan;
5. F. J. Schultz attended the MC&A Representatives Meeting held in Oak Ridge, August 21, 1984, and presented a discussion to the participants on nondestructive assay activities at ORNL; and
6. on August 23, 1984, F. J. Schultz presented a technical discussion on TRU waste assay to a Federal Republic of Germany waste management group.

AN INTERESTING APPLICATION

An interesting application employing the active scan of the NAS was recently undertaken at ORNL. Enriched uranium reactor fuel elements were assayed prior to placement in the core of the Oak Ridge Research Reactor (ORR). The ^{235}U content of each fuel element was verified by the assay for safeguards accounting.

One fuel element was chosen as a "standard" to establish the calibration constants used in calculating the fissile mass. Additional elements were considered "unknowns" and were assayed based upon the calibration constants obtained from the assay of the first fuel element. The results are presented in Table 6. The accepted fuel element loading is 285 g of ^{235}U.

Table 6. NAS examination of ORR fuel elements

Fuel element identification	Grams ^{235}U
T470	286
T471	289
T472	289
T473	288

Average grams ^{235}U = 288.0 ± 1.4 g

Percent error = 1.1%

Future experiments will encompass the assaying of fuel elements containing smaller loadings of ^{235}U in the 250- to 275-g range. This will establish the sensitivity of the measurement technique. That is, does the NAS possess the ability to detect a difference of ±10 g (or less) on a total loading of 285 g (a 3.5% sensitivity)?

TRU WASTE DRUM SAMPLING FACILITY

As mentioned in the previous topical report ORNL-6007, a validation study has been undertaken at ORNL to determine the ability of the NAS to accurately predict the known concentration of TRU isotopes contained in a waste drum. Drums would be nondestructively assayed at the TWDAF and then transferred to another facility for a destructive examination. This examination would include removing the waste from the drum, segregating the waste according to waste type (cellulosics, plastics, metals, and glass and ceramics), preparing the waste for sampling (shredding and crushing), and analyzing representative samples for TRU isotopic content.

A facility, the TRU Waste Drum Sampling Facility, has been built at ORNL to perform the operations described above. The operational procedures,[7] as well as the equipment, have been examined and reviewed by various ORNL safety committees and were found to meet all safety and radiation-hazard criteria set forth in the ORNL safety and health physics manuals. Permission has been given to the facility operators to begin "hot" operations, that is, to examine waste drums.

Recently, after much deliberation and discussion, ORNL has recommended to DOE-ORO to discontinue the destructive examination of waste drums.[8] Several reasons were given for this decision. They are briefly discussed below.

1. Operating personnel would be exposed to safety and radiation hazards during drum opening and loading procedures. These risks are minimal, but still exist.
2. A nondestructive verification involving calibration sources in known matrices would establish the accuracy of the NDA technique more directly and at less cost.
3. Early cost estimates for the destructive examination have increased by 60%. More manpower and time are required for the examination than were previously estimated.

It has been recommended[8] that the following be implemented:

1. the pace of the ORNL-LANL nondestructive verification program be accelerated, and
2. a meeting with personnel from the TRU Waste Systems Office (TWSO), DOE-ORO, ORNL, and LANL be held to discuss the above topics and determine the course of action.

The facility remains in a standby condition until a final decision is made.

CONCLUSIONS

The neutron assay system for the examination of bulk transuranic waste in 208-L and 114-L drums has been sufficiently developed and demonstrated by LANL and ORNL for use by most DOE-DP sties. Other sites, such as the Savannah River Plant and the Rocky Flats Plant, generate a minimum number of isotopes per waste drum. The plutonium isotopes, mainly ^{239}Pu and ^{240}Pu, and ^{235}U are the dominant species. Interpretation of the active and passive neutron data is greatly simplified in these cases (see data interpretation of SRP drums in previous section). Also, in many instances, the waste matrix is better characterized and more homogeneous than is the ORNL waste.

The ORNL CH-TRU waste is the most complex in the system. Over 20 alpha-emitting TRU isotopes, several of which are spontaneous fission neutron emitters as well, have been identified. The higher neutron backgrounds generated by some of these isotopes interfere to various degrees with the nondestructive neutron assay of many waste drums. The system is now capable of assaying the majority of the CH-TRU wastes in the DOE-DP system. Work planned for fiscal year 1985 will determine if those complex portions of ORNL CH-TRU waste can be examined with the data produced being of the same quality that the system is capable of producing for the less complex waste at other sites. It is clear, however, that more research in the area of passive neutron data interpretation is required before an accurate estimate of the total TRU content of ORNL CH-TRU waste is achieved.

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6. L. Rice, III, Operation Manual for Sandia MA165 Neutron Generator System, SAND80-1405, Sandia National Laboratory, November 1980.

7. F. J. Schultz et al., ORNL TRU Waste Drum Assay and Sampling Facilities: Operational Procedures Manual, ORNL/CF-83/253, January 1984.

8. T. H. Row, Director, Nuclear and Chemical Waste Programs, Oak Ridge National Laboratory, letter to C. L. Matthews, Chief, Fission Reactor Branch, U.S. Department of Energy, Oak Ridge Operations, August 14, 1984.

APPENDIX A

Segmented Gamma Scanner System Software Program, CAN

```
Hello, this is the CAN program.

I can offer you the option, before we begin, of seeing
a checklist for the scanner and data acquisition hardware.
Do you want to see that checklist? Please type "YES" or
"NO", with a carriage return. YYESES

  This is the setup/startup checklist for the Los Alamos/ ORNL gamma
ray scanner program. This program will operate the CANBERRA Barrel
Scanner to obtain one 8000-channel spectrum from each vertical segment
in the waste drum. These spectra are stored on the diskette you specify
in the form readable by the LRS 3500 Multi-Channel Analyzer (MCA)
programs. We will now go through the scanner hardware step by step.

   First, turn on the CANBERRA 2225B SCAN TABLE CONTROLLER's AC power.
Then touch the green button for marked "table rotation on". The barrel
should start to rotate. If the barrel does not rotate, the most
probable cause is a "LOCKUP" caused by over-running the vertical travel
stops. This happens when the controller receives a confusing sequence
of manual-operation signals. It also happens if the CAMAC was turned
off with the CONTROLER on. The SCAN TABLE CONTROLLER documentation
tells how to open the back of the scanner and drive the table back into
the normal operating range. If the barrel is rotating, test the
vertical motion by pushing the ADVANCE and RETURN buttons. Push them
only one at a time---do not push another until vertical motion has
stopped.

   When you have done this push the CONTinue key on the keypad.



   The Lithium-Drifted Germanium detector (GeLi) is powered from NIM
bin number 1. Verify that the AC power is ON. Note that there is a
High Voltage (HV) power supply for the GeLi in that NIM bin and a
circuit breaker on the AC power line. If the AC power should go off,
the circuit breaker will keep the AC power off to keep the detector
from being damaged when the HV supply comes back on. If the AC power
has gone off, you must turn on the dectector high voltage as follows:
   1) Turn the HV power supply down to zero and then OFF.
   2) Reset the AC circuit breaker.
   3) Verify that NIM AC switch is ON and power is ON.
   4) Turn the HV supply ON.
   5) Verify that HV polarity is POSITIVE.
   6) Raise the high voltage to +2800 volts, no faster than 100 volts
per second.
```

The GeLi preamplifier power comes from the back of the CANBERRA 2020 amplifier in NIM bin number 1. The GeLi signal connects to the 2020 amplifier input. The amplifier output connects to the LeCroy 3512 Analog-to-Digital Converter (ADC) in the CAMAC mini-crate. Please verify all that.

When you are ready to proceed touch CONTinue.

The GeLi signal from the CANBERRA 2020 amplifier should connect to the input of the LeCroy 3512 ADC in the mini-crate slots 1 and 2. The 3512 connects to a LeCroy 2551 scaler in slot number 4 by a rear-panel connection cable. The 2551 input number zero should connect to the JOERGER 10-MHz clock in slot 5.
The 2551 collects the spectrum acquisition times, both live time (corrected for ADC digitizing time) and clock time (normal elapsed time). Those times are printed for each segment's spectrum at the close of the segment's spectrum acquisition time. Check those on the printout to verify that these modules are working.
The KINETIC SYSTEMS 3063 in slot 6 connects to the 2225B SCAN TABLE CONTROLLER. That allows the LeCroy 3500 to control the vertical motion. When the 3500 program begins to position the scan table watch for the table's motion. Successful positioning verifies that connection.

Touch CONTinue when you are ready to proceed.

Each spectrum file I write can have the spectrum energy calibration built into the file. To do this I need a file named CALIB.00 stored on this diskette. Any time the GeLi setup is changed, and perferrably every morning, this file should be re-made so that its energy calibration is correct. Here's how to make that file:

1) Put this diskette in drive B and GAMMA mKIIIA in drive A
2) Boot the MCADOS operating system and run GOTOMCA
3) Boot the operating system (again) and run GAMMA
4) Select SYSTEM ARCHETECTURE and set up for:
    a) set up a 3511 in slot 1 of the mini-crate
5) Reset and select MODULE SETUP and set up that 3511 for
    a) Spectrum name CALIB
    b) Spectrum size 8K
    c) Select the 8K memory segment
6) Acquire a spectrum of known sources and initiate an energy calibration as described in the MCA Operator's Manual, chapter XI-C.
7) Use function DISK to select drive B.
8) Use function SAVE to record the spectrum as experiment "00".

(Touch CONTinue to continue)
Thank you. I will be able to proceed with the data acquisition now.

```
Now I will ask the line printer for a new page.
Please watch for the page ejection. If the printer is not
"ONLINE" or if the printer is otherwise unable to respond
I will not be able to proceed either. You can restart the
printer by turning the printer OFF and ON. That will reset
the printer's internal microprocessor. Then you should
simultaneously type "RESET" and "HERE IS" on the 3500.
That will reset the 3500's microprocessor. Then boot the
operating system and start the CAN program again.
```

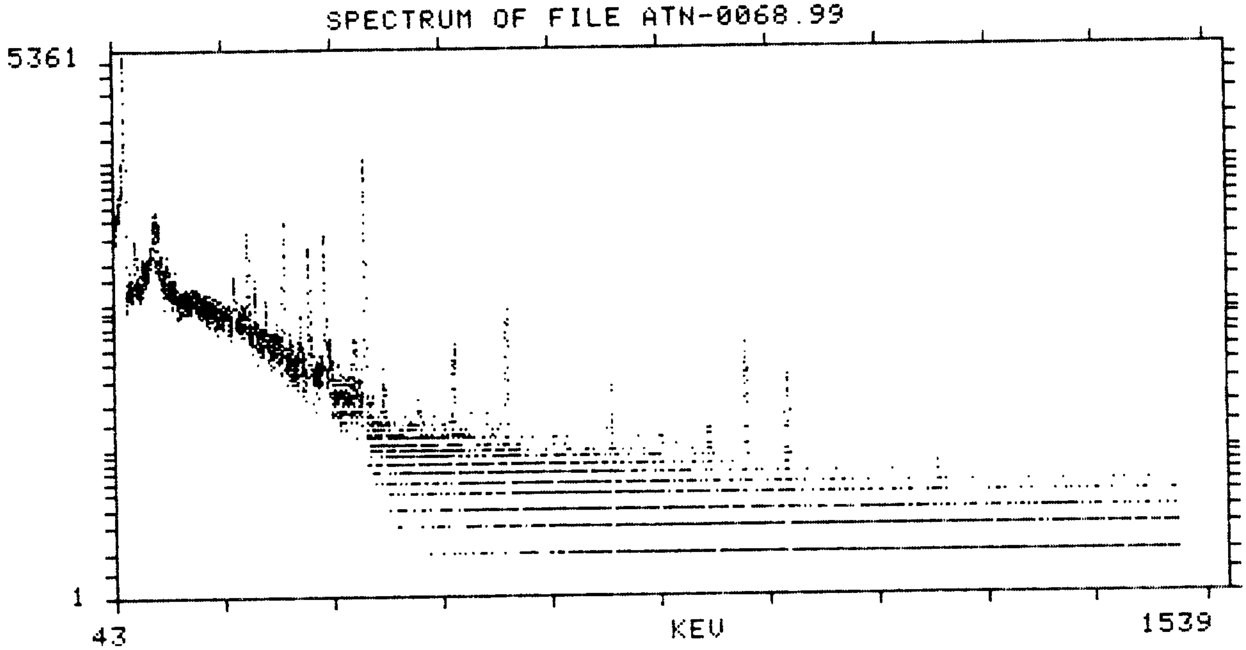
```
The printer seems to have responded normally.
I will proceed to collect the information I need to set up
this scan.


I can plot the spectra. I can plot all the spectra
or I can plot only the first spectrum, or I can plot
none of the spectra. Please type "ALL" or "ONE" or "NONE",
and end it with a carriage return. NONWE

How many segments do you want to measure?
Type the number with a carriage return. 1



I am designed to operate from a diskette in the "A"
diskette drive, but I can record the segment spectra on the
A drive or the B drive. Please select the drive on which
the spectra will be recorded. Type "A" or "B", and end with
a carriage return. (Incidentally, if this program is not
running from the A diskette drive, simulatneously type
CONTROL and C. Then move the diskette to the A drive and
start the program again.) Note that I will need to read
your diskette for available space, so you must have the
diskette in the drive and the drive door closed before you
tell me which drive you want the spectra recorded on. B



The MCADOS operating system (under which the
analysis programs operate) would find  400 Kilobytes
of space available on diskatte B.


I find enough space to record the spectra.

Please do not open the diskette drive door again until
I have finished recording the spectra. If you open the
door the operating system will declare the diskette
to be READ ONLY (R/O) which will stop my attempt to
record the spectra.


Please set the SEGMENT SIZE CONTROL on the CANBERRA
model 2225B SCAN TABLE CONTROLLER to the value  3425
I cannot check that when I authorize the TABLE CONTROLLER
to move the barrel. Consequently that must be set correctly.
I will wait until you check it and push the "START" KEY.
Then I will set up the scanner's starting location.
That should take only a few seconds.
I will monitor the "STOP" Key if you want to stop the setup process.
```

```
How many seconds may I spend on each segment?
Type the number of seconds with a carriage return. 10

The total time required will be approximately
     52. seconds.
The run should finish at approximately 16: 0:12

What is the name of this barrel? The name you
give me will be used to label the data storage files.
Type the name followed by a carriage return. TEST

Can scan procedure for barrel TEST
Begun  8/29/84 15:59:28

Type any comments you have about this barrel.
You may use as many lines as you want. End each line
with a carriage return. Type "*" at the start of a line
when you are finished typing comments.

*
```

```
Thank you. I am ready to begin the scan now. I hope you
are confident that all the equipment is ready. Remember
you can stop the scan with the STOP key on the keypad.

I will start the scan when you push the START key on the
keypad.
```

```
Scan in process, now in segment  1
Spectrum for segment  1 completed  8/29/84 16: 0: 0
Live Time=   10.0 Clock Time=   10.0
Dead Time=    .%
File name is TEST    01

Spectrum has energy range from      52.19 to    1833.00


Scan ended normally


Scan ended normally.
I will sum the spectra.


Summed spectra written in file TEST    .99
File has total live time of        10.


Spectrum has energy range from      52.19 to    1833.00
```

71/72



Analysis of spectrum ATN-0068.99 at 0/ 0/ 0 0: 0: 0
Acquisition time in seconds = 537.0

APPENDIX B

Neutron Assay System Software Program, SNEUT

```
Disk Format Version 4.1

Format disk in drive B ? Y
Format D(ouble), S(ingle), or O(ld) ? D
Formatting
Verifying
Copying System Tracks

Format disk in drive B ? N


COPYING -
NEUT.COM
ANEUT.COM
ACONF.COM
PIP.COM
FORMAT.COM
SNEUT.COM
PNEUT.COM

B>SNEUT

PROGRAM SNEUT OF 7-31-84  OAK RIDGE NATL LAB
PROGRAM NEUT OF 7-31-84  OAK RIDGE NATL LAB




MAIN MENU
ST--INITIAL CONFIGURATION
DD--DISPLAY DATA SAVED IN SUMMARY DATA BASE
EX--EXIT TO CP/M
AS--ASSAY IN STAND ALONE MODE
AD--ASSAY IN REMOTE COMPUTER MODE
RD--RETRIEVE DATA IN REMOTE COMPUTER MODE
ID--INITIALIZE SUMMARY DATA BASE
PB--ACQUIRE PASSIVE BACKGROUND
RC--RECALCULATE ASSAY FROM RAW DATA
SELECT OPTION?ST
```

```
FOLLOWING OPTIONS ARE IN EFFECT
PRINT OPTION--SH
ACTIVE ASSAYS--YE
PASSIVE ASSAYS--YE
PASSIVE BACKGROUND CORRECTIONS--YE
PASSIVE COUNT TIME(SEC)--       600.
SELECT OPTION TO MODIFY
PO--PRINT OPTION
AA--ACTIVE ASSAYS
PA--PASSIVE ASSAYS
PB--PASSIVE BACKGROUND CORRECTIONS
PC--PASSIVE COUNT TIME
NO--NONE
ENTER OPTION?PO

LO--LONG PRINT OUT
SH--SHORT PRINT OUT
NO--NO PRINT OUT
SELECT ASSAY PRINT OPTION?LO

FOLLOWING OPTIONS ARE IN EFFECT
PRINT OPTION--LO
ACTIVE ASSAYS--YE
PASSIVE ASSAYS--YE
PASSIVE BACKGROUND CORRECTIONS--YE
PASSIVE COUNT TIME(SEC)--       600.
SELECT OPTION TO MODIFY
PO--PRINT OPTION
AA--ACTIVE ASSAYS
PA--PASSIVE ASSAYS
PB--PASSIVE BACKGROUND CORRECTIONS
PC--PASSIVE COUNT TIME
NO--NONE
ENTER OPTION?AA

DO YOU WISH TO PERFORM ACTIVE ASSAYS?YES/NOYE

FOLLOWING OPTIONS ARE IN EFFECT
PRINT OPTION--LO
ACTIVE ASSAYS--YE
PASSIVE ASSAYS--YE
PASSIVE BACKGROUND CORRECTIONS--YE
PASSIVE COUNT TIME(SEC)--       600.
SELECT OPTION TO MODIFY
PO--PRINT OPTION
AA--ACTIVE ASSAYS
PA--PASSIVE ASSAYS
PB--PASSIVE BACKGROUND CORRECTIONS
PC--PASSIVE COUNT TIME
NO--NONE
ENTER OPTION?PA
```

```
DO YOU WISH TO PERFORM PASSIVE ASSAYS?YES/NOYE

FOLLOWING OPTIONS ARE IN EFFECT
PRINT OPTION--LO
ACTIVE ASSAYS--YE
PASSIVE ASSAYS--YE
PASSIVE BACKGROUND CORRECTIONS--YE
PASSIVE COUNT TIME(SEC)--      600.
SELECT OPTION TO MODIFY
PO--PRINT OPTION
AA--ACTIVE ASSAYS
PA--PASSIVE ASSAYS
PB--PASSIVE BACKGROUND CORRECTIONS
PC--PASSIVE COUNT TIME
NO--NONE
ENTER OPTION?PB

ARE PASSIVE ASSAYS TO BE MADE WITH A BACKGROUND
CORRECTION?YES/NO
NOTE THAT A YES ANSWER REQUIRES THAT A BACKGROUND
PREVIOUSLY TAKEN IN MODE PB EXIST IN A FILE
CALLED BACKGROU
ALSO NOTE THAT CALIBRATED MASS MEASUREMENTS ARE
MADE ONLY WHEN A BACKGROUND CORRECTION IS MADE
ENTER YOUR ANSWER?YES/NONO

FOLLOWING OPTIONS ARE IN EFFECT
PRINT OPTION--LO
ACTIVE ASSAYS--YE
PASSIVE ASSAYS--YE
PASSIVE BACKGROUND CORRECTIONS--NO
PASSIVE COUNT TIME(SEC)--      600.
SELECT OPTION TO MODIFY
PO--PRINT OPTION
AA--ACTIVE ASSAYS
PA--PASSIVE ASSAYS
PB--PASSIVE BACKGROUND CORRECTIONS
PC--PASSIVE COUNT TIME
NO--NONE
ENTER OPTION?PC

ENTER NEW PASSIVE COUNT TIME?(SEC)60
```

```
FOLLOWING OPTIONS ARE IN EFFECT
PRINT OPTION--LO
ACTIVE ASSAYS--YE
PASSIVE ASSAYS--YE
PASSIVE BACKGROUND CORRECTIONS--NO
PASSIVE COUNT TIME(SEC)--       60.
SELECT OPTION TO MODIFY
PO--PRINT OPTION
AA--ACTIVE ASSAYS
PA--PASSIVE ASSAYS
PB--PASSIVE BACKGROUND CORRECTIONS
PC--PASSIVE COUNT TIME
NO--NONE
ENTER OPTION?NO




MAIN MENU
ST--INITIAL CONFIGURATION
DD--DISPLAY DATA SAVED IN SUMMARY DATA BASE
EX--EXIT TO CP/M
AS--ASSAY IN STAND ALONE MODE
AD--ASSAY IN REMOTE COMPUTER MODE
RD--RETRIEVE DATA IN REMOTE COMPUTER MODE
ID--INITIALIZE SUMMARY DATA BASE
PB--ACQUIRE PASSIVE BACKGROUND
RC--RECALCULATE ASSAY FROM RAW DATA
SELECT OPTION?ID

DATA BASE BEING INITIALIZED




MAIN MENU
ST--INITIAL CONFIGURATION
DD--DISPLAY DATA SAVED IN SUMMARY DATA BASE
EX--EXIT TO CP/M
AS--ASSAY IN STAND ALONE MODE
AD--ASSAY IN REMOTE COMPUTER MODE
RD--RETRIEVE DATA IN REMOTE COMPUTER MODE
ID--INITIALIZE SUMMARY DATA BASE
PB--ACQUIRE PASSIVE BACKGROUND
RC--RECALCULATE ASSAY FROM RAW DATA
SELECT OPTION?AS

ENTER RUN NUMBER?[1,25]1

ENTER PRIMARY IDENTIFICATION?[15 CHAR]SNEUT TEST
```

```
ENTER SECONDARY IDENTIFICATION?[15 CHAR]PRINT OUT

ENTER CONTENT CODE?[10 CHAR]

ENTER CONTAINER WT IN KG?1

RUN NUMBER         1
PRIMARY ID  SNEUT TEST
SECONDARY ID  PRINT OUT
CONTENT CODE
CONTAINER WT(KG)        1.
IS ABOVE INFORMATION CORRECT?YES/NOYE

RUN NUMBER FROM NEUT        1
PROGRAM ANEUT OF 7-31-84--OAK RIDGE NATL LAB
CONFIGURATION FILE ANSETUP NOT FOUND.
DO YOU WISH TO CONFIGURE? YES/NO YE

PROGRAM ACONF OF 7-31-84--OAK RIDGE NATL LAB
THIS PROGRAM READS ANY PRESENT CONFIGURATION FILE
ANSETUP FOR PROGRAM ANEUT AND ALLOWS THE USER TO
CHANGE THE CONFIGURATION AT WILL.
NOTE THAT ANY PRESENT CONFIGURATION FILE WILL BE
DESTROYED UPON TERMINATION OF THIS PROGRAM.
THIS PROGRAM WILL DISPLAY DEFAULT VALUES FOR THE
PARAMETER QUESTIONS.  A CARRIAGE RETURN WILL RETAIN
THE DISPLAYED DEFAULT.
INPUT DATA IS CHECKED IN A CURSORY FASHION FOR
VALIDITY.  USE THIS PROGRAM CAUTIOUSLY.  YOU MAY
VIEW THE DATA IN FILE ANSETUP AFTER CREATION.
USE THIS PROGRAM TO DO THE VIEWING AND MAKING
ANY CHANGES.
DO YOU WISH TO CONTINUE? YES/NO YE

CRATE NUMBER?:         0

SLOT NUMBER OF SH TOT 3521?:         1

SLOT NUMBER OF FM 3521?:         2

SLOT NUMBER OF 14 MEV 3521?:         3

NUMBER OF PULSES FIRED BY NEUTRON GENERATOR?:
      2000

INTEGRATION LIMITS IN CHANNELS OF SHTOT
FISSILE SIGNAL?:        71,       470

INTEGRATION LIMITS IN CHANNELS OF SHTOT
BACKGROUND SIGNAL?:       571,       970
```

```
INTEGRATION LIMITS IN CHANNELS OF FM
SIGNAL?:        71,       470

INTEGRATION LIMITS IN CHANNELS OF
14 MEV MONITOR?:        10,        20  71,470

LIMITS IN CHANNELS OF LEAST SQUARES CURVE
FIT TO FM SIGNAL?:       150,       300

ENTER PROPORTIONALITY CONSTANT AND OFFSET
RESPECTIVELY FOR CALIBRATION?:    5000.0,      13.0

ENTER LABEL IDENTIFYING CALIBRATION ISOTOPE?
U-235   :

SAVE SHTOT DATA IN DISK FILES? YE/NO:YE

SAVE FM DATA IN DISK FILES? YE/NO:YE

SAVE 14 MEV DATA IN DISK FILES? YE/NO:YE

LABEL?[20 CHAR]:


CONFIGURATION IS COMPLETE
DO YOU WISH TO CHANGE THIS CONFIGURATION? YE/NO  YE

PROGRAM ACONF OF 7-31-84--OAK RIDGE NATL LAB
THIS PROGRAM READS ANY PRESENT CONFIGURATION FILE
ANSETUP FOR PROGRAM ANEUT AND ALLOWS THE USER TO
CHANGE THE CONFIGURATION AT WILL.
NOTE THAT ANY PRESENT CONFIGURATION FILE WILL BE
DESTROYED UPON TERMINATION OF THIS PROGRAM.
THIS PROGRAM WILL DISPLAY DEFAULT VALUES FOR THE
PARAMETER QUESTIONS.  A CARRIAGE RETURN WILL RETAIN
THE DISPLAYED DEFAULT.
INPUT DATA IS CHECKED IN A CURSORY FASHION FOR
VALIDITY.  USE THIS PROGRAM CAUTIOUSLY.  YOU MAY
VIEW THE DATA IN FILE ANSETUP AFTER CREATION.
USE THIS PROGRAM TO DO THE VIEWING AND MAKING
ANY CHANGES.
DO YOU WISH TO CONTINUE? YES/NO YE

CRATE NUMBER?:         0

SLOT NUMBER OF SH TOT 3521?:         1

SLOT NUMBER OF FM 3521?:         2

SLOT NUMBER OF 14 MEV 3521?:         3
```

```
NUMBER OF PULSES FIRED BY NEUTRON GENERATOR?:
      2000

INTEGRATION LIMITS IN CHANNELS OF SHTOT
FISSILE SIGNAL?:        71,       470

INTEGRATION LIMITS IN CHANNELS OF SHTOT
BACKGROUND SIGNAL?:       571,       970

INTEGRATION LIMITS IN CHANNELS OF FM
SIGNAL?:        71,       470

INTEGRATION LIMITS IN CHANNELS OF
14 MEV MONITOR?:        71,       470

LIMITS IN CHANNELS OF LEAST SQUARES CURVE
FIT TO FM SIGNAL?:       150,       300

ENTER PROPORTIONALITY CONSTANT AND OFFSET
RESPECTIVELY FOR CALIBRATION?:    5000.0,      13.0

ENTER LABEL IDENTIFYING CALIBRATION ISOTOPE?
U-235   :

SAVE SHTOT DATA IN DISK FILES? YE/NO:YE

SAVE FM DATA IN DISK FILES? YE/NO:YE

SAVE 14 MEV DATA IN DISK FILES? YE/NO:YE

LABEL?[20 CHAR]:


CONFIGURATION IS COMPLETE
DO YOU WISH TO CHANGE THIS CONFIGURATION? YE/NO  NO

DO YOU WISH TO EXECUTE ANEUT? YES/NO  YE

RUN NUMBER FROM NEUT        1
PROGRAM ANEUT OF 7-31-84--OAK RIDGE NATL LAB
RUN          1     DRUM SNEUT TE     15:54: 2     8/28/84
FIRE  2000 PULSES FROM NEUTRON GENERATOR WHEN READY
ACQUISITION WILL CONTINUE FOR SPECIFIED NUMBER OF PULSES
OR UNTIL STOP BUTTON IS PRESSED
ACQUISITION ENDED--SCAN LIMIT--       2000. PULSES READ
```

```
ACTIVE NEUTRON INTERROGATION REPORT


RUN      1   DRUM SNEUT TE   15:54: 2     8/28/84

RAW DATA
SHIELDED TOTALS(  71, 470)        121.
SHIELDED TOTALS( 571, 970)         34.
FLUX MONITOR(  71, 470)      52327.
2ND FLUX MONITOR(  71, 470)      58044.

CURVE FITTING RESULTS
   151 POINTS WERE FITTED TO      1290.5EXP(     -.001090*T)
DWELL TIME         10.0 MICROSECONDS/CHANNEL
CHI SQUARE        131.1  WITH   148  DEGREES OF FREEEDOM
CHI SQUARE/DEGREE          .89
MAX RESIDUAL        2.818  AT TIME       1505.  MICROSECONDS
DIE AWAY TIME       917.57  MICROSECONDS
HALF LIFE       635.88  MICROSECONDS
U-235    FISSILE EQUIVALENT (CORRECTED)         -9.  MILLIGRAMS
```

```
ACTIVE NEUTRON INTERROGATION REPORT


RUN      1   DRUM SNEUT TE   15:54: 2     8/28/84

RAW DATA
SHIELDED TOTALS(  71, 470)        121.
SHIELDED TOTALS( 571, 970)         34.
FLUX MONITOR(  71, 470)      52327.
2ND FLUX MONITOR(  71, 470)      58044.

CURVE FITTING RESULTS
   151 POINTS WERE FITTED TO      1290.5EXP(    -.001090*T)
DWELL TIME        10.0 MICROSECONDS/CHANNEL
CHI SQUARE       131.1  WITH   148  DEGREES OF FREEEDOM
CHI SQUARE/DEGREE         .89
MAX RESIDUAL       2.818  AT TIME       1505.  MICROSECONDS
DIE AWAY TIME      917.57  MICROSECONDS
HALF LIFE      635.88  MICROSECONDS
U-235    FISSILE EQUIVALENT (CORRECTED)         -9.  MILLIGRAMS

SHIELDED TOTALS DATA SAVED IN SHTOT   01
FLUX MONITOR DATA SAVED IN FM      01
2ND FM DATA SAVED IN 14MEV   01
PROGRAM NEUT OF 7-31-84  OAK RIDGE NATL LAB
PROGRAM PNEUT OF 7-31-84
OAK RIDGE NATL LAB VERSION
RUN          1    DRUM  SNEUT TE    15:56:29     8/28/84


COUNT STARTED
COUNT ENDED--END OF TIME
```

```
PNEUT OF 7-31-84--OAK RIDGE NATL LAB
RUN     1   DRUM SNEUT TE    15:56:29    8/28/84
GATE CORRECTION FACTORS  1.000000 (70 USEC GATES)  1.000000 (250 USEC GATES)
FOLLOWING DATA IS NOT BACKGROUND CORRECTED

COUNTING TIME IS       64.14 SECONDS

     DETECTOR     COUNT      RATE          DETECTOR     COUNT      RATE

BARE DOOR 500      211.      3.29     SHLD DOOR 501        31.        .48
BARE RGHT 502      197.      3.07     SHLD RGHT 503        21.        .33
BARE BACK 504      138.      2.15     SHLD BACK 505        29.        .45
BARE LEFT 506      205.      3.20     SHLD LEFT 507        34.        .53
BARE TOP  508      176.      2.74     SHLD TOP  509        26.        .41
BARE BOTM            0.      0.00     SHLD BOTM 511        47.        .73
FLUX MONITOR         0.      0.00     14 MEV MONITO         1.        .02
SYSTEM TOTALS RATE       17.38     SHIELDED TOTALS RATE        2.93  (FROM PARTS)
NEUTRON COINCIDENCE

SHIELDED TOTALS                    188.+/-       13.71
SYSTEM TOTALS                     1115.+/-       33.39
1ST N 250 USEC GATES              1098.
1ST N 70 USEC GATES                186.
RANDOM 70 USEC GATES            641361.
RANDOM 250 USEC GATES            64136.
1ST N GATED 70 USEC TOTALS           2.
RANDOM GATED 70 USEC TOTALS        130.
1ST N GATED 250 USEC TOTALS         17.
RANDOM GATED 250 USEC TOTALS       285.

RANDOM COINCIDENT NEUTRONS/250 USEC GATE    .44437E-02
RANDOM COINCIDENT NEUTRONS/70 USEC GATE   .20269E-03

250 USEC GATE LIVE TIME          63.86 SEC
70 USEC GATE LIVE TIME        64.12 SEC
NET COINCIDENT NEUTRONS/250 USEC GATE   .11039E-01+/-   .37643E-02
NET COINCIDENT NEUTRONS/70 USEC GATE   .10550E-01+/-   .76033E-02

SYSTEM TOTALS RATE     17.385    +/-   .52064
SHIELDED TOTALS RATE   2.9313    +/-   .21378

NET COINCIDENT 250 USEC GATE NEUTRONS/LIVE TIME   .18981    +/-   .64724E-01
NET COINCIDENT 70 USEC GATE NEUTRONS/LIVE TIME   .30602E-01+/-   .22055E-01



REDUCED VARIANCE
Y=   .15184E-01    Q=   .30776E-05
MASS FROM COINCIDENCE COUNTING
USING 70 USEC GATE          0.00 GRAMS
USING 250 USEC GATE          0.00 GRAMS
USING RED VAR Q          0.00 GRAMS
SELECTED BY CONTENT CODE AND COUNT RATE          0.00 GRAMS
```

```
PNEUT OF 7-31-84--OAK RIDGE NATL LAB
RUN     1   DRUM SNEUT TE    15:56:29    8/28/84
GATE CORRECTION FACTORS  1.000000 (70 USEC GATES)  1.000000 (250 USEC GATES)
FOLLOWING DATA IS NOT BACKGROUND CORRECTED

COUNTING TIME IS       64.14 SECONDS

     DETECTOR      COUNT       RATE          DETECTOR      COUNT       RATE

BARE DOOR 500      211.        3.29     SHLD DOOR 501       31.         .48
BARE RGHT 502      197.        3.07     SHLD RGHT 503       21.         .33
BARE BACK 504      138.        2.15     SHLD BACK 505       29.         .45
BARE LEFT 506      205.        3.20     SHLD LEFT 507       34.         .53
BARE TOP  508      176.        2.74     SHLD TOP  509       26.         .41
BARE BOTM            0.        0.00     SHLD BOTM 511       47.         .73
FLUX MONITOR         0.        0.00     14 MEV MONITO        1.         .02
SYSTEM TOTALS RATE        17.38      SHIELDED TOTALS RATE         2.93  (FROM PARTS)
NEUTRON COINCIDENCE

SHIELDED TOTALS                        188.+/-       13.71
SYSTEM TOTALS                         1115.+/-       33.39
1ST N 250 USEC GATES                  1098.
1ST N 70 USEC GATES                    186.
RANDOM 70 USEC GATES                641361.
RANDOM 250 USEC GATES                64136.
1ST N GATED 70 USEC TOTALS               2.
RANDOM GATED 70 USEC TOTALS            130.
1ST N GATED 250 USEC TOTALS             17.
RANDOM GATED 250 USEC TOTALS           285.

RANDOM COINCIDENT NEUTRONS/250 USEC GATE    .44437E-02
RANDOM COINCIDENT NEUTRONS/70 USEC GATE   .20269E-03

250 USEC GATE LIVE TIME          63.86 SEC
70 USEC GATE LIVE TIME        64.12 SEC
NET COINCIDENT NEUTRONS/250 USEC GATE   .11039E-01+/-   .37643E-02
NET COINCIDENT NEUTRONS/70 USEC GATE   .10550E-01+/-   .76033E-02

SYSTEM TOTALS RATE     17.385    +/-   .52064
SHIELDED TOTALS RATE   2.9313    +/-   .21378

NET COINCIDENT 250 USEC GATE NEUTRONS/LIVE TIME   .18981    +/-   .64724E-01
NET COINCIDENT 70 USEC GATE NEUTRONS/LIVE TIME   .30602E-01+/-   .22055E-01


REDUCED VARIANCE
Y=   .15184E-01    Q=   .30776E-05
MASS FROM COINCIDENCE COUNTING
USING 70 USEC GATE          0.00 GRAMS
USING 250 USEC GATE          0.00 GRAMS
USING RED VAR Q         0.00 GRAMS
SELECTED BY CONTENT CODE AND COUNT RATE          0.00 GRAMS
```

```
MULTIPLICITY REPORT
PROGRAM PNEUT OF 7-31-84--OAK RIDGE NATL LAB
RUN     1   DRUM SNEUT TE   15:56:29    8/28/84


RAW MULTIPLICITIES
1ST NEUTRON 70 USEC GATE
        184.          2.           0.           0.           0.           0.
RANDOM  NEUTRON 70 USEC
      641233.        128.          1.           0.           0.           0.
          0.          0.           0.           0.           0.           0.
          0.          0.           0.
1ST NEUTRON 250 USEC GAT
        1081.         17.          0.           0.           0.           0.
RANDOM NEUTRON 250 USEC
       63858.        271.          7.           0.           0.           0.
PILEUP CORRECTED MULTIPL
70 USEC GATES
        184.+/-      13.567         2.0+/-       1.415         -.0+/-       .000
         -.0+/-       .00*          .*+/-        .00*         0.0+/-      0.000
250 USEC GATES
       1086.+/-      33.023        12.5+/-       4.153         -.2+/-       .049
         -.0+/-       .001          .0+/-        .00*         0.0+/-      0.000
LIVE TIME(S)                  64.1       63.9


70 USEC GATES
BACKGROUND CORRECTIONS
NO BACKGROUND                             COUNTING TIME       1.
GATE LENGTH    70000.  GATE DELAY
     2000. 1ST N GATES        0.  RAN N GATES        0. LIVE TIME       0.
MULTIPLICITY RATIOS
   93.721     +/-   67.864      1.00000     +/-   1.0187       -.34577E-03+/-  -.24907E-03
  -.14905E-05+/-  -.10736E-05   .83676E-09+/-   .95148E-09   0.0000     +/-   0.0000
MULTIPLICITY RATES
   2.8701     +/-   .21158      .30623E-01+/-   .22059E-01  -.10589E-04+/-   .63106E-05
  -.45643E-07+/-   .58871E-07   .25624E-10+/-   .22545E-10   0.0000     +/-   0.0000


250 USEC GATES
BACKGROUND CORRECTIONS
NO BACKGROUND                             COUNTING TIME       1.
GATE LENGTH   250000.  GATE DELAY
     2000. 1ST N GATES        0.  RAN N GATES        0. LIVE TIME       0.
MULTIPLICITY RATIOS
   87.090     +/-   29.132      1.00000     +/-   .47111       -.13790E-01+/-  -.45939E-02
  -.51095E-04+/-  -.17021E-04   .17285E-05+/-   .94897E-06   0.0000     +/-   0.0000
MULTIPLICITY RATES
   17.001     +/-   .51710      .19521      +/-   .65029E-01  -.26920E-02+/-   .76030E-03
  -.99742E-05+/-   .11275E-04   .33743E-06+/-   .14725E-06   0.0000     +/-   0.0000
```

```
RAW PASSIVE NEUTRON DATA SAVED IN FILE PASSIVE .01
PROGRAM NEUT OF 7-31-84  OAK RIDGE NATL LAB
ASSAY SUMMARY REPORT
PRIMARY ID     SNEUT TEST        SECONDARY ID   PRINT OUT
RUN NUMBER                   1
TIME/DATE OF ACTIVE   15:54: 2   8/28/84
TIME/DATE OF PASSIVE  15:56:29   8/28/84
PAS FILE NAME       PASSIVE 01
ACT FILE NAMES      SHTOT    01   FM       01    14MEV    01
CONTENT CODE                      CONTAINER WT(KG)             1.
FISSIL MASS(MG)     -.90587E+01   G TRU ACT(NCI/G)    -.10000E+01
T POW DEN(W/CFT)    -.10000E+01   T FISS M(G)         -.10000E+01
T ALPHA ACT(CI)     -.10000E+01   T THER POW(W)       -.10000E+01
SYSTEM TOT RATE      .17385E+02   SHIELD TOT RATE      .29313E+01
SH ACT SIG           .12100E+03   SH ACT BKR           .34000E+02
FM SIG               .52327E+05   THALF(US)            .63588E+03
AZERO                .12905E+04   2ND FM SIG           .58044E+05
P COUNT TIME(S)      .64136E+02   NO ACT PULSES              2000
250 CO RATE          .18981E+00   70 CO RATE           .30602E-01
RED VAR Q            .30776E-05   RED VAR Y            .15184E-01
CHISQ/DEG            .88550E+00
```

```
ASSAY SUMMARY REPORT
PRIMARY ID     SNEUT TEST        SECONDARY ID   PRINT OUT
RUN NUMBER                   1
TIME/DATE OF ACTIVE   15:54: 2   8/28/84
TIME/DATE OF PASSIVE  15:56:29   8/28/84
PAS FILE NAME       PASSIVE 01
ACT FILE NAMES      SHTOT    01   FM       01    14MEV    01
CONTENT CODE                      CONTAINER WT(KG)             1.
FISSIL MASS(MG)     -.90587E+01   G TRU ACT(NCI/G)    -.10000E+01
T POW DEN(W/CFT)    -.10000E+01   T FISS M(G)         -.10000E+01
T ALPHA ACT(CI)     -.10000E+01   T THER POW(W)       -.10000E+01
SYSTEM TOT RATE      .17385E+02   SHIELD TOT RATE      .29313E+01
SH ACT SIG           .12100E+03   SH ACT BKR           .34000E+02
FM SIG               .52327E+05   THALF(US)            .63588E+03
AZERO                .12905E+04   2ND FM SIG           .58044E+05
P COUNT TIME(S)      .64136E+02   NO ACT PULSES              2000
250 CO RATE          .18981E+00   70 CO RATE           .30602E-01
RED VAR Q            .30776E-05   RED VAR Y            .15184E-01
CHISQ/DEG            .88550E+00
```

```
DO YOU WISH TO PERFORM ANOTHER ASSAY?YES/NONO




MAIN MENU
ST--INITIAL CONFIGURATION
DD--DISPLAY DATA SAVED IN SUMMARY DATA BASE
EX--EXIT TO CP/M
AS--ASSAY IN STAND ALONE MODE
AD--ASSAY IN REMOTE COMPUTER MODE
RD--RETRIEVE DATA IN REMOTE COMPUTER MODE
ID--INITIALIZE SUMMARY DATA BASE
PB--ACQUIRE PASSIVE BACKGROUND
RC--RECALCULATE ASSAY FROM RAW DATA
SELECT OPTION?EX
 STOP
```

APPENDIX C

TRU Waste Drum Assay Facility
Assay Report

ORNL TRU WASTE DRUM ASSAY FACILITY (TWDAF)
BUILDING 7824

Assay Report page 1

Drum ATN- 2396

Batch No. 021684

Weight of drum 67 kg

Report date 9/19/84

Building of origin (waste generator) 3038

Segmented Gamma Scanner (SGS) Data — Date of Assay: 2/24/84

SGS gamma-ray data indicate presence of the following radionuclides:

^{239}Pu	x	Pu K X rays	x	^{249}Cf	
^{235}U	x	^{137}Cs	x	^{144}Ce	
^{233}U		^{228}Th		^{154}Eu	
^{241}Pu/^{237}U	x	^{231}Pa	x	^{60}Co	
^{243}Am/^{239}Np	x	^{227}Ac		^{106}Ru/^{106}Rh	
^{238}Pu		^{211}Pb		^{110m}Ag	
^{241}Am	x	^{250}Bk		^{125}Sb	
				^{134}Cs	

Comments:

Fissile: ^{239}Pu and ^{235}U

Presence of ^{235}U gamma indicates significant quantity is present (>5 g)

Neutron Assay System (NAS) Data — Date of Assay: 2/24/84

1. Active NAS Scan

Total corrected fissile mass content: 3,400 mg ^{239}Pu equivalent

Comments: matrix correction factor: 1.54

Disagreement with Item 3's information is probably due to thermal neutron skin effect. Plutonium and uranium are probably in source form; and, consequently, majority would not be "seen" by interrogating flux.

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BUILDING 7824

Assay Report

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2. Passive NAS Scan:

Total net passive neutron source strength: 9.94×10^3 n/s

SHTOT/SYSTOT	0.267
SYSTOT/SYSCOINC	10.9
SHTOT/SHCOINC	-
P3(SYS)/P2(SYS)	0.071 ± 0.010
P1(SYS/P2(SYS)	10.395 ± 0.190
P3(SH)/P2(SH)	-
P1(SH)/P2(SH)	-

Note: Mark NA if data are unavailable or unusable.

Comments: P3/P2 ratio indicates presence of ^{244}Cm.
P1/P2 ratio indicates presence of ^{240}Pu and ^{244}Cm mixture.
SYSTOT/SYSCOINC ratio indicates no excess neutrons from (α,n) reactions.

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3. Elemental and/or isotopic content of waste container, if available from sources other than NAS and SGS:

Note: If not available, check block and continue to Item 4 ☐

Element	Element wt (g)	Wt% Isotope	Isotope Wt
Pu	6	11.5% ^{240}Pu	5 g ^{239}Pu
U	26	93.3% ^{235}U	24 g ^{235}U

Source of information: ORNL Nuclear Materials Intra-Laboratory Transfer 1/23/84, Transaction No. 7278.

Comments: Large amounts of plutonium and uranium suggest discarded sources; thermal neutron skin effect will play a role.

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4. Interpretation of Data and Results:

For 6 g of 11.5% ^{240}Pu (or 0.7 g ^{240}Pu), one would expect a SYSCOINC rate of 8.8 cps. From the data, SYSCOINC = 127.7 cps. Therefore, 93% of the spontaneous fissions can be attributed to some isotope(s) other than ^{240}Pu. The neutron multiplicity ratio P3/P2 is very close to that obtained for a pure ^{244}Cm source. Also, the neutron multiplicity ratio P1/P2 indicates the presence of a mixture of ^{240}Pu and ^{244}Cm. Consequently, the passive neutron source signal is dominated by ^{244}Cm. From calibration data, the net passive neutron signal, after accounting for the small ^{240}Pu contribution, indicates the presence of 1 mg of ^{244}Cm.

Wt. Isotope	Specific Activity (Ci/g)	nCi/g
3,400 mg ^{239}Pu	6.2094×10^{-2}	3.15×10^{3}
700 mg ^{240}Pu	2.2806×10^{-1}	2.38×10^{3}
(non-TRU) 1 mg ^{244}Cm	8.1017×10^{1}	1.21×10^{3}

Total TRU content: 5,500 ± 1,100 nCi/g
Drum category: TRU
Color code: Red

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