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Idaho
National
Engineering
Laboratory

INEL Test Plan for Evaluating Waste Assay Systems

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EXECUTIVE SUMMARY

A test plan has been developed that provides a systematic approach for evaluating the capability of radioassay systems, such as those developed by commercial firms or the U.S. Department of Energy Environmental Management Office of Technology Development, for measuring transuranic (TRU) contaminated wastes.

A test bed is being established at the Idaho National Engineering Laboratory (INEL) Radioactive Waste Management Complex (RWMC). These tests are currently focused on mobile or portable radioassay systems. Prior to disposal of TRU waste at the Waste Isolation Pilot Plant (WIPP), radioassay measurements must meet the quality assurance objectives of the TRU Waste Characterization Quality Assurance Program Plan. This test plan provides technology holders with the opportunity to assess radioassay system performance through a three-tiered test program that consists of: (a) evaluations using non-interfering matrices, (b) surrogate drums with contents that resemble the attributes of INEL-specific waste forms, and (c) real waste tests. Qualified sources containing a known mixture and range of radionuclides will be used for the non-interfering and surrogate waste tests. The results of these tests will provide technology holders with information concerning radioassay system performance and provide the INEL with data useful for making decisions concerning alternative or improved radioassay systems that could support disposal of waste at WIPP.

Because of limited resources, technology holders are required to submit information concerning the radioassay system description, calibration method, data acquisition/reduction methods, means of determining total uncertainty, and applicability to measuring TRU-contaminated wastes. This information will be evaluated as part of pre-qualifying technology holders for demonstrating their systems at the INEL. Additionally, the technology holder must provide sufficient information to allow assessment of compliance with the existing safety and environment envelop for operations at the RWMC.

Each test period is expected to last four to six weeks. Following radioassay system set-up, non-interfering matrix tests will be completed to verify system calibration and capability for TRU waste measurements. Surrogate drum tests covering a majority of the large population of INEL TRU wastes (sludge, glass, combustibles, metals, graphite, and firebrick) in accessible storage will be completed using sources containing known amounts of Pu, Am, U, and (α,N) material. Finally, measurements of real TRU waste forms will be completed. Waste forms evaluated include those performed as part of the surrogate tests.

Results of the measurements will be reported by the technology holder to INEL staff for further evaluation. A report concerning the evaluation results will only be provided to the technology holder. The intent of this program is not to "certify' or "qualify" a particular radioassay system, but rather provide a systematic plan for testing performance. The results of these tests will be beneficial to evaluation and potential selection of alternative or improved assay methods to support INEL goals for waste disposal at WIPP.

CONTENTS

TABLES

ACRONYMS

AI	absorber index
ALARA	as low as reasonably achievable
ANSI	American National Standards Institute
CAA	Clean Air Act
CH	contact-handled
CH-TRU	contact-handled transuranic
CY	calendar year
DOE	Department of Energy
EDF	Engineering Design File
EM-50	Deputy Assistant Secretary, DOE Office of Technology Development
FGE	fissile gram equivalent
FRP	fiberglass-reinforced resin
IDC	item description code
IMWI	Idaho Mixed Waste Information
INEL	Idaho National Engineering Laboratory
LANL	Los Alamos National Laboratory
LLW	low-level waste
LWH	length-width-height
MDC	minimum detectable concentration
NAD	nuclear accident dosimeter
NDA	nondestructive assay
NEC	National Electric Code
NEPA	National Environmental Policy Act

OSHA	Occupational Safety and Health Administration
PAN	passive active neutron
PDP	Performance Demonstration Program
PE-Ci	plutonium equivalent curies
QA	quality assurance
QAO	quality assurance objective
QAPP	Quality Assurance Program Plan
RCRA	Resource Conservation and Recovery Act
RFP	Rocky Flats Plant
%RSD	percent relative standard deviation
RWMC	Radioactive Waste Management Complex
SDD	system design description
SWEPP	Stored Waste Examination Pilot Plant
TRU	transuranic
TRUPACT-II	Transuranic Package Transporter-II
TSA	Transuranic Storage Area
USQ	Unanswered safety question
VAX/GAP	a gamma-ray analysis program for a VAX computer
WAC	Waste Acceptance Criteria
WG	weapons grade
WIPP	Waste Isolation Pilot Plant

INEL Test Plan for Evaluating Waste Assay Systems

1. INTRODUCTION

1.1 Purpose

The Idaho National Engineering Laboratory (INEL) is providing a test bed where systems for the nondestructive assay (NDA) of transuranic (TRU) waste can be tested and evaluated. The unique feature of the INEL TRU assay waste test bed is that tests will be performed using a wide variety of real TRU waste forms of the type of interest to the INEL. Hence, a system's applicability to the assay of actual site-specific waste forms of interest to the INEL and of the type which must be assayed prior to shipment to the Waste Isolation Pilot Plant (WIPP) can be determined, and the ability of the system to meet specific quality assurance objectives (QAOs) in the TRU Waste Characterization Quality Assurance Program Plan (QAPP)[1] can be ascertained.

Until now, technology holders have usually relied on drum mockups of which bore little resemblance to actual waste forms for testing and evaluation purposes. When actual waste forms were available, tests and evaluations were not always performed in an unbiased fashion. For example, some assay systems were tested and evaluated using the same waste containers which were used to calibrate these systems. In addition, since the same waste forms have not been available to all technology holders, unbiased comparisons of the various assay systems have not been performed.

The INEL TRU waste assay test bed is available to all technology holders (commercial vendors, national laboratories, and others) who have developed assay systems for TRU waste. It will provide them the opportunity to test their systems, evaluate applicability to assay of INEL TRU waste forms, verify system performance characteristics (e.g., accuracy, sensitivity, total uncertainty, lower limit of detection), and demonstrate their systems' capabilities using real TRU waste forms. Although the test bed is primarily designed for the testing and evaluation of mobile or portable systems, it can accommodate stationary systems which can be brought in and operated on a flatbed trailer. Other stationary systems can be tested and evaluated at the technology holder's site (i.e., INEL personnel can bring surrogate drums to a technology holder's site and oversee the tests and evaluations), but this will be handled on a case-by-case basis.

1.2 Objectives of This Test Plan

The main objective of this test plan is to present the methodology that is used to evaluate TRU waste assay systems. This includes identification of the surrogate and waste form tests that will be used, the requirements for a candidate technology being selected to participate in an evaluation using the test bed, descriptions of the various waste forms at the INEL, and operational requirements. In addition, the method that will be used for evaluation of test results is outlined.

Section 1 gives background and general information about the test bed, the waste forms at the INEL, and some of the assay problems that the INEL has encountered. Section 2 addresses the methodology used to select a candidate for participation in a test bed evaluation. An overview of radioassay requirements, including quality assurance objectives, comprises Section 3. Section 4 describes the various waste forms at the INEL. The evaluation plan is discussed in Section 5, and operational constraints are discussed in Section 6. Section 7 outlines the methodology that the INEL will use to evaluate the test results.

1.3 General Conditions and Limitations of the Test Bed

The INEL TRU waste assay test bed will accept applications from potential participants beginning June 1, 1996, and the test bed will be available to begin assessments of participant's assay systems during the fourth quarter of FY-1996. Scheduling for the test bed will be dependent on RWMC operational support and availability of resources. It is expected that the basic evaluation time period will extend up to 4–6 weeks, with the first 1 to 1-1/2 weeks being allotted for system setup and calibration checkout. In addition, an optional evaluation period of 2–4 weeks may be available to a participant, depending upon how well his assay system performs during the basic evaluation.

Evaluations of assay systems will be performed using surrogates prepared by the INEL and TRU contaminated waste drums which have been characterized to some extent by the INEL. Each participant is required to assay first a set of noninterfering-matrix drums, then a basic set of surrogate drums, and finally a basic set of TRU contaminated waste drums. An optional set made up of surrogates, real TRU waste drums, and overpacked drums will be available to the participants if the INEL determines that more useful information can be obtained by using them and the participants desire to make use of them. Waste container size and type will be limited to 55-gallon waste drums and 55-gallon waste drums overpacked in 83-gallon drums only; no waste boxes or 83-gallon waste drums will be part of the evaluation program. Plutonium content will range from below 100 nCi/g to approximately 200 g in a drum.

The basic sets of surrogates and waste forms will encompass only the major population waste form categories of the accessible stored waste drums (see Sections 5.5 and 5.6 for details on the drums in the basic sets); low population waste forms will not be included in these basic sets. The optional set, however, may include some of the lower population waste categories. Assay of this optional set (or parts of it) is strictly up to the discretion of the participants and must be specifically invited by the INEL.

1.4 Background

In order to be allowed to ship TRU waste from a Department of Energy (DOE) facility to a storage location, the contents of the waste containers must be characterized (plutonium content, total α-activity, and isotopics) to ensure that transportation, storage, and/or disposal requirements are met. For example, before TRU waste is shipped from the INEL Radioactive Waste Management Complex (RWMC) to WIPP for storage, each container must undergo nondestructive examination and assay to ensure that all relevant transportation, storage, and safety requirements are met. The current assay systems may have some limitations in meeting these requirements for certain categories of waste containers and waste contents. Therefore, new or improved assay systems are being developed by both commercial and DOE organizations. As these new systems approach maturity, there is a desire by DOE and the INEL to have them tested, each system's capabilities evaluated, their applicability to assay of site-specific waste determined, and the attributes of the various systems compared. The

INEL TRU waste assay test bed provides a systematic approach for testing, evaluating, and comparing these assay systems using real TRU contaminated waste forms of the type of interest to the INEL.

The INEL is an ideal site for a test bed to evaluate TRU waste assay systems for a number of reasons. The INEL has a large amount of real waste that can be used in an evaluation of assay systems. The INEL has a mature radioassay program, has had vast experience in examining TRU waste (including first-hand knowledge of many of the problems encountered when attempting to assay real TRU waste forms), and has completed the only analysis for evaluating radioassay to QAPP requirements. Additionally, the INEL has a number of surrogate drums which were carefully designed to simulate the major waste forms at the INEL and into which qualified source materials can be placed in a multitude of configurations. Furthermore, the INEL is expected to be the first to ship waste to WIPP when WIPP becomes operational, and, in order to meet the settlement agreement, the INEL will be one of the most active users of these assay systems. Finally, alternative assay systems may be needed to support production operations to achieve settlement agreement milestones.

1.5 General Description of INEL TRU Waste and History of TRU Waste at the INEL

Over 90% of the contact handled TRU (CH-TRU) waste stored within the RWMC Transuranic Storage Area (TSA) was generated by Rocky Flats Plant (RFP) operations. The remaining non-RFP TRU wastes were generated by other DOE facilities. These generators include: Mound Laboratories, Argonne National Laboratory-East, Bettis Atomic Power Laboratory, and Battelle Columbus Laboratory. Only a limited amount of TRU waste has been generated by INEL operations. Approximately 65,000 cubic meters (2.3 million cubic feet) of CH-TRU waste is stored at the RWMC on above-ground asphalt pads covered with an earthen berm, inside air support buildings and inside RCRA compliant storage modules. The waste is contained in approximately 130,000 drums and 11,000 boxes. All drums are fabricated from steel, but the boxes are fabricated from steel, plywood, and fiberglass-reinforced polyester (FRP) coated plywood.

Approximately 98.5% of the drum containers are 55-gallon drums and 1.4% of the drum containers are 83-gallon drums. The remaining 0.1% are other size drums such as 30- and 100-gallon drums. However, if a drum deteriorates to a point where it poses an operational/storage concern, it is overpacked. Therefore, the percentage of 83-gallon drums will increase (possibly up to 10%) as the deteriorated 55-gallon drums are overpacked into 83-gallon drums.

CH-TRU waste is stored in both metal and wooden boxes. A variety of sizes of metal and wooden boxes were used. However, approximately 73% of the box containers are plywood and FRP coated plywood with approximate size of 7 × 4 × 4 ft [length-width-height(LWH)]. Approximately 20% of the box containers are metal Type I SANDBOXES (approximate outside dimensions of 7 × 4 × 4 ft (LWH)) and approximately 5% of the boxed waste is stored in metal M-III bins with approximate outside dimensions of 50.38 × 58.38 × 72.38 in. (LWH). The remaining 2% of the box containers are metal and wooden with a variety of sizes.

From 1970 to 1972, waste drums were prepared by lining them with one or two polyethylene drum bags. Cardboard liners might have been used to line the inner drum bag. After being filled with waste, each drum bag was sealed with tape. Boxes, made of plywood, were lined with a polyethylene box bag, and the bag was then lined with a cardboard liner. After the box was filled with waste, the bag ends and sides were folded toward the box center and tape was used to seal the folded edges.

The lining of the drums with 90-mil rigid polyethylene liners began in 1972. The drums were prepared by placing the rigid liner in the drum and then lining that with one or two polyethylene drum bags. After being filled with waste, the bags were sealed with tape, and the liner lid was sealed onto the liner. The coating of plywood boxes with FRP also began in 1972. Each box was lined with a polyethylene box bag and then with a cardboard liner. After the box was filled with waste, the bag ends and sides were folded toward the box center and tape was used to seal the folded edges.

Waste was generally packaged in a similar manner. Large or bulky items were placed directly into prepared drums or boxes. Smaller items were separately placed into plastic bags, polyethylene or glass bottles, cardboard cartons, and/or metal cans.

Each waste package stored at the TSA is assigned an item description code (IDC) that reflects the general contents of the container and identifies the generator and the process areas where the waste was generated. IDCs of similar waste characteristics were grouped together into matrix category groups. The results of this grouping for 11 waste categories representing the INEL stored CH-TRU contaminated wastes are summarized in Table 1-1. Initial characterization efforts, however, will be focused on RFP-generated wastes that are stored in drums and are in accessible inventory (i.e., wastes stored in the air support building and RCRA-compliant storage modules). Summary information by the 11 waste categories for the RFP-generated CH-TRU waste in accessible inventory is presented in Table 1-2. Details ont he IDCs contained in each of the 11 matrix category groups can be found in Reference 2. More detailed information on the waste by IDCs can be found in References 3 and 4.

There are few waste packages that can be described as completely homogeneous. However, the major constituents found in containers of the same IDC are essentially constant. Generally, each waste package includes the major waste form identified in the IDC description, packaging materials, and absorbent.

Based on what is known about the filling process, radial homogeneity would be expected for sludge drums. In order to test this hypothesis, the sludge waste categories are currently being investigated for radial homogeneity. Preliminary results based on destructive analysis of three cores from each of two elevations from each of five content code 001 and 007 waste drums indicate that there is radial homogeneity within 25% for these drums.

The unknown/unrecorded waste category primarily represents waste retrieved from shallow land burial during INEL Early Waste Retrieval and Initial Drum Retrieval projects conducted between 1974 and 1978. The uncategorized material represents waste placed in storage at the TSA prior to 1973. Specific content code information for each container was not recorded. However, the majority of these wastes were generated by RFP operations and are expected to be similar to the other wastes stored at the TSA.

Table 1-1. INEL Stored TRU Contaminated Waste Inventory.

Matrix Category Group	Drums Quantity	Drums Volume (m^3)	Drums Mass (kg)	Boxes/Bins Quantity	Boxes/Bins Volume (m^3)	Boxes/Bins Mass (kg)	Total Quantity	Total Volume (m^3)	Total Mass (kg)
Combustible	15,914	3,373.8	1,458,460.6	3,749	11,886.0	2,939,013.3	19,663	15,259.7	4,397,474.0
	11.4%	11.4%	7.4%	34.0%	33.9%	25.2%	13.1%	23.6%	14.0%
Filter	3,566	756.0	274,462.5	1,276	4,044.9	1,077,481.9	4,842	4,800.9	1,351,944.4
	2.6%	2.6%	1.4%	11.6%	11.5%	9.2%	3.2%	7.4%	4.3%
Graphite	2,383	505.2	261,995.0	1	3.2	879.9	2,384	508.4	262,874.8
	1.7%	1.7%	1.3%	0.0%	0.0%	0.0%	1.6%	0.8%	0.8%
Heterogeneous	5,799	1,229.8	683,693.7	604	2,022.4	792,871.1	6,422	3,256.2	1,480,385.5
	4.2%	4.2%	3.5%	5.5%	5.8%	6.8%	4.3%	5.0%	4.7%
Inorganic Non-Metal	5,775	1,224.3	608,383.1	485	1,550.3	579,755.9	6,260	2,774.6	1,188,139.0
	4.1%	4.1%	3.1%	4.4%	4.4%	5.0%	4.2%	4.3%	3.8%
Lead/Cadmium Metal Waste	3,725	789.7	408,868.7	4,574	14,521.7	5,829,900.1	8,299	15,311.4	6,238,768.8
	2.7%	2.7%	2.1%	41.5%	41.4%	49.9%	5.5%	23.7%	19.8%
Salt Waste	132	28.0	17,886.4	1	3.2	1,835.5	133	31.2	19,721.9
	0.1%	0.1%	0.1%	0.0%	0.0%	0.0%	0.1%	0.0%	0.1%
Soils	605	128.3	157,615.6	39	123.6	80,802.9	644	251.9	238,418.5
	0.4%	0.4%	0.8%	0.4%	0.4%	0.7%	0.4%	0.4%	0.8%
Solidified Inorganics	46,120	9,777.4	10,358,568.0	74	238.9	126,092.4	46,195	10,016.5	10,484,870.0
	33.1%	33.1%	52.3%	0.7%	0.7%	1.1%	30.7%	15.5%	33.3%
Uncategorized Metal	910	192.9	120,226.8	141	447.0	177,068.1	1,051	639.9	297,294.9
	0.7%	0.7%	0.6%	1.3%	1.3%	1.5%	0.7%	1.0%	0.9%
Unknown	54,302	11,512.0	5,444,477.7	72	252.0	67,435.5	54,374	11,764.0	5,511,913.2
	39.0%	39.0%	27.5%	0.7%	0.7%	0.6%	36.2%	18.2%	17.5%
Grand Total:	139,231	29,517.4	19,794,637.9	11,016	35,093.2	11,673,136.6	150,267	64,614.8	31,471,804.7

Matrix Category Groups described in EDF # RWMC-805 "Matrix Parameter Category Groups"

Data Source: IMWI Data Base as reported to BIR Rev. 3.

Table 1-2. INEL Accessibly Stored TRU Contaminated Waste Inventory.

Matrix Category Group	Drums			Boxes/Bins			Total		
	Quantity	Volume (m^3)	Mass (kg)	Quantity	Volume (m^3)	Mass (kg)	Quantity	Volume (m^3)	Mass (kg)
Combustible	1,862	454.3	214,373.0	931	3,060.9	720,350.0	2,793	3,515.2	934,723.0
	6.1%	6.9%	3.9%	51.6%	53.1%	41.5%	8.7%	28.5%	12.8%
Filter	2,029	425.1	147,497.0	150	481.2	123,490.0	2,179	906.3	270,987.0
	6.7	6.5%	2.7%	8.3%	8.3%	7.1%	6.8%	7.3%	3.7%
Graphite	1,402	294.5	149,487.0				1,402	294.5	149,487.0
	4.6%	4.5%	2.7%				4.4%	2.4%	2.0%
Heterogeneous	1,341	302.7	216,653.7	30	97.6	31,920.0	1,371	400.2	248,573.7
	4.4%	4.6%	3.9%	1.7%	1.7%	1.8%	4.3%	3.2%	3.4%
Inorganic Non-Metal	1,491	316.5	138,025.0	66	218.5	80,720.0	1,557	535.0	218,745.0
	4.9%	4.8%	2.5%	3.7%	3.8%	4.6%	4.8%	4.3%	3.0%
Lead/Cadmium Metal Waste	1,420	303.5	135,960.0	625	1,891.4	770,940.0	2,045	2,194.9	906,900.0
	4.7%	4.6%	2.4%	34.6%	32.8%	44.4%	6.4%	17.8%	12.4%
Salt Waste	55	11.8	5,150.0				55	11.8	5,150.0
	0.2%	0.2%	0.1%				0.2%	0.1%	0.1%
Soils				3	12.6	8,900.0	3	12.6	8,900.0
				0.2%	0.2%	0.5%	0.0%	0.1%	0.1%
Solidified Inorganics	20,513	4,412.0	4,521,237.0	1	0.3	58.0	20,514	4,412.3	4,521,295.0
	67.6%	67.1%	81.4%	0.1%	0.0%	0.0%	63.8%	35.8%	62.0%
Solidified Organics	89	19.0	11,824.0				89	19.0	11,824.0
	0.3%	0.3%	0.2%				0.3%	0.2%	0.2%
Uncategorized Metal	144	30.8	16,262.0				144	30.8	16,262.0
	0.5%	0.5%	0.3%				0.4%	0.2%	0.2%
Unknown	8	1.8	760.6				8	1.8	760.6
	0.0%	0.0%	0.0%				0.0%	0.0%	0.0%
Grand Total:	30,354	6,572.0	5,557,229.3	1,806	5,762.5	1,736,378.0	32,160	12,334.4	7,293,607.3

Matrix Category Groups described in EDF # RWMC-805 "Matrix Parameter Category Groups"

Data Source: SWEPP track portion of Transuranic Waste Data Base (TWDB).

Sludge categories represent the highest number of drummed waste, followed by combustible waste. Assuming that the unknown/uncategorized waste distributions are similar to the remaining distribution, approximately 55% of the drums contain sludge forms and 20% of the drums contain combustibles.

The major portion of the boxed waste is metals and combustibles, followed by mixed waste (paper, metals, glass, etc.) and insulation/filters. These four categories of waste represent approximately 95% of the boxed waste.

Lead-lined drums are found throughout the different waste categories. Liquids are primarily expected in the uncemented sludge, resins, and mixed waste categories. Particulate can be expected in the insulation and filter category wastes; graphite and nonmetallic molds and crucible waste; concrete, dirt, and brick waste forms; and the soils, asphalt, and ash waste forms.

1.6 Current INEL Nondestructive Radioassay Systems

There are two radioassay systems in use at the INEL RWMC, both are located at the Stored Waste Examination Pilot Plant (SWEPP). One is a passive active neutron (PAN) system which is used to determine plutonium content. The second is a passive gamma-ray system which is used to provide isotopics. The following is a brief description of these two systems.

1.6.1 Passive Active Neutron Assay System

The SWEPP radioassay system is a second generation PAN assay system developed in the early 1980s by Los Alamos National Laboratory (LANL) for the U.S. DOE and delivered to the INEL in 1983. This system was designed to assay drums containing transuranic contaminated waste. Later a similar system was built by LANL and delivered to the INEL for the purpose of assaying boxes containing transuranic waste. Even though much of the formalism will apply to both the drum and box assay systems, the specifics in this section will address only the drum system.

The SWEPP drum assay system is described in an INEL internal document by Becker[5]; for more details the reader is referred to that document. This system consists of a shielding housing which surrounds the drum on all four sides, top and bottom. Each side of the housing contains moderator (i.e., graphite, polyethylene), thermal, and low-energy neutron shielding (i.e., cadmium, boron), and ^{3}He neutron detectors. There are two types of detector assemblies contained in each side of the assay system: bare detectors and shielded detectors. The shielded detectors are grouped into detector packages where each package is surrounded by thermal and low-energy neutron shield consisting of cadmium and borated rubber. Inside the cadmium and borated rubber are three or four ^{3}He neutron detectors surrounded by polyethylene. This type of detector assembly is sensitive to fast neutrons and insensitive to thermal and low-energy neutrons. The bare detectors are also ^{3}He detectors surrounded by polyethylene but are not shielded by cadmium or borated rubber. In this configuration they are sensitive to all neutrons.

The assay system operates in two modes: passive and active. In the passive mode the detector assemblies (bare and shielded) detect neutrons produced by spontaneous fission and (α,n) interactions in the waste matrix. Differentiation between the fission neutrons and the (α,n) neutrons is accomplished by coincidence event counting. In this type of counting, a coincidence event is recorded when two or more neutrons are detected by the system within a specified time window. Two coincidence windows are used: one is 35 μs long and looks for coincidence events from the shielded detectors in the enclosure and the other is 250 μs long and looks for coincidence events from all detectors (shielded and bare) in the enclosure.

In addition to the coincidence counting, single event counting is also accumulated during the passive mode. The single event counting data are used to derive chance coincidence corrections to the coincidence data and also to arrive at a Moderator Index. This index is defined as the systems totals (shielded + bare) singles count divided by the shielded totals singles count.

In the active mode the shielded detectors are used to detect neutrons produced by stimulated fission resulting from thermal neutron interrogation. The interrogation neutron source for the active mode is a Zetatron 14 MeV neutron generator located at one corner inside the system shield enclosure. The high energy neutrons are moderated to thermal via the moderator in the enclosure walls and varying amounts of moderator in the waste matrix. For the active mode the signal of interest is taken from a gated count of the shielded detectors for the time window from 700 μsec to 2700 μsec following each neutron burst from the neutron generator. This time window was selected to allow the fast neutrons from the generator to thermalize in the enclosure and thereby have a higher probability to stimulate fission in the ^{239}Pu and at the same time the thermalized interrogation neutrons are not detectible by the shielded detectors. To account for background, another count window is opened from 5.7 msec to 15.7 msec after the each neutron burst. It is expected that at this time window only background neutrons will present.

Also during the active mode two monitors are used to monitor the interrogation neutron flux and the effective transmission of interrogation neutrons through the contents of the drum. The first monitor, called the cavity monitor, consists of a set of bare ^{3}He detectors mounted inside the cavity along an upper corner. The second monitor, called the barrel flux monitor, is a single ^{3}He detector mounted at the center of the back wall of the assay system enclosure inside a cadmium collimator so that the detector's field of view is the center of the drum. These two monitors are gated with the same time window as the shielded detectors during the active mode. The ratio of the cavity monitor count during active mode to the barrel monitor count during active mode is referred to as the absorber index.

The Moderator Index (from the passive mode count) and the Absorber Index (from the active mode count) are used in the analysis algorithm to arrive at correction factors which are supposed to correct for moderator and absorber effects on the measured responses (both active and passive responses). The corrected responses are used to determine the measured plutonium mass. Therefore, both the active and passive counts must be completed to obtain the needed correction factors. Three measured mass values are obtained by the system for each measurement sequence (passive count + active count); i.e., a mass value determined from the active mode count, a mass value determined from the passive short-gate coincidence count, and a mass value determined from passive long-gate coincidence count.

The two coincidence counts in the passive mode and the gated count in the active mode are used to produce three assay values of the plutonium in each waste drum. However, not all three values are valid over the mass range and waste forms covered in SWEPP waste, and a set of selection algorithms is included in the system software to determine which of the three assay values should be used in the waste certification documentation. For all waste forms except sludge, only one of the two passive mass values is selected as the reported mass based on the best relative uncertainty. The active mass is used as the reported mass for sludge waste forms.

The basic calibration of the PAN system is performed using standard sealed neutron sources in an empty waste drum. The original calibration was performed by LANL prior to delivery of the SWEPP system at the INEL. It has been checked repeatedly since as part of the SWEPP operational quality check program.

Listed below are the basic equations used by the PAN assay system to determine the mass assay values.

$$Mass_A = C_A * (Net\ Shielded\ Count)_A * CF_A \tag{1-1}$$

$$Mass_{lg} = C_{lg} * (Long\text{-}Gate\ Coincidence\ Rate)_P * CF_{lg} \tag{1-2}$$

$$Mass_{sg} = C_{sg} * (Short\text{-}Gate\ Coincidence\ Rate)_P * CF_{sg} \tag{1-3}$$

where

$Mass_A$ = Pu mass as determined from the active mode

$Mass_{lg}$ = Pu mass as determined from the passive mode and the long-gate coincidence method

$Mass_{sg}$ = Pu mass as determined from the passive mode and the short-gate coincidence method

C_A, C_{lg}, C_{sg} are the base calibration coefficients for the active, long-gate coincidence, and short-gate coincidence modes, respectively

CF_A, CF_{lg}, CF_{sg} are the matrix correction factors for the active, long-gate coincidence, and short-gate coincidence modes, respectively.

The effects of waste matrix, etc., on the base calibration were estimated during the original calibration series and an algorithm for determining the correction factors was developed by LANL. The correction factors were determined empirically using simulated waste drums in which generic materials (e.g., vermiculite, boric acid, sand and metal scraps) were used to simulate the waste matrix. The basic assumption in the development of the simulated waste was that the matrix was uniform, the source distribution was uniform and that each waste drum was filled to near the volume capacity of the drum. Over the years, there have been small changes made to the correction factor algorithm, but the basic premises (i.e., uniform matrix and uniform source distributions) have not changed.

1.6.2 Passive High Resolution Gamma System

The SWEPP passive high resolution gamma-ray spectrometry system consists of four high purity germanium detectors mounted inside a Canberra Q2 gamma shield. This gamma shield is composed of a 15cm thick steel enclosure capable of accepting a 55-gallon drum. The detectors are stacked vertically such that each detector looks at a different elevation on the drum. Currently there is no collimation associated with each detector. During a count the drum is rotating.

The principal use of the gamma system is to arrive at isotopic ratios (in most cases relative to ^{239}Pu). These ratios are used in combination with the plutonium assay values derived from the PAN system to give the specific isotopic activity. Typically the following isotopic ratios are determined:

$^{238}Pu/^{239}Pu$

$^{240}Pu/^{239}Pu$

$^{241}Pu/^{239}Pu$

$^{241}Am/^{239}Pu$

$^{235}U/^{239}Pu$.

In cases where there is only a barely detectible or no measurable ^{239}Pu activity and there is measurable ^{235}U activity, these ratios are determined relative to ^{235}U.

The analysis of the gamma spectrum from each detector is done using the gamma-ray analysis program for a VAX computer (VAX/GAP) peak search and fitting routine. The VAX/GAP results are then used in a special routine that produces the isotopic ratios for each detector spectrum and then combines single detector results to arrive at the weighted average of each ratio for the drum.

Each isotopic ratio is determined by comparing a peak area of the particular nuclide activity in question with that of a peak in the ^{239}Pu activity which is within a few keV energy of the former. Using peaks close in energy helps to minimize attenuation effects in the resulting ratio. The underlying assumption in this technique is that the two activities are coming from the same regions in space. This assumption is probably good for ratios involving the plutonium activities, but is questionable for ratios involving americium and uranium relative to plutonium.

The germanium detectors in the SWEPP gamma system are 10% efficiency coaxial detectors with an energy resolution of ≈ 600 eV at 120 keV. This energy resolution is needed in order to resolve the gamma peaks used in the analysis described above. The efficiency was chosen as a compromise between efficiency and energy resolution. Since most of the gamma lines used in this analysis are below 200 keV, a higher efficiency based on cobalt activity does not necessarily equate to an improved sensitivity at the energies used in the analysis.

1.7 Known Issues and Problems with Waste Assay

The radioassay program at the INEL has identified a number of issues and problems with the current methods used to assay waste forms. The following describes the contributors to the uncertainties for the PAN system, the limitations of the current SWEPP assay systems, and other issues which have been found to affect the accuracy of assay results.

1.7.1 Contributors to the Assay Uncertainties for the PAN System

There are three main contributors to the assay uncertainties for the PAN system:

Base Calibration—In the base calibration, the system response is measured for a well characterized neutron source (i.e., known neutron strength and elemental and chemical composition) at specified positions in an empty waste drum. There are three primary uncertainties associated with the base calibration. The first is the uncertainty for the source strength which includes any decay corrections which are applied and the number of neutrons produced per decay. The second is the uncertainty about the elemental and chemical composition of the neutron source material. The elemental and chemical composition can significantly affect the reported neutron source strength by producing an unknown number of neutrons produced by (α,n) interactions in the source. The third uncertainty is the counting statistics associated with the base calibration data acquisition.

Matrix and Source Effects—In quantifying the estimates for systematic biases and uncertainties, the major questions are: How valid is the uniform matrix and uniform source premise used in the PAN algorithm to the application of assaying a particular class of waste, and what kind of errors are introduced as a result? Listed below are the specific ways that real waste may differ from the uniform matrix and uniform source premise:

1. Source composition effects
2. Non-uniform matrix absorption
3. Non-uniform matrix moderation
4. Non-uniform source distribution
5. Variations in source particle size
6. Significant voids in the matrix
7. Shadow shielding of one region by high neutron absorption in another region
8. Waste elemental composition not addressed by the calibration routine.

(α,n) Source Interference—In addition to the matrix and source introduced errors there are also uncompensated effects resulting from (α,n) reactions occurring in the waste. Since the (α,n) reactions only produce one neutron per reaction, the coincidence counting method in the passive mode should differentiate between neutrons produced by fission (more than one neutron per fission) and neutrons

produced by (α,n) reactions. However, the coincidence counting method will have a contribution due to accidental or chance coincidences. There are standard techniques to correct the coincidence counting data for these spurious events and these techniques work well when the corrections for chance coincidences is small compared to the real coincidence rate. In those cases where the (α,n) source strength is clearly dominant over the fissile neutron source strength (i.e., the chance coincidence rate is dominant over the true coincidence rate), there is a very large uncertainty associated with the correction for chance coincidence events.

In addition, high count rates will also lead to counting losses which are not compensated for in the simple correction applied in the assay system analysis routine. For example, the standard corrections applied for counting losses are based on the assumption of random events and are not applicable to correlated events as is the case in coincidence counting. The random event based corrections are valid when the correction is small but not when the counting loss is the same order of magnitude as the basic count rate. There are drums at RWMC where the neutron count rate is high enough that this circumstance applies. Under these situations the corrections are considered suspect and contribute significantly to the overall uncertainty of the measurement.

Thus (α,n) interference and counting losses can be sources of significant uncertainties in the assay results. In fact, recent experience has indicated that these effects can be the dominant contribution to the uncertainty of the passive assay results.

1.7.2 Limitations of the Current SWEPP Nondestructive Assay Systems

The major limitation of the SWEPP gamma system is that it requires long counting times ($>$1h) to arrive at good statistical data in the gamma peaks of interest, and in some cases a 10 hour count time does not yield sufficient statistical data. There is an on-going initiative to determine what may be a reasonable cut off, below which drum-specific activity ratios are not required. Establishing this criterion will greatly enhance the effective production capacity of this technique. An initiative has also been started to look at alternative analysis techniques which utilize all the peaks associated with a nuclide activity rather than just one or two peaks.

The second most common deficiency of the SWEPP gamma system and the current analysis is that the gamma peak energies of interest are at low energies and are severely affected by matrix attenuation and self-absorption. For example, in sludge a 30g sample of plutonium in the center of a drum is not detectible by the SWEPP gamma system.

The SWEPP PAN radioassay system also has severe limitations for some waste matrices. For example, matrix effects occur because of density and moisture (i.e., hydrogen) content. These matrix effects will attenuate neutrons and change the neutron energy. There will also be source effects due to varying source isotopics (e.g., Pu isotopes, ^{233}U, ^{235}U, etc.). The composition of the source material will affect the (α,n) production rate. High radiation fields (often due to large amounts of ^{241}Am in a waste drum) can affect the assay results. In addition, the INEL has observed other problems such as variability in packaging and neutron channeling. All these matrix effects must be taken into account (e.g., by correction factors) in coming up with the assay results.

All of these effects tend to decrease the sensitivity and accuracy of an assay method, increase the total uncertainty in the results, and make it more difficult for an assay technique to meet QAPP requirements.

2. SELECTION OF CANDIDATE TECHNOLOGIES FOR TEST BED EVALUATION

2.1 Introduction

This test bed will be available to technology holders who can pass a preliminary screening: a preliminary technical evaluation of their technique, technically justify their approach, and show applicability of their technology to INEL waste, i.e., they must demonstrate general performance characteristics such as portability, sensitivity, range, accuracy, throughput, and area of focus. In addition, it is required that participants will have calibrated their system prior to coming to the INEL. If they wish to checkout their calibration after setup at the INEL, they will be expected to bring their own calibration drum(s) for this purpose. The INEL, however, will supply source material for use in these calibration drums.

2.2 Technology Holder's Responsibilities

Participation in an INEL TRU waste assay test bed evaluation of an assay technology is on an invitation only basis. It is only intended for testing and evaluation of assay systems that are mature and that have direct applicability to INEL waste assay needs. It is not generally intended for "proof-of-principle" experiments or for system calibration purposes. The system must have been thoroughly tested and calibrated prior to arrival at the INEL. Only limited time (1-1/2 week maximum) will be allowed at the test bed for system setup, checkout, and calibration checks. Also, only limited facilities (source material for use in a technology holder's calibration drum) will be available for system calibration checks.

In order to be considered for an invitation, a technology holder must submit to the INEL detailed information about his system together with supporting data. This information and data must be sufficient to allow an assessment of the potential technical capability regarding high population fraction INEL waste forms. Potential technical capability is defined in terms of the compliance demonstration requirements of the DOE-Carlsbad Area Office (CAO) TRU Waste Characterization QAPP. This QAPP delineates quality assurance objectives that must be met for waste to be disposed at the Waste Isolation Pilot Plant. Hence, requested information and data will relate to calibration technique, apparatus configuration response characteristics, response correction approaches where inadequate dynamics exist to measure an attribute, bias and precision element identification, error quantification and propagation technique, low-level waste (LLW)/TRU segregation performance, and source mass/activity operational range.

It is recognized that due to the developmental nature of most waste NDA systems, a complete set of data will, in many cases, not be available for all the requested elements of the pre-evaluation plan. In the event requested information and data are not available, the technology holder can supply partial or related information/data. If the technology holder deems the data to be inadequately developed for dissemination, a description of planned activities, methods, and schedule to acquire the data and/or to develop the technique is sufficient. Although the lack of information does not disqualify potential participation in a test bed evaluation, it does require that efforts be identified to establish such information and data. Refusal to provide information or data due to proprietary claims is not acknowledged as a valid reason for withholding technical capability data. Such information is required to support QAO audits and must of necessity be sufficiently divulged if the system is to be used to generate data for the National TRU Waste Characterization Program.

Preliminary capability assessment information and data will be requested of a prospective participant via a technical questionnaire as described in Section 2.3. This questionnaire is part of the TRU NDA System Evaluation Request Package. The second major part of the TRU NDA System Evaluation Request Package is the operational questionnaire described in Section 2.4, which requests information concerning safety, environmental, and other operational requirements such as power and space.

The first step in obtaining an invitation to participate in a test bed evaluation is accomplished by submitting the TRU NDA System Evaluation Request Package. The contact for obtaining this evaluation package is T.L. Clements, Jr. The package will be reviewed by INEL technical staff, and if approved, a test bed evaluation of the technology holder's system will be scheduled.

2.3 Contents of the Technical Questionnaire

The following is an overview of the data and information requested in the technical questionnaire.

2.3.1 System Description

A general description of the fundamental operating principles of the system, including the physical configuration and the assay parameters which the system addresses (e.g., total fissile content, isotopics, etc.) is required. This entails a description of the characteristic radiation detection and signal processing technique. For example, a passive neutron waste assay system is to be described in terms of detector type and signal and data processing methods. The physical configuration (e.g., number/location of detectors, dimensions/locations of neutron moderator and shielding) is to be described with associated drawings. A description and basis for modifications or enhancements to the base detection technique principle are to be provided. The required information associated with this request item consists of a general overview of the system design and detection principles.

2.3.2 Calibration Technique

A detailed discussion of the calibration technique is required. This includes the following topics:

- Describe the general calibration process, including the primary source attribute and instrument response parameter(s) used for calibration, how correlations are established, etc.

- Describe the apparatus, the procedure for establishing the calibration, and the distribution of source material. For example, is source material placed in an empty 55-gallon drum or one which contains surrogate material? Are a number of point sources simultaneously distributed or is one source measured in many differing volume element locations and a composite response determined by combination of the data?

- Discuss whether separate calibrations are made for differing matrix configurations and source distributions. If standard matrices are used to derive calibrations, describe the standard set and quality assurance measures associated with their fabrication and maintenance.

- Indicate whether the source(s) used for calibration have associated pedigrees specifying a primary attribute traceable to a recognized reference base.

- Describe the configuration and composition of the sources used for calibration (e.g., Pu oxide foils, Pu/diluent mixtures, dimensions, masses), the mass range that the calibration standards encompass, and the range through which the system calibration is valid.

- Provide correlation results (including curve fit of calibration data, counting statistics associated with the calibration data acquisition).

2.3.3 Data Acquisition/Reduction

A detailed discussion of the method of data acquisition/reduction is required which includes the following topics:

- Describe the data acquisition system in terms of signal processing and storage.

- Describe how the data are used to account for variations in source radionuclide composition from drum to drum and at different volume elements within a drum.

- Describe how variations in source configuration (i.e., diffuse, aggregate, lumps, and combinations thereof) are accounted for in the data acquisition and reduction routine.

- Delineate whether the chemical composition of source material affects response of the NDA system. For example, depending on the chemical compound of the fissile material, various reactions can result yielding interfering radiations, the most common being the (α,n) effect. State the limiting value of the interfering parameter which can be tolerated by the system.

- Describe the effect of varying source spatial distributions on the response of the instrument and the means that are used to correct for variations in instrument response due to source spatial distribution.

- Define the characteristic radiation emission rate acquisition capabilities.

- Describe how attenuation effects due to waste matrix density and elemental composition variations and how nonuniform density distribution effects are handled.

- Describe how elemental composition variations from drum to drum and intra-drum are accounted for.

2.3.4 Assay Uncertainty Determination

Compliance with the QAPP total uncertainty QAO requires an accounting of all biases and precision components and the implementation of an appropriate error propagation technique. Therefore, the technically defensible means to demonstrate the method of determining error components and their combination and propagation must be described (or, if the method is still in the developmental phase, a description of the planned method), including justification of any use of assumptions in place of actual error component measures.

2.3.5 Realm of Application

No single waste NDA technology can accommodate the entire spectrum of waste forms of interest to the INEL. Hence, a definition is required of the operational realm of the technology holder's waste NDA system in terms of waste form and waste form attribute (e.g., waste container size, matrix density, radiation emission rate, source chemical composition, variations in source radionuclide composition, variations in source configuration, etc.). Parameters such as sensitivity, accuracy, and precision should be included. In addition, data to back up all claims should be included. Where operational experience is not available, estimations of performance or simple acknowledgement of lack of experience is acceptable.

2.3.6 Segregation Capability

The ability of a waste NDA system to provide a waste segregation capability at the 100 nCi/g alpha activity concentration level is of considerable interest to the INEL. Therefore, information regarding the detection technique utilized for segregation is required. This includes lower limit of detection data and a description of the derivation method.

2.3.7 Throughput Rate

Since throughput rate is very important to the INEL, information concerning throughput rate is required as a function of TRU mass loading (e.g., acquisition time at low TRU concentration values[100 nCi/g], etc.). Data concerning system preparation (i.e., operability verification) and data acquisition time per drum are required.

2.3.8 Operational Experience

The maturity of the NDA system is of interest to the INEL. Therefore, a description of the state of the system (i.e., prototype or production) is required. Performance data from any operational use of the NDA system should also be provided, including a description of the operational campaign, waste form types addressed, calibration methods employed, and any intercomparison data available.

2.4 Contents of the Operational Questionnaire

The following is an overview of the data and information that are requested in the operational questionnaire:

- A list and brief discussion (where appropriate) of the operational needs and requirements of the prospective participant's system is required. This includes, but is not limited to power and other utilities, space, environmental (e.g., temperature range, humidity) requirements; time requirements (e.g., time require for system setup and checkout, time required to perform assay run, analysis and data workup time [i.e., time required after assay run to finalize assay results]); and assistance required from INEL personnel (including needs for radioactive and source material).

- A general overview of applicable National Environmental Policy Act (NEPA), safety, and other operational issues is required, including copies of all available documents. See Section 6 for a more detailed discussion on operational constraints and considerations.

2.5 Support to be Supplied by the INEL

The technology holder is expected provide for the shipment of his system to and from the test bed and for his staff's participation in the evaluation. The INEL will provide local support required for the evaluation, including review of the applicant's application, limited assistance from the test coordinator and INEL scientist(s), facility utilities and hookups, radcon technicians, waste drum tracking and drum handling support, and evaluation of the resulting data supplied by the participant. In addition, the INEL will provide limited site-specific training (if required) and limited source material for system calibration checks.

Participants are expected to operate their systems, then analyze the resulting data and provide an assay result report to the INEL for each drum assayed. The INEL will not supply assistance in performing system checkouts (other than supplying source material), waste form or surrogate drums to be used for calibration purposes, or access to INEL areas other than the test bed area.

The participants will be expected to assay all drums in each requested set. At a minimum, a set of noninterfering matrix drums must be assayed. Depending on the results of the noninterfering matrix drum assays, assay of additional sets of drums (e.g., surrogate set, waste form set, optional set consisting of additional surrogate and/or waste form drums) may be requested by the INEL. The participant is expected to complete the assay of all drums in each set requested by the INEL (with the exception of the optional set).

Participant personnel must be U.S. citizens in order to be allowed access to the test bed. Non-citizens will not be allowed access to any INEL facilities. Participants will not be allowed to handle either radioactive or source material unless they are trained and qualified to do so (i.e, unless they are DOE Radiation Workers and Fissile Material Handlers). If the participants do not possess the required training, INEL will provide assistance in handling radioactive and source material.

3. OVERVIEW OF RADIOASSAY REQUIREMENTS

The intent of Sections 3 and 4 is to identify the key requirements that any newly proposed nondestructive assay technology must meet. In order to meet INEL needs, any TRU waste assay system must satisfy quality assurance requirements for the assay of the waste forms present at the INEL. This section presents key technical and functional requirements that any advanced technology must meet, and Section 4 gives a general description of the waste forms to which the assay system must be applicable.

3.1 Quality Assurance Requirements

Any advanced technology method for NDA of waste containers must address and achieve the established QAOs. The QAOs establish minimum performance requirements for measurement systems used to generate waste characterization data. It must be demonstrated and technically justified that the technique is appropriate for the specific waste for which it is applicable. The rationale for using the technique should include the physical form of the waste, the radionuclide content, and the waste generating process. The total uncertainties in the assay must be calculated using the terms derived for compliance with the QAO for Total Uncertainty and reported with the data. The actual precision and accuracy values obtained for waste containers will be a function of the waste type, total TRU content, its distribution, and characteristics of the measurement instrumentation. The QAOs for precision, accuracy, minimum detectable concentration (MDC), completeness, and total uncertainty have been determined and are presented below and summarized in Table 3-1.

The following information concerning the QAOs for precision, accuracy, sensitivity limits, MDC, total uncertainty, and completeness was extracted from Section 9 of the National TRU Program Quality Assurance Plan.[1] Appendix A contains Section 9 of the QAPP in its entirety. A draft revision of the QAPP is in review which, when released, may modify some of the QAO requirements discussed below.

3.1.1 Precision

The precision of each measurement techniques must be determined through replicate processing of a waste container containing a known quantity of the radioactive material of interest. Demonstration of compliance with the QAO for precision shall be by replicate processing of a waste container (208-liter [55-gallon] drum) containing the quantities of TRU isotopes indicated in Table 3-1 for each range for which the measurement system is to be qualified. The activity shall be distributed in a well-characterized, non-interfering matrix and shall not be one of the standards used to calibrate the counting system. A total of 15 replicate counts shall be obtained with removal of the waste container from the measurement system and reinsertion of the waste container into the measurement system between measurements. The precision shall be computed as the percent relative standard deviation (%RSD) of the distribution of these replicates.

Table 3-1. Quality assurance objectives for nondestructive assay.

Range of waste activity in α-Curies[a]	Nominal compliance point α-Curies[a] (g WG Pu)[b]	Precision[c] (%RSD)	Accuracy[d] (%R)	Total uncertainty[e]	Completeness[f]	MDC (nCi/G)[g]
0	0					60
>0 to 0.04	0.008 (0.1)	≤20	75–125	Low 40% High 175%	100	
>0.04 to 0.4	0.08 (1.0)	≤15	50–150	Low 30% High 200%	100	
>0.4 to 4.0	0.8 (10)	≤10	50–150	Low 30% High 200%	100	
>4.0	12.8 (160)	≤5	75–125	Low 50% High 150%	100	

a. Applicable range of TRU activity in a 208-liter (55-gallon) drum to which the QAOs apply, units are Curies of alpha-emitting TRU isotopes with half-lives greater than 20 years.

b. The nominal activity (or weight of Pu) in the 208-liter (55-gallon) drum used to demonstrate that QAOs can be achieved for the corresponding range in column 1, values in parentheses are the equivalent weights of weapons grade plutonium (WG Pu), fifteen years after purification; for purposes of demonstrating QAOs, "nominal" means within ±10 percent.

c. ± one relative standard deviation based on fifteen replicate measurements of a non-interfering matrix.

d. Ratio of measures to known values based on the average of fifteen replicate measurements of a non-interfering matrix.

e. 95-percent confidence bounds of all propagated uncertainties (confidence bound divided by true value, expressed as a percent).

f. Valid radioassay data are required for all waste containers

g. As defined by equation 3.1.

3.1.2 Accuracy

Accuracy is determined through replicate processing of a waste container containing a known quantity of radioactive material of interest. Accuracy is calculated from the ratio of the mean measured estimate to the known value for an accepted calibration or verification standard. Calibration standards are those used to determine the response characteristics of a measurement system. Verification standards are used to test the validity of a calibration independently of the original calibration standards. Demonstration of compliance with the QAO for accuracy shall be by replicate processing of a waste container (208-liter [55-gallon] drum) containing the quantities of TRU isotopes indicated in Table 3-1 for each range for which the measurement system is to be qualified. This activity shall be in the form of a verification standard (i.e., it shall be characterized as well as the calibration standards) but it may not be one of the calibration standards nor shall it be derived from or calibrated against one of the calibration standards. The activity shall be distributed in a well-characterized, non-interfering matrix and shall not be one of the standards used to calibrate the counting system. A total of fifteen replicate counts shall be obtained with removal of the waste container between measurements. The accuracy shall be computed as the %R of the known value.

3.1.3 Sensitivity Limits

Discrimination between LLW and TRU waste may only be made with systems for which adequate sensitivity limits have been documented. The ability to achieve the required detection limit in Table 3-1 must be demonstrated for each waste type/method combination planned for use. The detection limit is defined to be that level of radioactivity which, if present, will yield a measured value less that the critical limit with 5-percent probability. The critical limit is defined as that value which measurements of the background will exceed with 5-percent probability.

3.1.4 Minimum Detectable Concentration

The detection limit to be used is the MDC, which is a level of activity that is practically achievable with the given instrument, analytical method, and analyte/matrix combination. The MDC considers not only the instrument characteristics (background and efficiency), but all other factors and conditions which influence the measurement. It is an *a priori* (before the fact) estimate of the activity concentration that can be practically achieved under a set of typical measurement conditions. These would include the waste quantity, counting time, matrix specific corrections, decay corrections, and any other factors that comprise the activity concentration determination. The MDC is an *a priori* estimate of the detection capabilities of a given measurement system and method. It is based on the premise that from a knowledge of the background count and other measurement system parameters, an *a priori* limit can be estimated for a particular measurement.

The MDC is defined as

$$MDC = K_1K_2(2.71 + 4.65*s_b). \qquad (3\text{-}1)$$

where

K_1 is the proportionality constant relating the detector response (counts) to the activity, such as, $K=1/e$, where e is an overall detection efficiency, or $K=1/l_r e_r$, where l_r is the gamma-ray emission probability per decay and e_r the detection efficiency for the gamma ray

K_2 is the factor which relates the total activity determined by the measurement system to an activity concentration in waste under a given set of measurement conditions, for example, the weight of waste assayed and a self-absorption correction

s_b is the standard deviation of the background.

This equation incorporates the following assumptions:

- The preselected risk for concluding falsely that activity is present above the critical level and the predetermined degree of confidence for correctly detecting its presence above the critical level are 5 percent and 95 percent, respectively;

- In the vicinity of the MDC, the gross measurement counts and background counts will be approximately equal.

Calculations used to demonstrate attainment of the QAO for the MDC should use typical or average values for the parameters comprising K_2 in Equation 3-1. Demonstration of compliance with the QAO for MDC may be by replicate processing of an approximately sized waste container containing only a well-characterized, non-interfering matrix with no added activity. A total of fifteen replicate counts shall be obtained with unloading and reloading between replicates. The MDC shall be computed using the variance of the background count and Equation 3-1 or the analogous computation using all parameters appropriate to the measurement method.

3.1.5 Total Uncertainty

Total uncertainty includes propagated uncertainty for all corrections and factors applied to the analysis of real wastes to compensate for inhomogeneities and matrix interferences. The QAO for total uncertainty is intended to include estimates of the cumulative uncertainties from all correction factors and adjustments which are applied to the analytical data to compensate for inhomogeneous distribution, shielding, self-absorption, attenuation, and other matrix effects. Uncertainties in any parameters influencing the computation of radioactivity content must be included in the calculation of total uncertainty. This specifically includes parameters such as the isotopic ratios when assumed to be constant or determined from data sources external to the actual measurements. The ability to achieve the QAO for total uncertainty will be demonstrated through propagation of all uncertainties and documentation of all applied correcting factors; their derivation, source or justification; the range of waste measurements to which they will be applied; and the uncertainty associated with each factor. The uncertainties at the 95% confidence level from all correcting factors shall be propagated along with estimates of the uncertainty due to any other source of precision error. It is this value which must meet the QAO for total uncertainty. The QAOs for total uncertainty are expected to be achievable in the presence of backgrounds generated by alpha- and gamma-emitting sources and in the presence of interfering quantities of neutron and gamma absorbing and moderating material, as is the case for much of the waste encountered.

3.1.6 Completeness

Acceptable data shall be obtained for 100 percent of the waste containers characterized for disposal. Acceptable radioassay data shall consist of data on the radioactivity content of the waste package obtained from measurement systems which have been demonstrated to have met all the relevant QAOs for radioassay.

3.2 WIPP-WAC and TRUPACT-II SARP Requirements

In addition to the above QAOs required by the National TRU Program QAPP, the WIPP Waste Acceptance Criteria (WAC)[6] and the Transuranic Package Transporter-II (TRUPACT-II) Safety Analysis Report for Packaging (SARP)[7] contain the following additional requirements which may impact the radioassay method.

3.2.1 Liquids

The WIPP-WAC states that liquid waste is not acceptable at the WIPP. Containers shall contain less than 1 inch (2.5 cm) of liquid in the bottom of the container. In no case shall the total liquid volume exceed 2 liters in a 55-gallon drum. The TRUPACT-II SARP requires that the total volume of residual liquid in a payload container shall be less than 1 volume percent of the payload container.

3.2.2 Criticality

The WIPP-WAC requires that the fissile or fissionable radionuclide content, in terms of ^{239}Pu fissile-gram equivalent (FGE), of CH-TRU payload containers shall be no greater than 200 g per 55-gallon drum. The ^{239}Pu FGE shall be calculated using the methods detailed in Section 9.4 of Appendix 1.3.7 of the TRUPACT-II SARP. In addition, the TRUPACT-II SARP requires that the maximum allowable FGE quantity (i.e., 200 g) include two times the measurement error.

3.2.3 Pu-239 Equivalent Activity

The WIPP-WAC requires that untreated CH-TRU waste shall not exceed 80 plutonium equivalent curies (PE-Ci) of activity per 55-gal drum. The PE-Ci is to be calculated as shown in Appendix A of Reference 7 and is reproduced in Appendix B of this test plan.

3.2.4 Contact Dose Rate

The WIPP-WAC requires that CH-TRU waste payload containers shall have a maximum contact dose rate (beta + gamma + neutron) at any point no greater than 200 mrem/hr.

3.2.5 Thermal Power

The WIPP-WAC requires that individual CH-TRU waste payload containers in which the average thermal power density exceeds 0.1 watt/ft^3 (3.5 watts/m^3) shall have the thermal power recorded in the data package. In addition, the TRUPACT-II SARP requires that the thermal limit for total decay heat from all CH-TRU waste payload containers in a TRUPACT-II be 40 watts.

3.2.6 TRU Alpha Activity Concentration

The WIPP-WAC requires that CH-TRU have a lower total alpha limit of 100 nCi/g of waste matrix material. A propagated error shall be included in the calculation of the lower limit of activity concentration.

3.3 Functional Requirements

The major RWMC production objective (driven by the settlement agreement) is to certify at least 15,000 drums for shipment from INEL to WIPP by the end of CY-2002. In order to meet this objective, the production schedule requires 3 to 4 drums to be assayed per hour. In addition, the assay system must be designed so that during production it can be operated reliably by trained technicians, i.e., it must not require a physicist as an operator.

4. WASTE FORM DESCRIPTIONS

4.1 General Description of Waste Forms at the INEL

More than half of the total TRU waste inventory stored at the INEL RWMC consists of uncemented inorganic sludges and combustibles. Other high population waste forms are cemented sludge, uncemented organic sludge, metals, glass, and raschig rings. These six categories account for over 80% of the total waste. Lower population waste forms include filters and insulation; nonmetal (graphite) molds and crucibles; particulate wastes such as soils and dirt; and firebrick, concrete, and asphalt. Table 1-1 contains more details about the total inventory of waste stored at the INEL RWMC.

All of the waste inventory stored at the INEL RWMC, however, is not readily accessible and, therefore, will not be included in the initial shipments to WIPP. Of the readily accessible waste (i.e., the first waste that will be shipped to WIPP to satisfy the settlement agreement) greater than 50% consists of uncemented inorganic sludge. Other major categories include uncemented organic sludge, combustibles, filters, metals, and glass. It is the assay of the readily accessible drum inventory which is of prime importance to the INEL at this time and, therefore, will be of prime importance for the assay system evaluations performed at the test bed. Waste in these categories will make up the bulk of the set of waste forms which a participant will be requested to assay if his system performs adequately on the Performance Demonstration Program (PDP) and surrogate sets. Other waste forms will be included in the optional set.

4.2 Description of Accessible Waste Forms

This section presents brief descriptions of the contents (including information on weights and source material contents) of the higher population waste drums which are accessible and, therefore, subject to shipment to WIPP to satisfy the settlement agreement. All of these waste forms came from Rocky Flats, consequently, they contain varying amounts of weapons grade (WG) plutonium, americium, and uranium. These waste forms (with the exception of content codes 337, 374/960, 376, 432, and 339/463, which are in the optional waste form set) comprise the basic waste form drum set which a participant may be requested to assay. Information in this section was obtained from References 3 and 4. More details concerning these waste forms can be found in Appendix C.

4.2.1 Uncemented Inorganic Sludge (content code 001/002/800)

This waste is a wet sludge precipitate generated by processing liquid wastes such as ion exchange column effluents, distillates, caustic soda solution, etc. produced by Rocky Flats plutonium recovery operations. The waste was packaged in 55-gallon drums to which Portland cement was added to absorb any free liquids. The average drum weight for content code 001 is 490 lbs, and the range is 118 to 933 lbs. Plutonium and americium inventories average 4.3 g Pu and 1.8 g Am and range from 0 to 157 and 52.9 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present. The average drum net weight for content code 800 (which replaced content code 001 in 1986) is 369 lbs, and the range is 157 to 614 lbs. Plutonium, americium, and uranium inventories average 4 g, 0.9 g, and 0.7 g, respectively; they range from 0 to 32 g, 0 to 3.9 g, and 0.3 to 1.7 g respectively. The average drum net weight for content code 002 is 528 lbs, and the range is 210 to 952 lbs. Plutonium and americium inventories average 0.2 g Pu and 0.0 g Am and range from 0 to 8.9 and 7.1 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

4.2.2 Uncemented Organic Sludge (content code 003)

This waste comes from the processing of organic wastes generated at the various plutonium and nonplutonium operational areas at Rocky Flats. Organic waste is processed for packaging by blending approximately 30 gallons of organics with 100 pounds of calcium silicate. The average drum weight for this waste form is 509 lbs, and the range is 89 to 910 lbs. Plutonium and americium inventories average 0.3 g Pu and 0.0 g Am and range from 0.0 to 16.0 and 1.2 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

4.2.3 Solidified Organics (content code 801)

This waste consists of cemented waste oils and solvents that were generated as a result of machining and tool degreasing. Envirostone emulsifier, gypsum cement, and accelerator were mixed with the waste to solidify it. The average drum net weight for this waste form is 453 lbs, and the range is 122 to 598 lbs. Plutonium inventories average 3.26 g Pu and range from 0 to 70 g. About 1 g Am is reported in each drum. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

4.2.4 Special Setups (content code 004)

This waste consists of liquids absorbed on a cement mixture, the liquids containing plutonium complexing chemicals such as alcohols, organic acids, Versenes (trademark for a series of chelating agents based on EDTA). The waste was packaged in 55-gallon drums to which a mixture of Portland cement and pipe insulation cement (e.g., magnesia cement) was added to absorb any free liquids. The average drum weight for this waste form is 585 lbs, and the range is 102 to 1076 lbs. Plutonium and americium inventories average 1.0 g Pu and 0.0 g Am and range from 0.0 to 22.7 and 2.4 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present. The average drum net weight for content code 802 is 559 lbs, and the range is 239 lbs to 649 lbs. Plutonium and uranium inventories average 5 g Pu and 37 g ^{235}U and range from 0 to 21 g and 1 to 73 g, respectively. No average or range for americium has been reported. This does not, however, preclude the possibility that americium may be present.

4.2.5 Solidified Lab Waste (content code 802)

This waste consists of liquid lab waste containing hydrochloric acid. Portland cement and absorbent cement is added to the waste to immobilize it. The average drum net weight for this waste form is 559 lbs, and the range is 239 to 649 lbs. Plutonium and uranium inventories average 5 g Pu and 37 g ^{235}U and range form 0 to 21 g and 1 to 73 g, respectively. No average or range for americium has been reported. This does not, however, preclude the possibility that americium may be present.

4.2.6 Cemented Inorganic Sludge (content code 007)

This waste is a variant of the content code 001 packaging configuration. The content code 007 configuration differs from that of content code 001 in that the sludge material itself has approximately 50 pounds of Portland cement uniformly mixed and distributed. The average net drum weight for content code 007 is 410 lbs, and the range is 117 to 650 lbs. Plutonium and americium inventories average 0 g Pu and 0 g Am and range from 0 to 29 and 0.06 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

4.2.7 Solidified DCP Sludge (content code 803)

This waste consists of clarifier slurry from the Radioactive Decontamination Process, Acid Neutralization Process wastes, and acid descaling solution from the Evaporation Process. It was then cemented in the Direct-Cementation Process (DCP) prior to packaging in 55-gal. drums. The average drum net weight is 530 lbs, and the range is 157 to 672 lbs. Plutonium and americium inventories average 0.3 g Pu and 0.1 g Am and range from 0 to 4.3 and 0.3 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

4.2.8 Solidified Bypass Solids (content code 807)

This waste consists of immobilized materials from the Decontamination-Precipitation and Neutralization Process in the Liquid Waste Treatment Facility, clarifier slurry from the Radioactive Decontamination Process, Acid Neutralization Process wastes, and acid descaling solution from the Evaporation Process. It was transferred to 55-gal. drums where it was mixed with Portland cement and diatomite. Content code 807 was created in 1987 to replace content code 007. The average drum net weight is 353 lbs, and the range is 10 to 509 lbs. Plutonium and americium inventories average 2 g Pu and 0.01 g Am and range from 0 to 161 and 0.155 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

4.2.9 Cemented Sludge (content code 292)

This waste consists primarily of sludge generated from filter plenums, pumps, and incinerator off-gas systems. It may also contain a limited number of surgeons' gloves. Portland cement is added to the sludge for absorption of free liquid. The average drum weight for this waste form is 265 lbs, and the range is 111 to 522 lbs. Plutonium and americium inventories average 23.6 g Pu and 0.0 g Am and range from 0.0 to 162 and 2.0 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

4.2.10 Combustibles—dry (content code 330)

This waste consists primarily of dry combustibles such as paper, rags, plastics, surgeons' gloves, cloth overalls and booties, cardboard, wood, wood filter frames, polyethylene bottles, and laundry lint. The average drum weight for this waste form is 183 lbs, and the range is 82 to 561 lbs. Plutonium and americium inventories average 0.5 g Pu and 0.0 g Am and range from 0.0 to 45 and

27 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

4.2.11 Combustibles—moist (content code 336)

This waste consists of damp or wet line- and some nonline-generated combustible wastes such as paper, rags, and KimWipes. Other combustibles which might be present include plastics, surgeons' gloves, canvas, wood, cardboard, polyethylene bottles, and rubber. The average drum weight for this waste form is 197 lbs, and the range is 91 to 596 lbs. Plutonium and americium inventories average 0.3 g Pu and 0.0 g Am and range from 0.0 to 55 and 45 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

4.2.12 Metals—unleached (content code 480)

This waste consists of non-stainless steel metals (such as iron, copper, aluminum) and stainless steel which may be in the form of gloveboxes, glovebox windows, furnaces, lathes, drill presses, ducting, piping, angle iron, tanks, downdraft tables, part-carriers, respirator filters, ultrasonic cleaners, control panels, electronic instrumentation, vacuum sweepers, pumps, motors, railing, stairs, metal racks and trays, hotplates, empty metal produce and paint cans, carts, power tools, hand tools, chairs, desks, tables, typewriters, filing cabinets, crushed 55-gallon drums, etc. The average drum weight for this waste form is 253 lbs, and the range is 90 to 795 lbs. Plutonium and americium inventories average 3.6 g Pu and 0.0 g Am and range from 0.0 to 129 and 20 g, respectively. Uranium-235 inventories of 12 g have been reported.

4.2.13 Metals—leached (content code 481)

This waste consists of non-stainless steel metals (such as iron, copper, and aluminum) and primarily stainless steel in the form of small hand tools, valves, trays, clamps, pipe, etc. The average drum weight for this waste form is 295 lbs, and the range is 103 to 658 lbs. Plutonium and americium inventories average 21.6 g Pu and 0.0 g Am and range from 0.0 to 116 and 3.0 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

4.2.14 Metals—heavy non-stainless steel metals (content code 320)

This waste consists primarily of tantalum components such as crucibles, funnels, funnel inserts, and pour-rods. Other metals may include tungsten, platinum, and lead. The average drum weight for this waste is 222 lbs, and the range is 101 lbs to 576 lbs. Plutonium and americium inventories average 30 g Pu and 0 g Am and range from 0 to 183 g and 6.3 g, respectively. No average or range for uranium has been reported. Uranium-235 inventories up to 3 g have been reported.

4.2.15 Glass (content code 440)

This waste consists of glass in the form of sample vials and bottles, lead-taped sample vials, ion exchange columns, dissolver pots, laboratory glassware such as Pyrex flasks and beakers, glovebox windows (glass, Plexiglass, leaded glass), and crushed ground glass. The average drum weight for this waste form is 232 lbs, and the range is 88 to 922 lbs. Plutonium and americium inventories average 5.2 g Pu and 0.0 g Am and range from 0.0 to 182 and 11.0 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

4.2.16 Glass—unleached raschig rings (content codes 441)

This waste consists of borated-glass rings used to minimize neutron multiplication in liquid storage tanks. The average drum weight for this waste form is 194 lbs, and the range is 100 to 563 lbs. Plutonium and americium inventories average 7.9 g Pu and 0.0 g Am and range from 0.0 to 132 and 4.0 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

4.2.17 Glass—leached raschig rings (content code 442)

This waste is the same as content code 440 except that the Raschig rings have been leached. The average drum weight for this waste form is 182 lbs, and the range is 105 to 484 lbs. Plutonium and americium inventories average 2.1 g Pu and 0.0 g Am and range from 0.0 to 49 and 2.0 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

4.2.18 Graphite Molds (content code 300)

This waste consists of graphite molds used in casting plutonium metal. The average drum weight for this waste form is 254 lbs, and the range is 110 to 473 lbs. Plutonium inventories average 9.9 g Pu and range from 0.0 to 61 g. No average or range for americium has been reported. This does not, however, preclude the possibility that americium may be present. Uranium-235 inventories average 3 g and range from 1 to 5 g.

4.2.19 Graphite Cores (content code 301)

This waste is very similar to content code 300, since a graphite core is part of a shaped mold used in casting plutonium metal. The waste in this content code will consist of both graphite molds and cores. The average drum weight for this waste form is 260 lbs, and the range is 164 to 471 lbs. Plutonium inventories average 12.6 g Pu and range from 0.0 to 45 g. No average or range for americium has been reported. This does not, however, preclude the possibility that americium may be present. No uranium has been reported to be present, but small amounts (up to several grams) of ^{235}U can be expected, based on the reported inventories for content codes 300 and 303.

4.2.20 Scarfed Graphite Chunks (content code 303)

This waste is content code 300 graphite which has been scarfed (i.e., cleaned using a rotary-type sanding tool) to remove recoverable plutonium. Use of this content code began in the early 1980s. The average net drum weight for this waste form is 175 pounds, and the range is 29 to 214 pounds. Plutonium inventories average 18.4 g Pu and range from 0.0 to 95 g. No average or range for americium has been reported. This does not, however, preclude the possibility that americium may be present. Uranium-235 inventories average 4.29 g and range from 1 to 9 g.

4.2.21 Firebrick (content code 371)

This waste consists primarily of whole and broken pieces of construction bricks, cinderblocks, and firebrick. The average drum weight for this waste form is 361 lbs, and the range is 102 to 770 lbs. Plutonium inventories average 3.7 g Pu and range from 0.0 to 89 and 2.0 g. No averages or ranges for americium or uranium have been reported. This does not, however, preclude the possibility that americium and uranium may be present.

4.2.22 Plastic and Nonleaded Rubber (content code 337)

This waste consists of various types of plastics such as polyethylene, PVC, Teflon, and nonleaded rubber items. The waste may be in the form of bags, sample vials, bottles, sheeting, and surgeons' gloves. The average drum weight for this waste form is 170 lbs, and the range is 81 to 474 lbs. Plutonium and americium inventories average 0.8 g Pu and 0.0 g Am and range from 0.0 to 49 and 10 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

4.2.23 Blacktop, Concrete, Dirt, and Sand (content codes 374/960)

This waste form consists of blacktop, concrete, reinforced concrete, cinderblocks, bricks,dirt, and sand. Content code 960 was replaced by content code 374 in 1973. The average drum weight for content code 374 waste is 390 lbs, and the range is 125 to 756 lbs. Plutonium and americium inventories average 0.5 g Pu and 0.0 g Am and range from 0.0 to 44 and 1.0 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present. The average drum weight for content code 960 waste is 439 lbs, and the range is 131 to 796 lbs. Plutonium inventories average 0.4 g Pu and range from 0.0 to 137. No averages or ranges for americium or uranium have been reported. This does not, however, preclude the possibility that americium and uranium may be present.

4.2.24 Cemented Filter Media (content code 376)

This waste consists primarily of filter media (pre-1979) and filter media and whole filters (since 1979). The waste also contains limited amounts of insulation waste such as asbestos gloves and fireblankets. Portland cement has been added to all waste packages in order to neutralize any residual nitric acid that may be present. The average drum weight for this waste form is 200 lbs, and the range is 100 to 409 lbs. Plutonium and americium inventories average 22.2 g Pu and 0.1 g Am and range from 0.0 to 189 and 16 g, respectively. Uranium-235 inventories average 14 g and range from 1 to 23 g.

4.2.25 Cemented Resins (content code 432)

This waste consists of anion and cation exchange resins used in the purification and recovery of plutonium and americium. These resins are solidified with Portland cement. The average drum weight for this waste form is 273 lbs, and the range is 101 to 481 lbs. Plutonium and americium inventories average 31.2 g Pu and 0.3 g Am and range from 0.0 to 195 and 4.5 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

4.2.26 Leaded Rubber Gloves and Aprons (content codes 339/463)

This waste consists of leaded glovebox gloves and aprons. It may also contain limited amounts of unleaded gloves, lead bricks, and lead sheeting. Content code 463 was replaced with content code 339 in 1973. The average drum weight for content code 463 is 368 lbs, and the range is 160 to 620 lbs. Plutonium inventories average 14.3 g Pu and range from 0.0 to 57 g. No averages or ranges for americium or uranium have been reported. This does not, however, preclude the possibility that americium and uranium may be present. The average drum weight for content code 339 is 339 lbs, and the range is 130 lbs to 534 lbs. Plutonium and americium inventories average 24.5 g Pu and 0.0 g Am and range from 0 to 98 g and 4.0 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

5. TEST PLAN DESCRIPTION

5.1 Information the INEL Desires to Obtain from the Evaluation

This test plan provides the technology holder the opportunity to demonstrate their assay system in a systematic and defensible manner and provides the INEL with information needed to support management decisions relative to existing assay capability. The INEL desires to (1) identify and assess the state-of-the art in waste assay technology and (2) determine how well newly-developed radioassay systems can meet the specific INEL assay needs and requirements. In particular, the INEL desires to evaluate each system's strengths, weaknesses, and ability to determine the following for the high population accessible waste forms (and, if possible, for some of the lower population waste forms):

- Fissile content
- Total alpha activity
- Isotopics.

In order to accomplish this, method performance data and quality assurance data (i.e., technical performance of the system) must be obtained so that the INEL can evaluate a system's ability to adequately assay the high population accessible waste forms at the INEL and meet the QAO requirements. Specific parameters required to do this include:

- Accuracy
- Precision (reproducability)
- Sensitivity
- Range
- Lower limit of detection
- Total uncertainty
- Throughput.

In addition, the INEL desires to evaluate:

- Each system's ability to determine ability to discern between Pu and (α,n), i.e., a system's ability to obtain an accurate assay for Pu in the presence of high (α,n) activity.

- The effects of the matrix (e.g., moisture, density, heterogeneity, neutron moderating and absorbing material) on the assay results.

- Source position effects on the assay results for various matrix conditions.

Obtaining isotopics is a requirement of the QAPP and, therefore, is of high priority to the INEL, and any complete assay system must have this capability. It is recognized, however, that a technology holder may desire to test and evaluate a system component which does not include this feature. The test bed will therefore accommodate evaluating a partial system.

5.2 Test Period

It is anticipated that the full basic evaluation (which consists of the noninterfering matrix, surrogate, and waste form sets) will require approximately 4-6 calendar weeks. The test period may be abbreviated if a system does not exhibit sufficient performance. However, if a system is performing sufficiently well, the technology holder may have the opportunity to extend the evaluations by an additional 2-4 weeks so that the optional set of surrogates and waste forms can be assayed. In any event, the participant is expected to complete the assay of each set of drums requested by the INEL for the basic evaluation.

5.3 General Test Conditions

The participant will be allowed up to 1-1/2 weeks for system setup, checkout, and calibration checks. Source material for calibration checkout will be provided by the INEL, but no calibration drums will be available. It is expected that the assay method will be at a nearly stand alone stage of development prior to arrival at the INEL and that calibration drums will be provided by the participant.

Each system will be tested over a range of plutonium concentrations of interest to the INEL (i.e., from below 100 nCi/g to 200 g Pu in a drum). Since the INEL must make decisions at the 100 nCi/g level, the system should have a lower limit of detection sufficiently well below 100 nCi/g, so that the accuracy and total uncertainty at the 100 nCi/g level meets QAO requirements. The americium content of the drum will also be varied from 0 to about a gram in order to evaluate the system's ability to discriminate between plutonium and (α,n) components and to provide an accurate plutonium assay in the presence of high radiation levels. The ^{235}U content of a drum will also be varied from 0 to a few grams.

Source material will consist of combinations of plutonium, ^{235}U, ^{238}U, and ^{241}Am. The actual combinations of these source materials will cover the range found in the high population accessible drums.

The tests will be limited to 55-gallon waste drums and 83-gallon overpacks only. No waste boxes or 83-gallon waste drums will be used. The 83-gallon overpacks will consist of a 55-gallon drum placed in an 83-gallon drum, the void being filled with low density packing material.

5.4 Noninterfering Matrix Drum Set

The first set of drums that a participant assays will consist of only noninterfering matrix drums. The INEL has five noninterfering matrix drums which may be used: three contain no matrix and two contain a noninterfering matrix material (i.e., ethafoam). Appendix D contains more details about the noninterfering matrix drums. Source material will be placed in the drums in a configuration known only to the INEL. The purpose of the noninterfering matrix drums is for a check of the system calibration and an initial determination whether the system has capabilities for TRU waste assay (i.e., whether the system can meet the QAOs for accuracy, precision, total uncertainty, etc. for the easiest case). If the system cannot perform adequately on the noninterfering matrix drum test, proceeding to the surrogate or real waste form drums will not yield much benefit and the test will be concluded.

5.5 Surrogate Drum Set

If a system performs adequately on the noninterfering matrix drum set, the evaluation will progress to the surrogate drum set. The objective of the surrogate drum set is to evaluate the system's capability to meet the QAOs for various waste forms, waste form configuration, source configuration, and (α,n) intensities. A variety of surrogate drums (sludge, glass, combustibles, mixed metals, graphite molds, firebrick, and filters) are available. Each surrogate has been carefully designed to simulate a given content code waste form. Details of these surrogate drums are contained in Appendix E.

The basic surrogate drum set will consist of 65 drum measurements as follows:

- Sludge surrogate for content code 001—15 drums containing various amounts of Pu, U, Am, and (α,n) material
- Glass surrogate for content code 440—6 drums containing various amounts of Pu, Am, and (α,n) material
- Glass (raschig rings) surrogate for content code 442—6 drums containing various amounts of Pu, Am, and (α,n) material
- Combustibles (heterogeneous) surrogate for content code 330—8 drums containing various amounts of Pu, U, Am, and (α,n) material
- Combustibles (vermiculite and plastic) surrogate for content code 330—6 drums containing various amounts of Pu, U, Am, and (α,n) material
- Mixed metals surrogate for content code 480/481—6 drums containing various amounts of Pu, Am, and (α,n) material
- Metals (valrath cans) surrogate for content code 480/481—6 drums containing various amounts of Pu, Am, and (α,n) material

- Graphite molds surrogate for content code 300—6 drums containing various amounts of Pu, Am, and (α,n) material

- Firebrick surrogate for content code 371—6 drums containing various amounts of Pu, Am, and (α,n) material

83-gallon overpacks of surrogate drums (i.e., drums from the basic surrogate drum set which have been packed in 83-gallon drums, with the voids filled with noninterfering material)- 6 drums total of various surrogates containing various amounts of Pu, Am, U, and (α,n) material.

The source loadings for these surrogate drums will vary within the following ranges:

Pu: 0 to 200g per drum

Am: 0 to 1/2g per drum

^{235}U: 0 to 3g per drum

^{235}U/Pu mass ratio: 0 to 200

(α,n): 0 to 10^6 n/s per drum.

Although the lower limit for plutonium extends to 0g (i.e., a drum may contain americium or uranium but no plutonium), assay for plutonium will only be evaluated down to approximately 100 nCi/g. The uranium used in the tests will be enriched (i.e., a few percent ^{235}U).

5.6 Waste Form Drum Set

If a system performs adequately on the surrogate drum set, the evaluation will progress to the waste form drum set. The objective of this drum set is to evaluate the system's capabilities for the assay of real waste forms. The systems's ability to handle the problems and situations (e.g., high (α,n) components, non-uniform matrix, variable source material configuration, clumping, etc.) that occur with real waste forms will also be evaluated to determine the applicability of the system to the assay of INEL waste forms. Waste form drums which have been characterized to the extent possible by the INEL will be used for these evaluations. Due to the variable extent to which these waste form drums have been characterized, the exact amount of source material may not be known for each drum. Descriptions of the various waste form drums can be found in Section 4.2 and in Appendix C.

The INEL characteristics reports and Engineering Design Files (EDFs) for the waste content codes of the drums that make up this set will be available to the participants upon request. In addition, the following information about each of the specific waste form drums that is included in the set to be assayed is available to the participant upon request:

- RTR images (if needed by algorithm and requested by participant)

- Weights

- Contact radiation doses.

The basic waste form drum set will consist of 27 drums of the following:

- Content codes 001/002/800: uncemented inorganic sludge—5 drums

- Content codes 003/801: uncemented organic sludge—3 drums

- Content code 004/802: special setups—1 drum

- Content code 007/803/807: cemented inorganic sludge—1 drum

- Content code 292: cemented sludge—2 drums

- Content code 300/301/303: graphite—2 of content code 300, 1 of content code 303

- Content code 330/336: combustibles—2 of each (4 total)

- Content code 440/442: glass—2 of each (4 total)

- Content code 480/481: mixed metals—2 drums.

- Content code 320: heavy, non-stainless steel metals-2 drums.

83-gallon overpacks of waste form drums (i.e., drums from the basic waste form drum set which have been packed in 83-gallon drums)—6 drums total of various content codes

Section 4.2 and Appendix C contain information concerning the range of weights, contents, and source strengths for these content code drums.

5.7 Optional Drum Set

If a system performs adequately on the waste form drum set and if the INEL deems it advantageous to perform further evaluations (e.g., in order to get more information about system capabilities, the system's ability to handle special problems and situations, and system performance for some of the lower population waste forms), the participant will have the option to assay the optional drum set. This drum set will consist of a mixture of (1) surrogates from the surrogate drum set but with differing amounts and distributions of source material; (2) additional waste form drums of the same content codes as in the waste form drum set; (3) waste form drums of content codes not in the basic waste form drum set; (4) additional 83-gallon overpacks (i.e., drums from the basic surrogate and waste form drum sets which have been packed in 83-gallon drums, with the voids filled with noninterfering material); and (5) additional INEL test drums (filled with dry Portland cement and filled with salt [NaCl]).

The waste form drums in this optional drum set which are not in the basic waste form drum set include the following content codes:

- Content code 337: plastic and nonleaded rubber
- Content code 339/463: leaded rubber gloves and aprons
- Content code 374/960: blacktop, concrete, dirt, and sand
- Content code 376: cemented filter media
- Content code 432: cemented resin.

Section 4.2 and Appendix C contain information concerning the range of weights, contents, and source strengths for these content code drums.

The exact number and contents of the optional drum set a participant will be requested to assay is not fixed and will vary, based upon results of tests using the basic surrogate and real waste form drum sets.

5.8 Conduct of Tests

This section contains both general and specific information concerning the conduct of the assay system evaluations.

Test bed evaluations of assay systems will be conducted in a step-wise manner using several sets of drums. An assay system will not progress to a subsequent set of drums unless it performs adequately (as determined by the INEL) on the current set. Once the participants begin assaying the INEL specified noninterfering matrix, surrogate, and waste form drum sets, they must operate their system using their normal operating procedures (i.e., extra long assay times will not be allowed). In general, the participant will be allowed a maximum assay time of 50 minutes (longer assay times can be negotiated). One hour will be allowed for data analysis and workup. Assay results must be turned in to the INEL within one hour after the end of the assay of a drum. These results will be used by the INEL to evaluate the current capabilities of a system.

Since the INEL is interested not only in the current performance capabilities of a system under operational conditions, but also in the ultimate capabilities of a system, a participant will be allowed to submit amended assay results up to a week after an assay is made. This will give a participant time to perform further work on his algorithm, adding any new corrections required by a drum's matrix or source loading. These amended results will be used by INEL to evaluate the future potential of a system.

The first set of drums provided to the participant will be the noninterfering matrix set of drums. If the system does adequately with noninterfering matrix drums (i.e., passes QAO requirements), then the participant will be allowed to progress to the basic set of surrogate drums.

If the system adequately evaluates surrogate drums (i.e., passes QAO requirements), then the participant will be allowed to progress to the basic set of real waste form drums. For each waste form drum, the INEL acquires historical data about each drum, RTR images, and weights. If the participant's algorithm requires any of these data, the participant may request the needed pieces of data prior to assaying a waste form drum.

If the INEL determines that there is value in evaluating the system using the optional set of surrogate and waste form drums and if the participant agrees, then the participant will be invited to assay the optional set of surrogate and waste form drums.

5.9 Assay Data and Results the INEL Requires

The minimum data and information that must be provided to the INEL within one hour after the conclusion of an assay on a drum (either noninterfering matrix, surrogate, or waste form drum) consists of following:

- The general report the participant would normally issue as an assay report, which includes the assay values, isotopics individual uncertainties, total propagated uncertainty, lower limits of detection for source materials (both for those not detected and for the detected ones).

- Specific data (negotiated between the participant and the INEL and based on the participant's method) the participant required to arrive at the general assay report including:

  - Gross count rates

  - Coincidence count rates

  - Gamma spectra.

6. OPERATIONAL CONSTRAINTS AND CONSIDERATIONS

6.1 RWMC Operations Authorization Basis

Prior to accepting a system for technical evaluation at the test bed, reviews to ensure that operation of the system will fall within the test bed's safety envelope and to determine any impacts to the RWMC Operations authorization basis must be performed. A minimum of three months will be required to review the impacts of the proposed technology system to the RWMC Environmental and Safety operations authorization basis. To complete this review the technology holder must submit a complete System Design Description (SDD). If sufficient information is not provided, additional time may be required for the review and subsequent approval before accepting the system to test at RWMC. If the system is outside the envelope of the RWMC Operations authorization basis requiring significant safety documentation or permit changes, extensive time (up to one year) may be required to make the required documentation modifications and get the necessary approvals to initiate facility modifications in preparation for testing the system at RWMC.

The SDD shall contain sufficient information to allow evaluation of impacts to the RWMC Environmental and Safety operations authorization basis, including completion of an Unanswered Safety Question evaluation for compliance with RWMC Safety Analysis Report, completion of a Clean Air Act permit evaluation, completion of an Environmental Checklist evaluation for a NEPA determination, completion of an evaluation for compliance with the RCRA Part B Permit, and completion of an evaluation for compliance with National Emission Standards for Hazardous Pollutants (NESHAPS), etc. Reviews by environmental, radcon, industrial safety and hygiene, fire protection, and quality personnel are required.

The SDD shall include a description of the equipment, detailed operating procedures, site system setup and checkout procedures, system site removal procedures, calibration check requirements, precautions for normal operations, special instructions for abnormal conditions, safety (including radiation, contamination, and fire monitoring and alarms), and drawing lists.

As part of the SDD drawings, schematics, and manuals shall be provided to identify, install, operate, maintain, and repair the system equipment, components, or modules. Catalog and specification data sheets on significant off-the-shelf items used shall be included. These data sheets shall give the name of the manufacturer, catalog figure identification, trade names, performance, and electrical control diagrams if applicable.

The SDD shall also provide details of the assay system requirements, including system physical attributes (size, weight, equipment footprint, floor loading, etc.), operational space requirements, utilities, environment, communications, fire alarms, radiation fields, radiation and contamination monitoring equipment, and consumable support services (liquid nitrogen, diesel fuel).

6.2 Siting Considerations

The location for siting the system for testing and evaluation is limited to existing facilities and available areas within the TSA at the RWMC. In addition, the testing must be located to minimize impact to on-going operations within the RWMC. The most practical available locations for siting a system are within a RCRA compliant storage building or on an asphalt pad out-of-doors. The RCRA compliant storage building is warehouse type building with no heat. The potential siting locations do not have capabilities to handle discharged liquids. As a result, the system should be designed to preclude any liquid discharges (e.g., closed-loop cooling systems should be provided, if required). Due to facility limitations, systems which are mobile, trailer-mounted are best suited for siting at the RWMC. Systems which are mounted on a flatbed trailer should also be acceptable. It is assumed that no physical changes to the building will be required to facilitate the setup, operation, and removal of the system. Each system will need to be evaluated on a case-be-case basis to define a suitable siting location and to determine the impacts to the RWMC Operations authorization basis.

6.3 Safeguards and Security

A description of any radioactive source materials which will be provided by the technology holder as part of the instrument or for calibration checks shall be specified so that any Safeguards and Security evaluations can be performed.

6.4 Conduct of Operations

A detailed operation description of the system shall be provided. The description shall also include technology holder personnel who will be present for operation/data analysis and the functions that they will perform. The technology holder is responsible for the setup/checkouts, operation, data reduction/analysis. The technology holder is also responsible for the removal of test systems after the testing and evaluations are completed. Description of INEL personnel and equipment support required for setup/checkouts, operation, and removal of the system shall also be provided. Activity based schedules shall also be provided for setup/checkouts and removal of the systems, as well as time required for typical assay runs and data reduction/analysis.

6.5 Operational Safety Requirements

Safety of workers shall be maintained by meeting or exceeding industrial safety and health standards, including National Electric Code, American National Standards Institute, Occupational Safety and Health Administration (OSHA) Safety and Health Standards (Title 29 CFR Part 1910 and Title 29 CFR Part 1926).

The system shall be engineered to meet the requirements of the DOE Radiological Control Manual such that personnel radiation exposure is maintained as-low-as-reasonably-achievable. The system shall be engineered to eliminate streaming effects resulting from operation of any radiation generators or radioactive sources.

To the extent practical, personnel safety and property protection, during both normal and abnormal operations, shall be provided by design features not by means of administrative controls; any exceptions shall be described in detail for evaluation.

After setup and prior to initial operation of the system at the test bed, INEL health & safety personnel will inspect the system and review its operation to ensure that INEL health & safety requirements are met.

6.6 Training Requirements

Access to the RWMC is controlled. These controls are in place to protect the worker by ensuring worker awareness through training and to maintain compliance with laws, regulations, and permits. The training requirements listed below are the minimum requirements for unescorted entry into the Transuranic Storage Area of the RWMC. Certain operations may require additional training, such as specified in job-specific permits and procedures (e.g., Safe Work/Radiological Work Permits, Technical Procedures, etc.). The minimum requirements for unescorted access requirements into the TSA of the RWMC include: Health and Safety "Blue Card" or Construction "Orange Card" training; RWMC Access training; Radiological Worker I for general entry and handling of sealed radioactive sources or Radiological Worker II for entry into high radiation areas, contamination areas, high contamination areas, and airborne radioactivity areas; 24 hour Occupational Safety and Health Act (OSHA) training, RCRA training, and Emergency Preparedness training. In addition, in order to be allowed to handle fissile material, Fissile Material Handler training is required.

6.7 RWMC Operating Schedule

RWMC normal operating hours are from 7:00 AM to 5:30 PM, Monday through Thursday. Any operation of equipment during off-hours will require prior approval and the addition of RWMC personnel coverage. All drum handling (including noninterfering matrix, surrogate, waste form, and all of the participant's calibration drums) must be performed by INEL personnel. In addition, unless participant personnel have the required training, all handling of radioactive sources and fissile material must be performed by INEL staff.

7. EVALUATION OF TEST RESULTS

The objective of the INEL is to perform an impartial evaluation of the ability of a waste NDA system with respect to the INEL's high inventory, accessible waste forms. This involves assessing the system's ability to meet TRU Waste Characterization QAPP requirements for these waste forms and determining if an assay system can help INEL with any waste stream or with any existing or anticipated problem area. For example, the ability to handle waste forms with the following problem areas will be evaluated:

- High (α,n) activity
- Variable source chemical composition
- Variable radionuclide composition
- Clumping of source material
- Inhomogeneous matrix, including source location, voids, and variable fill height
- High and variable moderator/absorber content
- Very high count rates.

In addition, since segregation is of prime importance to the INEL, the system's ability to segregate at the 100 nCi/g level will be evaluated.

For drums where the INEL knows the contents (e.g., non-interfering matrix and surrogate drums), the evaluation will follow a modified PDP evaluation method (but with no replicates). The results provided by the participant's system will be compared to the known values. Special attention will be paid to the uncertainties and the method for determining them.

The set of real waste form drums will be selected based on the INEL's ability to bound the inventory. Results obtained by the system will then be evaluated relative to these bounds.

This evaluation will involve the following:

- Compare results with known (or known range of) values
- Compare uncertainties with QAPP requirements
- Compare lower limits of detection with QAPP requirements (i.e., assess the ability to segregate at the 100 nCi/g level)
- Assess a system's ability to accurately assay a given waste form, such as sludges, combustibles, etc.

- Assess the ability to provide an accurate assay in the presence of varying source strengths, locations in a drum, nonhomogeneous matrices

- Assess the ability to adequately handle waste form problem attributes, such as high (α,n), fissile material clumping, high density matrix, etc.

At the end of each series of assays (i.e., at the end of the non-interfering matrix drum, basic surrogate drum set, basic waste form drum set, etc.), a preliminary report will be given to the participant. A more detailed preliminary assessment will be available one week after completion of the assay of a drum set. A final report concerning the evaluation of an assay system will be produced within 1-2 months after a system leaves the test bed. This final report will include evaluations of the current capabilities of the system (obtained by evaluation the participant's results submitted within one hour after completion of the assay) together with the INEL's assessment of the future capabilities (obtained from the participant's results submitted up to one week after the assay). It only report results and will not contain any suggestions or corrective actions.

The INEL's preliminary evaluation of the participant's data (prior to the official issuance of the INEL report) is only for use by the participant and will not be disseminated.

The evaluation that the INEL performs is not intended to be a certification or qualification process for the NDA system. Therefore, the participant cannot claim that its system "passed an INEL qualification."

The data the participant supplies to the INEL will be for the laboratory's own private use. Any public release of this information will have to be agreed upon by both the INEL and the participant before release. Any restricted data or information supplied to a participant (e.g., restricted information concerning drum contents, INEL or Rocky flats site operations, etc.) will be clearly identified as such.

8. REFERENCES

1. U.S. Department of Energy, *Transuranic Waste Characterization Quality Assurance Program Plan*, CAO-94-1010, April 30, 1995.

2. Matrix Parameter Category Groups, reference EDF-RWMC-805, INEL-95/029, 12/13/95.

3. T.L. Clements, Jr., "Content Code Assessments for INEL Contact-Handled Stored Transuranic Wastes," EG&G Idaho, Inc., WM-F1-82-021, October 1992.

4. "Idaho National Engineering Laboratory Code Assessment of the Rocky Flats Transuranic Waste," WASTREN, Inc., INEL-95/0281, July 1995.

5. Idaho National Engineering Laboratory, "Drum Neutron Counter Chamber and Detector Configuration," EDF# RWMC-606, April 26, 1993.

6. E.W. Killian and J.K. Hartwell, "VAXGAP: A Code for the Routine Analysis of Gamma-Ray Pulse Height Spectra on a VAX Computer, EGG-2533, May 1988.

7. U.S. Department of Energy, "Waste Acceptance Criteria for the Waste Isolation Pilot Plant," DOE/WIPP-069, Revision 5, April 1996.

8. U.S. Department of Energy, "Safety Analysis Report for the TRUPACT-II Shipping Package (SARP)," U.S. NRC Docket No. 71-9218.

Appendix A

Section 9 of National TRU Waste Characterization Program QAPP

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9. NONDESTRUCTIVE ASSAY

Numerous RA techniques are available to determine the TRU content of bulk waste. RA methods may include both nondestructive and destructive techniques.

Nondestructive assay (NDA) techniques allow an item to be assayed without altering its physical or chemical form. NDA techniques can be classified as active or passive. Passive NDA is based on the observation of spontaneously emitted radiations created through radioactive decay of the isotopes of interest or their radioactive daughters. Most active NDA is based on the observation of gamma or neutron radiation that is emitted from a target isotope when that isotope undergoes a transformation resulting from an interaction with stimulating radiation provided by an appropriate, external source.

Destructive RA refers to the radiochemical analysis of a representative sample collected from the waste. The sample is physically and/or chemically processed for subsequent analysis by radioactivity counting or other instrumental techniques. Radiochemistry methods will be discussed in a future revision of the CAPP. Throughout this section, references to "RA measurement" systems shall include only NDA systems.

NDA methods can not directly identify and quantify all the individual radionuclides of interest. Therefore, some NDA techniques are commonly used in conjunction with isotope ratio calculations using data from other sources. Destructive RA techniques are used to directly quantify the radioisotopic content of identified, homogenous waste streams. Any NDA, destructive RA, or combination of these methods are acceptable as long as they address and achieve the QAOs of the Program. The selected methods may incorporate supporting data from acceptable knowledge, such as isotope ratios of scaling factors, when such data can be supported by auditable QA records.

It is not intended that the QAOs contained in this document be interpreted as being the only criteria for establishing acceptability of NDA measurement systems. The QAOs published in this document for NDA systems are used to establish minimum performance requirements for measurement systems used to generate waste characterization data for the Program. Parties responsible for determining the acceptability of NDA measurement systems for purposes other than TRU waste characterization for WIPP may establish requirements in addition to or in lieu of the QAOs for this Program. Such requirements do not affect the obligation to meet the QAOs of this Program for systems generating waste characterization data for WIPP.

For the purposes of the Program, two parameters describing the waste must be known; the total alpha activity and the activity of the individual isotopes present. The total alpha activity is a controlling variable for the amount of radiolysis and associated radiolytic gas generation. The activities of individual isotopes are needed to determine fissile gram equivalent and to perform other required calculations. If a waste stream may be contaminated with radioactive materials of variable or unknown isotopic ratios. This does not preclude the use of acceptable knowledge for the determination of isotope ratios at some facilities, but does require that the bases for the isotope ratios which are used be documented and supportable. The measurement of total alpha activity and independent determination of isotopic ratios, obtained by nondestructive and/or destructive RA or acceptable knowledge, are considered adequate for use in the Program.

9.1 Quality Assurance Objectives

Each participating site must use one or more RA techniques. Each site shall demonstrate and technically justify that the RA techniques used are appropriate for the specific wastes to which they are applicable. The rationale for using a specific assay technique should include the physical form of the waste, the radionuclide content, and the waste generating process. In all cases, the total uncertainties in the assay must be calculated using the terms derived for compliance with the QAO for Total Uncertainty and reported with the data. The actual precision and accuracy values obtained for waste containers will be a function of the waste type, total TRU content, its distribution, and characteristics of the measurement instrumentation. The QAOs for precision, accuracy, minimum detectable concentration (MDC), completeness, and total uncertainty are summarized in Table 9-1. The QAO parameters are defined for the general case in Section 3.2. QAOs for NDA are specified over several different ranges of interest. These ranges are somewhat arbitrary but convenient divisions which are expected to have differing contributions to inventory or are related to significant cutoffs (e.g. for shipping). Participating sites need only demonstrate for individual measurement systems that the QAOs can be achieved for the respective ranges over which that system will be used. Additional details on the individual QAO parameters are given below.

Table 9-1. Quality assurance objective for nondestructive assay.

Range of waste activity in α-curies[a]	Nominal compliance point α-curiesa (g WG Pu)[b]	PARAMETER				
		Precision[c] (%RSD)	Accuracy[d] (%R)	Total uncertainty[a]	Completeness (%)	MDC (nCi/g)[g]
0	0					60
>0 to 0.04	0.008 (0.1)	≤20	75-125	Low 40% High 175%	100	
>0 to 0.04	0.08 (1.0)	≤15	50-150	Low 30% High 200%	100	
>0.4 to 4.0	0.8 (10)	≤10	50-150	Low 30% High 200 %:	100	
> 40.0	12.8 (160)	≤5	75-125	Low 50% High 150%	100	

a. Applicable range of TRU activity in a 208-liter (55-gallon) drum to which the QAOs apply, units are Curies of alpha-emitting TRU isotopes with half-lives greater than 20 years.

b. The nominal activity (or weight of Pu) in the 208-liter (55-gallon) drum used to demonstrate that QAOs can be achieved for the corresponding range in column 1, values in parentheses are the approximate equivalent weights of weapons grade plutonium (WG Pu), fifteen years after purification; for purposes of demonstrating QAOs, "nominal" means within ≠10 percent.

c. ≠ one relative standard deviation based on fifteen replicate measurements of a non-intertering matrix

d. Ratio of measured to known values based on the average of fifteen replicate measurements of a non-interfering matrix see Section 9.8 for additional details

e. 95-percent confidence bounds of all propagated uncertainties (Confidence bound divided by true value, expressed as a percent).

f. Valid radioassay data is required for all waste containers, see Section 9.8 for additional details

g. As defined in Sections 9.1 and 9.6

Precision

The precision of each measurement technique must be determined through replicate processing of a waste container containing a known quantity of the radioactive material of interest. The specific method for demonstrating compliance with the QAO for precision in RA is described in detail in Section 9.6.

Accuracy

Accuracy is determined through replicate processing of a waste container containing a known quantity of the radioactive material of interest. Accuracy is calculated from the ratio of the mean measured estimate to the known value for an accepted calibration or verification standard. Calibration standards are those used to determine the response characteristics of a measurement system. Verification standards are used to test the validity of a calibration independently of the original calibration standards. Whenever possible, both radioactive calibration and verification standards shall be obtained from sources which maintain measurement systems traceable to NIST. Evidence of such traceability and certificates for individual standards shall be obtained from the standards suppliers. The specific method for demonstrating compliance with the QAO for accuracy in RA is described in detail in Section 9.6. The bias of a RA technique or measurement system is defined as the systematic error component of the total uncertainty. The systematic error is constant for the test or test conditions. For the Program the determination of accuracy is also an estimate of the bias of a measurement system.

Sensitivity limits

Discrimination between LLW and TRU wastes for the Program may only be made with systems for which adequate sensitivity limits have been documented. The ability to achieve the required detection limit in Table 9-1 must be demonstrated for each specific waste type/method combination planned for use in the Program.

For the Program, detection limits will be defined to be that level of radioactivity which, if present will yield a measured value less than the critical limit with 5-percent probability. The critical limit is defined as that value which measurements of the background will exceed with 5-percent probability.

Minimum Detectable Concentration

The detection limit used in the Program is the MDC. This concept corresponds to a level of activity that is practically achievable with a given instrument, analytical method, and analyze/matrix combination. The MDC considers not only the instrument characteristics (background and efficiency), but all other factors and conditions which influence the measurement. It is an a *priori* (before the fact) estimate of the activity concentration that can be practically achieved under a set of typical measurement condition. These would include the waste quantity, counting time, matrix specific corrections, decay corrections, and any other factors that comprise the activity concentration determination. It is useful for establishing that some minimum overall measurement conditions can be met. Any of several factors under operator control could be varied to obtain the required MDC.

The MDC is an a *priori* estimate of the detection capabilities of a given measurement system and method. It is based on the premise that from a knowledge of the background count and other measurement system parameters, an a *priori* limit can be estimated for a particular measurement.

The MDC is defined on the basis of statistical hypothesis testing for the presence of activity. This approach is common to many authors and has been described extensively (Currie 1968; EPA 1980).

The derivation will not be repeated here, however, the MDC may be calculated from:

$$MDC = K_1 K_2 \ (2.71 \ + \ 4.65 \ =_3) \qquad (9\text{-}1)$$

Where

K_1 is the proportionality constant relating the detector response (counts) to the activity, such as, $K = 1/\alpha$ where α is an overall detection efficiency, or $K = 1/1_r \alpha_r$ where 1_r is the gamma ray-emission probability per decay and α, the detection efficiency for the gamma ray;

K_1 is the factor which relates the total activity determined by the measurement system to an activity concentration in waste under a given set of measurement conditions, for example, the weight of waste assayed and a self-absorption correction;

s_b is the standard deviation of the background.

This equation incorporates the following assumptions:

- The preselected risk for concluding falsely that activity is present above the critical level (α) and the predetermined degree of confidence for correctly detecting its presence above the critical level ($1-\beta$) are 5 percent and 95 percent, respectively
- In the vicinity of the MDC, the gross measurement counts and background counts will be approximately equal.

This equation represents the simplest case. Alternate equations have been described for multi-component and spectrometry based systems (Pasternack and Harley 1971; Fisenne et al. 1973). Sites may propose calculational bases more appropriate to their measurement systems. Such alternate methods must be described in SOPs and incorporate the same risks of false detection and false non-detection as are described above. Calculations used to demonstrate attainment of the QAO for the MDC should use typical or average values for the parameters comprising K_2 in Equation 9-1. The specific method for demonstrating compliance with the QAO for MDC in RA is described in detail in Section 9.6.

Total Uncertainty

Total uncertainty includes propagated uncertainty for all corrections and factors applied to the analysis of real wastes to compensate for inhomogeneities and matrix interferences. The ability to achieve the QAO for total uncertainty is not demonstrated solely from specific measurements. The ability to achieve this QAO will be determined from an evaluation by an expert review team of the propagation of all uncertainties as documented by the site. The QAOs for total uncertainty are expected to be achievable in the presence of backgrounds generated by alpha and gamma emitting sources and in the presence of interfering quantities of neutron and gamma absorbing and moderating material, as is the case for much of the waste encountered in the Program. The specific method for demonstrating compliance with the QAO for total uncertainty in RA is described in detail in Section 9.6.

Completeness

Acceptable RA data shall be obtained for 100 percent of the waste containers characterized for disposal. Acceptable radioassay data shall consist of data on the radioactivity content of the waste package obtained from measurement systems which have been demonstrated to have met all the relevant QAOs for radioassay. RA data shall be validated according to the requirement in Section 9.6 prior to shipment of the waste to WIPP.

Comparability

For purposes of the Program, when multiple systems are planned for use in determining the same or comparable parameters, the participating sites shall perform multiple, independent RAs of a sample of waste containers. Data from these multiple, independent radioassays shall be reported to CAO in the semi-annual QA reports in accordance with Section 2.2 of this QAPP as evidence of method comparability.

Regardless of the number and type of RA methods in use, each site shall participate in relevant interlaboratory comparison programs. In this context, "relevant" means the measurement in any environmental or waste media of any parameter required in the waste characterization program using a measurement system or method planned for use in the waste characterization program. Data from such programs shall be reported to CAO for evaluation. Where existing programs are inadequate, modified or new programs will be developed to ensure that an appropriate program is available for each general class of RA.

9.2 Methods Requirements

Any RA method may be used as long as the documented performance characteristics of the method meet the program QAOs. Only systems being used for discriminating TRU from LLW must meet the QAO for MDC. When waste concentrations significantly exceed the LLW/TRU cutoff, operator controlled parameters (e.g., counting time) may be modified within preestablished limits as long as QAOs for precision continue to be met.

This section describes certain general provisions which will be applicable to all types of radioactivity measurements performed under the Program. Performance of software controlling the measurement process and analyzing data shall be demonstrated and documented in accordance with ASME NQA-1, Element 11, Supplement 11S-2 (ASME 1994). Performance may be demonstrated by the use of test problems and\or in the context of testing the performance of the measurement system with QC samples. Software testing must cover the full range of expected applications of the system.

NDA Methods

A variety of NDA technologies may be effective in meeting the requirements of the Program. Table 9-2 identifies a number of such instrument systems which are in use at various DOE and/or contractor testing facilities. The list is neither complete nor limiting and is meant to illustrate the breadth of choice available. QAOs for the project may be met with the listed systems or by modifications, functionally equivalent alternatives, multiple combinations, or hybrid or the systems. The following discussion is intended to provide clarification of the table entries.

Whenever applicable, the assay procedures cited in ASTM (1989a), ASTM (1989b), ASTM (1991b), ASTM (1992), and NRC standard practices and guidelines (NRC 19984) are recommended for use at all testing facilities. These procedures require the use of proper calibration standards, proper equipment and equipment setup, avoidance of practices (such as misalignment of the waste package) known to result in inaccurate assays, attention to proper record-keeping and equipment maintenance, and safe operation of the equipment.

NDA SOPs must instruct operators to perform all necessary background and performance checks prior to performing any assays of waste containers. These performance check data must be checked against predetermined acceptance criteria. If any criterion is not met, remedial action must be taken. Each site must include or reference in SOPs its method for determining and recording the acceptance criteria. The remedial action may include a repetition of the background and/or standards measurements. The disposition and use of any TRU waste assays performed during a period ending with a suspect performance check or during any resulting investigation or remedial action must be documented and justified.

Table 9-2. NDA methods for potential use for TRU waste assay.

Types of measurements	Methods
Gamma-Ray measurements	High resolution spectro copy (Intrinsic Germanium) transmission corrected gamma-ray measurements • Segmented gamma-ray scanner • Computed tomographic gamma ray scanner
Passive neutron measurements	Shielded neutron assay probe totals Counter passive neutron coincidence counter Advanced Matrix Corrected Passive Neutron Counter (Add-A-Source)
Passive/action neutron measurements	Am-U Source Driven Coincidence Counter Callfomium Delayed-neutron Counter (Shuffler) Neutron generator differential die-away counter Combined thermal/epithermal neutron counter
Thermal neutron capture	Callfomium delayed-neutron counter Neutron generator differential die-away counter Combined thermal/epithermal neutron counter

SOPs for NDA systems must contain all necessary instructions for the operation of computerized data acquisition systems. Such software instructions shall include explanations of required input, options, and prohibitions for operators when exercising any interactive portions of the software.

Regardless of source, the procedures are subject to the following provisions:

- The procedures must be codified in the facility as SOPs which have been written, approved, and controlled under the provisions of the site QAPjP or a QA program with equivalent provisions for procedural control.

- The procedures must have been internally demonstrated in the facility and have documented performance characteristics which meet the QAOs of this program.

9.3 Quality Control

RA is a quantitative measurement of key radioactivity parameters of the contents of a waste container. NDA systems must be checked through the use of calibration check and background waste containers as well as replicate determinations. As discussed in this section, routine performance checks shall be performed on all RA systems according to approved SOPs. All RA systems shall be operated in statistical control as determined by the control limits established by these site SOPs.

Each participating site must perform, and report in its semi-annual management reports to CAO, all required instrument performance parameters for each instrument used to perform measurements intended for use in the Program. MDCs for system used to distinguish between LLW and TRU waste must meet the QAO specified in Section 9.1.

If any QC measurement fails to meet Program criteria, the analytical measurement may not be continued prior to taking appropriate corrective action. This section outlines the minimum QA/QC operations necessary to satisfy the analytical requirements of the Program.

9.3.1 Measurement System Checks

This section discusses additional QC testing for radioactivity measurement systems. It includes calibration and routine performance testing requirements used to ensure that measurement systems are in control and meet the performance specifications established for that measurement systems to demonstrate compliance with the QAPP QAOs.

Instrument Calibration

Specific guidelines for instrument calibration are given in Section 9.5. Instruments must be calibrated at the frequencies specified in Section 9.5.

Instrument Performance Checks

Although the efficiency factors vary for every sample geometry, radiation counting systems are in a sense "blind" to the conditions outside the detector which produce the radiation being measured. Because of this it is usually possible to verify the proper function of the instruments with rugged, long-lived sources. Since the data obtained for these "check" sources is not directly used to calculate analytical data, they do not have to ne NIST traceable, but only need to be adequately characterized for the proposed usage. The principal requirements for such sources are that they be long-lived, simple to reposition with respect to the detector(s), of sufficiently high activity to obtain adequate counting statistics in short count times, and relatively insensitive to handling.

Each of these conditions contributes to a situation where the sources can be easily and quickly counted. If long-lived and rugged, the sources' data should vary slowly with time in an easily predictable manner. For each instrument system used in radioactivity analyses, routine performance checks of efficiency, background, and energy resolution (for spectrometry systems) shall be performed. Data shall be logged, plotted on control charts and compared to preset control limits. These data shall be delivered with the analytical data, covering the time period over which the analyses were actually performed. Performance checks for non-spectrometric instruments shall include

- Efficiency checks
- Background checks

Performance checks for spectrometric instruments must also include

- Energy calibration checks
- Energy resolution checks

Except for system backgrounds, instrument performance checks shall be performed and documented at least twice each shift. These checks shall be performed prior to any actual waste measurements on each work shift and after completion of all waste measurements for the shift. When shift operations are continuous or overlapping, the performance checks for the end of the shift completing work can be the same performance checks as those done at the beginning of the shift starting work. This procedure verifies acceptable performance of the measurement system.

The required frequencies for background measurements will be a function of the variability of the background signal and the analytical use of the background data. Backgrounds acquired over long count times, with low variability, and not used directly in the processing of analytical data need not be counted daily. Backgrounds used directly in the analytical data calculations must be counted on a frequency consistent with the potential variation of the background signal and the performance of the analytical measurements with which the backgrounds are associated. Site SOPs shall indicate the frequency of background measurements for each measurement system used in the Program.

Replicate Counts

Independent replicate measurements, at least duplicates, must be performed on 10 percent of the waste containers in accordance with the QAPjP and SOPs.

9.3.2 Intercomparison Programs

Most QC measurements take place in a closed system within a laboratory or measurement organization. Intercomparison programs provide a mechanism for comparing laboratory performance with that of other organizations performing measurements for the same analytes under comparable conditions. Participating RA testing facilities may possess neat identical systems or may have significant differences, including operation under differing calibration regimes or utilization of systems with entirely different measurement principles.

Sites using NDA methods shall participate in any measurement comparison program(s) sponsored or endorsed by the NTP team leader. Such programs may be conducted as part of the PDP, through the NDA/NDE Interface Working Group (IWG), and/or through other third parties.

9.3.3 NDA Operator Training

Present-day NDA units are highly automated, computer-based systems. The instruments are computer-controlled using interactive software. Only trained personnel shall be allowed to operate the assay equipment. Standardized training requirements for RA operators must be based upon existing industry standard training requirements of ASME NQA-1, Element 2, with the exception of Supplement 2S-2 (ASME 1994). Requalification of operators must be based upon evidence of continued satisfactory performance and must be done at least every two years. Unsatisfactory performance shall result in disqualification of the operator. Retraining and demonstration of satisfactory performance are required before an operator is again allowed to operate an RA system.

9.4 Instrument Testing, Inspection, and Maintenance Requirements

RA measurement systems must be calibrated and maintained in accordance with controls established and implemented in the site QAPjPs and SOPs, respectively. SOPs must cover the routine system calibration, performance checks, and operation of the system. For any types of RA systems which are addressed by ANSI, ASTM or other consensus standards, the site SOPs must be consistent with all relevant provisions of these standards.

9.5 Calibration Procedures and Frequencies

All radiation measurement instruments must be calibrated for the specific analysis of interest. This involves the determination of the counting efficiency or some other form of response factor. Because counting efficiencies and response factors may very with the isotope of interest, mode of decay, energy of decay, presentation geometry, and many other parameters, a unique calibration is required for each type of analysis system. Each counting system must be subjected to a complete calibration appropriate to its planned usage and based on applicable consensus standards such as those published by ASTM. Each calibration must be fully supported with records which can be tracked to standards obtained from suppliers maintaining measurement systems traceable to NIST. Once established, the calibration is valid until a preset time limit has been exceeded or the instrument fails other performance checks. Complete verification of calibration of NDA for at least one counting geometry/sample matrix combination must be repeated at least annually.

Primary calibration standards shall be obtained form NIST, the New Brunswick Laboratory, or from suppliers maintaining measurement systems traceable to NIST whenever such standards are available. When standards are not available from such suppliers, the actual standards used shall be calibrated against primary standards obtained from NIST or from suppliers maintaining measurement systems traceable to NIST. The documentation of this cross-calibration shall be retained as a QA record. Working calibration standards shall be prepared using isotopes, geometries, and matrices having characteristics as close as possible to those expected for actual samples without compromising the quantitative integrity or homogeneity of the standard.

The range of applicability of system calibrations must be specified in site SOPs. If assay measurement values fall outside the applicable range, assay measurements must be repeated on alternate measurement systems covering the required range or other appropriate corrective actions must be taken and documented.

The commonly accepted techniques of transmission and live-time corrections to compensate for matrix variations present within a container are acceptable for the NDA techniques. Computer programs used to calculate activities of radioisotopes may use correction algorithms to compensate for some waste characteristics such as waste density, gamma absorption, neutron moderator, and neutron absorption indices. Calibration of RA measurement systems which utilize such correction factors shall include the determination of calibration factors and functional relationships to other waste parameters as part of the system calibration. Each site must determine and document the range of waste types to which it will apply any given calibration and set of correction factors.

All computer programs and revisions thereof shall be documented, verified and validated as required by ASME NQA-1, Element 11 and Supplement 11S-2, "Supplement Requirements for Computer Program Testing," (ASME 1994) before initial use for production of analytical data. Verification shall include both verification of the algorithm used and test runs of the program comparing the program output to true values. Test runs shall exercise all default and boundary values of parameters. Programs shall be documented in accordance with *Standard for Software User Documentation (ANSI 1987)*. Documentation of computer programs shall include, at a minimum

- Program name
- Revision number
- Revision date
- Author(s)
- Program application
- Programming language (including version numbers of all compilers, linkers, etc.
- Operating system
- Required hardware
- Descriptions of algorithms used
- User's manual
- Listing of Code
- Examples of input and output forms
- Results of test cases
- Copies of external data files
- Lists of default parameters
- Records of review and approval

Individual(s) responsible for the following functions must be identified:

- System operation and maintenance, including documentation and training
- Database integrity, including data entry, data updating and QC
- Data and system security, backup and archiving

All RA equipment shall receive routine performance checks for such parameters as system counting efficiency and system background. Spectrometry based systems shall also receive routine performance checks for energy calibration and resolution. Routine performance checks shall be performed with check sources which are stable and constant or which change only by well-established and predictable quantities (e.g., radioisotope decay). Site SOPs for performance checks shall state the standards used, frequencies for each test, record keeping, control limits, and corrective actions to be taken when the control limit is exceeded. Control charts (e.g., based on acceptable ranges or variances) shall be used to track trends in the parameters measured in the performance checks. Performance checks shall be performed and documented at least twice each shift. These checks shall be performed prior to any actual waste measurements on each work shift and after completion of all waste measurements for the shift. When shift operations are contiguous or overlapping, the performance checks for the end of the shift completing work can be the same performance checks as those done at the beginning of the shift starting work.

9.6 Data Management

The results of RA for each waste container must be documented and available to the data user. Requirement for RA data reduction, validation, and reporting are presented below.

Data Reduction

The reduction of RA data may be accomplished using computer software that is specifically designed for the particular assay being performed. The software may vary from site to site. This software and/or other data reduction procedures must be specified in site QAPjPs and supporting SOPs.

Although generalized equations containing parameters commonly used to calculate the radioactively in a given analysis can be written, not all parameters will be used in every technique. Additional or more complex calculations may be required for methods involving multiple measurements, the analysis of spectrometric data, or active interrogation techniques. The exact algorithms used by each site must be contained in the site-specific technical documentation.

Data Validation

All RA data must be reviewed and approved prior to being reported. The validation process is outlined in Section 3.0 and includes verification that the QAOs in Table 9-1 have been met. The demonstration that QAOs have been met for specific measurement systems need only be made for the ranges in Table 9-1 for which the measurement system will actually be used. These demonstrations will be made with all instrument control parameters set at specific values (e.g., a specific count time). The values for all parameters critical to the demonstration the QAOs have been met must be maintained the same for actual waste measurements as were used for the QAO demonstration. The following discussion provides details for demonstrating compliance with the QAOs for precision, accuracy, MDC, and test uncertainty.

Sites shall demonstrate compliance with the QAO for precision by replicate processing of a waste container (208-liter [55-gallon]drum) containing the quantities of TRU isotopes indicated in Table 9-1 for each range for which the measurement system is to be qualified. The activity shall be distributed in a well-characterized, non-interfering matrix and shall not be one of the standards used to calibrate the counting system. A total of fifteen replicate counts shall be obtained with removal of the waste container from the measurement system and reinsertion of the waste container into the measurement system between measurements. The precision shall be computed as the %RSD of the distribution of these replicates as defined in Equations 3-2 and 3-3.

For systems using smaller volumes than the standard 208-liter (55-gallon) drum, the activity used shall be proportional to the concentration obtained by having the TRU activity distributed in a 208-liter (55-gallon) drum. Sites using destructive RA shall demonstrate compliance by carrying 15 replicates through the entire analytical process.

Sites shall demonstrate compliance with the QAO for accuracy by replicate processing of a waste container (208-liter [55-gallon] drum) containing the quantities of TRU isotopes indicated in

Table 9-1 for each range for which the measurement system is to be qualified. This activity shall be in the form of a verification standard, that is, it shall be characterized as well as the calibration standards described in Section 9.5 but is may not be one of the calibration standards nor shall it be distributed in a well-characterized, non-interfering matrix and shall not be one of the standards used to calibrate the counting system. A total of fifteen replicate counts shall be obtained with removal of the waste container form the measurement system and reinsertion of the waste container into the measurement system between measurements. The accuracy shall be computed as the %R of the known value as defined in Equation 3-5. When using Equation 3-5, C_m is the average result of the fifteen replicate determinations and $C_{???}$ is the known value for the waste container used in the measurements.

For systems using smaller volumes than the standard 208-liter (55-gallon) drum, the activity used shall be proportional to the concentration obtained by having the TRU activity distributed in a 208-liter (55-gallon) drum. Sites using destructive RA shall demonstrate compliance by carrying 15 replicates through the entire analytical process.

Sites may demonstrate compliance with the QAO for MDC by replicate processing of an appropriately sized waste container containing only a well-characterized, non-interfering matrix with no added activity. A total of fifteen replicate counts shall be obtained with unloading and reloading between replicates. Sites may propose alternate methods for determining the variance of the background for specific measurement conditions. Any such alternate methods must be fully justified and demonstrated to be more appropriate to the measurement system and specific conditions for which it is proposed. The MDC shall be computed using the variance of the background count and Equation 9-1 or the analogous computation using all parameters appropriate to the measurement method.

The QAO for total uncertainty is intended to include estimates of the cumulative uncertainties from all correction factors and adjustments which are applied to the analytical data to compensate for inhomogeneous distribution, shielding, self-absorption, attenuation, and other matrix effects. Such methods may be unique to measurement systems, waste types, and sites. They may incorporate data from other measurements, be computed from information obtained from the measurement itself, or be average factors obtained from experiments. Uncertainties in any parameters influencing the computation of radioactivity content must be included in the calculation of total uncertainty. This specifically includes parameters such as the isotopic ratios when assumed to be constant or determined from data sources external to the actual measurement. Demonstration of compliance with the QAO for total uncertainty is not obtained solely from measurements on the non-interfacing waste matrix although the results of such measurements may be used to estimate some of the contributing parameters.

To demonstrate compliance each site must document all such applied factors; their derivation, source or justification; the range of waste measurements to which they will be applied; and, the uncertainty associated with each factor. The uncertainties at the 95-percent confidence level from all correcting factors shall be propagated along with estimates of the uncertainty due to any other source of precision error. It is this value which must meet the QAO for total uncertainty. This demonstration must be available for each different set of correction factors as applied to different waste types (ANSI/ASMS 1985).

It is anticipated that compliance with this QAO will be evaluated in on-site reviews of the compliance packages by a team of knowledgeable experts in the field. The most probable source of members for the review teams is the existing NDA/NDE IWG and their associated staff. These individuals have the required theoretical and practical expertise as well as backgrounds in the TRU waste characterization area. Members of the IWG who are measurement staff at one site may serve as expert reviewers of measurement systems at other sites but not at their own site. The makeup, selection method, and role of the review team will be defined in a QA procedure by CAO. CAO will oversee the formation of this independent review team. This will provide the fairest possible evaluation of the site's compliance data.

Data Reporting

The results of RA must be documented and available to data users. RA testing facilities must retain all raw data in sufficient detail and with adequate support documentation to repeat all calculations as necessary. If activities of isotopes other than the nominal isotopes of interest are detected by an actual waste measurement, the activity of each of these isotopes must be reported as part of the waste assay for that container.

RA testing data must be reported to the site project office on a testing batch basis. A testing batch is a suite of waste containers undergoing RA using the same testing equipment. A testing batch can be up to 20 waste containers without regard to waste matrix.

Each RA testing facility is required to submit testing batch data reports for each testing batch to the site project office on approved standard forms. Site-specific documentation must include example forms that will be used for reporting. RA testing batch data reports shall consist of the following:

- Cover page that includes testing facility name, testing batch number, drum numbers included in that testing batch, and signature releases of RA testing personnel as described in Section 3.1.1

- Table of contents

- Data review checklists for each testing batch verifying that the data generation level review as described in Section 3.1.1 has taken place. Checklists must contain tables showing the results of the testing batch QC samples

- Separate testing report sheet(s) for each sample in the testing batch that includes

  - Title "Radioassay Data Sheet"
  - Methods used for NDA (i.e., procedure identification)
  - TRUCON code, Item Description Code, matrix parameter category, as applicable
  - Date of NDA examination
  - Total Pu-239 fissile gram equivalents (g) and associated uncertainty
  - Total alpha activity and associated uncertainty (Curies)
  - TRU activity and associated uncertainty (nCi/g)
  - Listing of individual radioisotopes present (Curies) and associated uncertainty (Curies)
  - Thermal power and associated uncertainty (W)
  - QC replicate (yes/no)
  - Operator signature/date
  - Reviewer signature/date

All associated uncertainties shall be reported at the 95-percent confidence level. A form containing all the information specified above must be completed and signed. Figure 1-5 indicated how the NDA data form should travel through the waste characterization process. In addition, RA testing facilities located on sites shall maintain the following items in their files, documented and retrievable by testing batch number. Contract RA testing facilities shall forward these items along with testing batch data reports to the site project office for storage in site project files.

- Original waste container COC forms
- All raw data, including instrument readouts, calculation records, and RA QC results
- All instrument calibration reports, as applicable

Appendix B

Calculation of ^{239}PU Equivalent Activity[B11]

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Calculation of ^{239}PU Equivalent Activity[B11]

Pu-239 equivalent activity is determined using radionuclide-specific weighting factors. To obtain this correlation, the 50-year committed effective dose equivalent (CEDE) or dose conversion factor (DCF) for a unit intake of each radionuclide will be used. These DCFs have been determined by the methodology described in International Commission on Radiological Protection (ICRP) Publications 26[B2] and 30[B3] are consistent with current DOE guidance[B4]. The Pu-239 equivalent activity (AM) can be characterized by:

$$AM=\sum_{i=1}^{k}\frac{A_i}{WF_i}$$

Where K is the number of transuranic (TRU) radionuclides, A_i is the total radioactivity of radionuclide 1, and Wf_i is the PE-Ci weighting factor for radionuclide 1.

Wf_i is further defined as the ratio:

$$WF_i=\frac{E_o}{E_i}$$

Where E_o (rem/μCi) is the 50-year CEDE due to the inhalation of Pu-239 particles with a 1.0 μm Activity median Aerodynamic Diameter (AMAD) and a weekly (W) pulmonary clearance class, and E_i (rem/μCi) is the 50-year CEDE due to the inhalation of radionuclide 1 particulates with a 1.0 μm AMAD and the pulmonary clearance class resulting in the highest 50-year CEDE.

The value of E_o and E_i may be obtained from DOE/EH-0071[B5]. Weighting factors calculated in this manner are presented below for selected radionuclides of interest.

To determine if a waste package with several radionuclides does not exceed 80 Ci Pu-239 equivalent, AM from the previous page must be less than or equal to 80.

No estimate of non-TRU radionuclides, except those within the scope of the above description, should be included.

Radionuclide	Pulmonary Clearance Class*	Weighting Factor	80 Ci Pu-239 Equivalent (CiE)
U-233	Y	3.9	312
Np-237	W	1.0	80
Pu-236	W	3.2	256
Pu-238	W	1.1	88
Pu-239	W	1.0	80
Pu-240	W	1.0	80
Pu-241	W	52.0	4160
Pu-242	W	1.1	88
Am-241	W	1.0	80
Am-243	W	1.0	80
Cm-242	W	20.0	2400
Cm-244	W	1.9	152
Cf-252	Y	3.9	312

* (W) Weekly (Y) Yearly

REFERENCES

B-1. Reprinted from Appendix A of Reference 7.

B-2. "Recommendations of the Internal Commission on Radiological Protection," ICRP Publication 26, January 1977.

B-3. "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, July 1978.

B-4. DOE Memorandum, April 25, 1985, R>W. Earl to C.N. Mitchell, "Radiological Siting Requirements DOE Order 6430.1, General Design Criteria."

B-5. "DOE/EH-0071, "Internal Dose Conversion Factors for Calculation of Dose to the Public," July 1988.

Appendix C

Descriptions of Waste Forms

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Descriptions of Waste Forms

This appendix presents more detailed descriptions (including chemical and physical attributes and packaging configurations) of the higher population waste drums which are accessible and, therefore, subject to shipment to WIPP to satisfy the settlement agreement. All of these waste forms came from Rocky Flats and, therefore, contain varying amounts of WG plutonium, americium, and uranium. These waste forms (with the exception of content codes 337, 374/960, 376, 432, and 339/463, which are in the optional waste form set) comprise the basic waste form drum set which a participant may be requested to assay. In 1972 the practice of inserting a 90-mil polyethylene liner into the drum prior to the addition waste was initiated. Information in this section was obtained from references 3 and 4.

C-1. UNCEMENTED INORGANIC SLUDGE (CONTENT CODES 001/002/800)

This waste is a wet sludge precipitate generated by processing liquid wastes such as ion exchange column effluents, distillates, caustic soda solution, etc. produced by Rocky Flats plutonium recovery operations. In some cases, the various sources of liquid waste were made basic by addition of sodium hydroxide, combined, and the plutonium and americium scavenged from the liquid by a carrier-hydroxide precipitate process. Coagulating agents [$Fe(SO_4)_3$, $MgSO_4$, $CaCl_2$, and flocculating agents] were added to form a precipitate which was subsequently filtered. The treatment process produced a precipitate of the hydrated oxides of iron, magnesium, aluminum, silicon, etc., which also carried the hydrated oxides of plutonium and americium. The precipitate or slurry was filtered to produce a sludge containing 50 to 70 weight percent of water with a consistency similar to paste or mortar.

The waste was packaged in 55-gal drums to which Portland cement was added to absorb any free liquids. First, Portland cement (3-5 lbs) was placed in the bottom of the drum. The drum was then lined with two 65-gal drum bags each containing additional amounts of Portland cement (3-5 lbs in the bottom of the outer bag and about 30 lbs in the bottom of the inner bag. In 1972 the practice of inserting a 90-mil polyethylene liner into the drum prior to the addition of Portland cement and the drum bags was initiated. The sludge material was then dispensed into the inner drum bag, and the bag was taped shut. Beginning in the Spring of 1982, 3-5 lbs of Portland cement was added to the top of the inner bag before it was taped shut. Then another layer (3-5 lbs) of Portland cement was placed over the top of the sealed inner bag, and the outer drum bag was then taped shut. In 1972, the practice of placing 1-2 quarts of Oil-Dri on top of the sealed outer drum bag was initiated; the polyethylene liner lid was sealed, and the drum lid and gasket were installed and secured. In the Spring of 1982, the practice of using a 3-12 lb layer of vermiculite in place of the Oil-Dry was initiated. In order to reduce radiation exposure from some drums, lead (in the form of lead sheeting placed immediately inside the drum wall, or, beginning in 1972, lead tape wrapping the outside of the 90-mil polyethylene liner) was used.

The average drum weight for content code 001 is 490 lbs, and the range is 118 to 933 lbs. Contact radiation doses average 22.9 mR/hr and range from 0 to 195 mR/hr. Plutonium and americium inventories average 4.3 g Pu and 1.8 g Am and range from 0 to 157 and 52.9 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present. The average drum weight for content code 002 is 528 lbs, and the range is 210 to 952 lbs. Contact radiation doses average 0.7 mR/hr and range from 0 to 8.9 mR/hr. Plutonium and americium inventories average 0.2 g Pu and 0 g Am and range from 0 to 8.9 g and 7.1 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present. The average drum net weight for content code 800 (which replaced content code 001 in 1986) is 369 lbs, and the range is 157 to 614 lbs. Contact radiation doses range for < 10 mR/ hr to 200 mR/hr. Plutonium, americium, and uranium inventories average 4 g Pu, 0.9 g Am, and 0.7 g ^{235}U and range from 0 to 32 g for Pu, 0 to 3.9 g for Am, and 0.3 to 1.7 g for U.

C-2. UNCEMENTED ORGANIC SLUDGE (CONTENT CODE 003)

This waste comes from the processing of organic wastes generated at the various plutonium and nonplutonium operational areas at Rocky Flats. The organic waste forms generated as a byproduct of plutonium fabrication operations are primarily comprised of trichloroethane, carbon tetrachloride, and machining and hydraulic oils (classified as Texas Regal oil). Organic wastes from nonplutonium areas includes similar components plus carbon tetrachloride, trichloroethylene, tetrachloroethylene, trace concentrations of organophosphates and nitrobenzene, and hydraulic and gearbox oils. Freon is also a significant component of the organic liquid in this sludge, estimated at 6.0% by mass. Unknown quantities of polychlorinated biphenyls were also processed in addition to the typical organic waste through 1979.

Organic waste is processed for packaging by blending approximately 30 gallons of organics with 100 pounds of calcium silicate in a continuous mixer. Oil-Dry compound was typically included in the blending process at a mass of approximately 15 pounds per drum. The resultant blending process product is a sludge material with a semi-solid paste or grease consistency. A four pound mass of Oil-Dry is placed in the bottom of the 55-gal drum to absorb potential oil migration from the two plastic 65-gal drum bags which are subsequently used to line the drum. Each of these bags also has four pounds of Oil-Dry placed in them prior to dispension of the solidified sludge material. Oil-Dry is also added to the top of the outer bag after the sludge has been dispensed.

The average drum weight for this waste form is 509 lbs, and the range is 89 to 910 lbs. Contact radiation doses average 0.4 mR/hr and range from 0 to 35 mR/hr. Plutonium and americium inventories average 0.3 g Pu and 0.0 g Am and range from 0.0 to 16.0 and 1.2 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

C-3. SOLIDIFIED ORGANICS (CONTENT CODE 801)

This waste consists of cemented waste oils and solvents that were generated as a result of machining and tool degreasing.

The waste was pumped into the inner of two PVC O-ring bags contained in a 55-gal. drum. Envirostone emulsifier, gypsum cement, and accelerator were mixed with the waste metered into the bag, and then water was added to the mixture. A mixer was lowered into the drum after all of the materials were added. The amount of materials added to the mixture was computer controlled. When the mixture began to stiffen, the mixer was removed, and the drum was placed aside for setup to occur. Following setup, the liner and drum were closed.

The average drum net weight for this waste form is 453 lbs, and the range is 122 to 598 lbs. Contact radiation doses for most drums is <10 mR/hr, but a few are in the 10-200 mR/hr range. Plutonium inventories average 3.26 g Pu and range from 0 to 70 g. About 1 g Am is reported in each drum. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

C-4. SPECIAL SETUPS (CONTENT CODE 004)

This waste consists of liquids absorbed on a cement mixture, the liquids containing plutonium complexing chemicals such as alcohols, organic acids, Versenes (trademark for a series of chelating agents based on EDTA).

The waste was packaged in 55-gal drums to which a mixture of Portland cement and pipe insulation cement (e.g., magnesia cement) was added to absorb any free liquids. First, thee drum was prepared by placing Portland cement (3–5 lbs) in the bottom of the drum. The drum was then lined with two 65-gal drum bags, the first one containing an additional amount of Portland cement (3–5 lbs in the bottom of the outer bag) and about 190 lbs of Portland cement and 50 lbs of pipe insulation cement in the inner bag. The prepared drum is placed on a drum roller and rolled to ensure mixing of the cements. In 1972 the practice of inserting a 90-mil polyethylene liner into the drum prior to the addition of Portland cement and the drum bags was initiated. Approximately 100 liters (26.4 gallons) of liquid waste material was made basic and then poured on the cement mixture in the inner drum bag and allowed to solidify. Approximately 10–15 lbs of Portland cement was then added on top of the cemented liquid waste before the bag was sealed. Then another layer (3–5 lbs) of Portland cement was placed over the top of the sealed inner bag, and the outer drum bag was then sealed. In 1972, the practice of placing 1–2 quarts of Oil-Dri on top of the sealed outer drum bag was initiated; the polyethylene liner lid was sealed, and the drum lid and gasket were installed and secured. In the Spring of 1982, the practice of using a 3–12 lb layer of vermiculite in place of the Oil-Dry was initiated.

Some drums may be filled with empty polyethylene bottles used to transport liquid waste. A small amount of Portland cement is added to each bottle before placement in the drum. Periodically, a drum will contain polyethylene bottles of cemented liquid wastes. The bottles had been filled with the cement mixture and sent to various small waste generators for addition of the liquid waste. The bottles were then collected and placed in a prepared 55-gal drum.

The average drum weight for this waste form is 585 lbs, and the range is 102 to 1076 lbs. Contact radiation doses average 1.2 mR/hr and range from 0 to 180 mR/hr. Plutonium and americium inventories average 1.0 g Pu and 0.0 g Am and range from 0.0 to 22.7 and 2.4 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

C-5. SOLIDIFIED LAB WASTE (CONTENT CODE 802)

This waste comes from analytical labs, research and development labs, and maintenance shops. It consists of liquid lab waste containing hydrochloric acid.

The waste is packaged in 55-gal. drums and then solidified. The waste is first made slightly basic by adding sodium hydroxide. It is then transferred into a prepared 55-gal. drum (prepared with an O-ring bag and a polyethylene bag inside the rigid liner). A maximum of 80 liters of waste solution could be added to a prepared drum. Portland cement (42.3% by weight) and Ramcote (22.8% by weight) absorbent cement is added to the waste (35% by weight) to immobilize it.

The average drum net weight for this waste form is 559 lbs, and the range is 239 to 649 lbs. Contact radiation doses are <10 mR/hr. Plutonium and uranium inventories average 5 g Pu and 37 g ^{235}U and range form 0 to 21 g and 1 to 73 g, respectively. No average or range for americium has been reported. This does not, however, preclude the possibility that americium may be present.

C-6. CEMENTED INORGANIC SLUDGE (CONTENT CODE 007)

This waste is a variant of the content code 001 packaging configuration. As noted above, the content code 001 package configuration consists of layers of Portland cement in the drum bottom, and bag liners followed by the sludge material itself, then topped with additional Portland cement and Oil-Dry. The content code 007 configuration differs from that of content code 001 in that the sludge material itself has approximately 50 pounds of Portland cement uniformly mixed and distributed.

The average net drum weight for this waste form is 410 lbs, and the range is 117 to 650 lbs. Contact radiation doses are low, <10 mR/hr, except for a very few drums. Plutonium and americium inventories average 0 g Pu and 0 g Am and range from 0 to 29 and 0.06 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

C-7. SOLIDIFIED DCP SLUDGE (CONTENT CODE 803)

This waste consists of clarifier slurry from the Radioactive Decontamination Process, Acid Neutralization Process wastes, and acid descaling solution from the Evaporation Process. The sludge is first dried and then cemented in the Direct-Cementation Process (DCP) prior to packaging in 55-gal. drums.

In the DCP, dried sludge, Portland cement, and water are metered by a computer and mixed using a paddle mixer to produce a cemented waste product. This product is transferred into a prepared (with an O-ring bag and a polyethylene bag placed inside the rigid liner) 55-gal. drum.

The average drum net weight for this waste form is 530 lbs, and the range is 157 to 672 lbs. Contact radiation doses are all < 10 mR/hr. Plutonium and americium inventories average 0.3 g Pu and 0.1 g Am and range from 0 to 4.3 and 0.3 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

C-8. SOLIDIFIED BYPASS SOLIDS (CONTENT CODE 807)

This waste consists of immobilized materials from the Decontamination-Precipitation and Neutralization Process in the Liquid Waste Treatment Facility, clarifier slurry from the Radioactive Decontamination Process, Acid Neutralization Process wastes, and acid descaling solution from the Evaporation Process.

The slurry is drawn through a filter drum where a sludge of precipitated solids precipitated solids is skimmed from the surface of the filter media. It is then transferred directly into a prepared (with a 14-mil PVC O-ring bag and a 5-mil polyethylene bag inside the rigid liner). Portland cement and diatomite are mixed with the waste to solidify it. Content code 807 was created in 1987 to replace content code 007.

The average drum net weight for this waste form is 353 lbs, and the range is 10 to 509 lbs. Contact radiation doses are < 10 mR/hr. Plutonium and americium inventories average 2 g Pu and 0.01 g Am and range from 0 to 161 and 0.155 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

C-9. CEMENTED SLUDGE (CONTENT CODE 292)

This waste consists primarily of sludge generated from filter plenums, pumps, and incinerator off-gas systems. It may also contain a limited number of surgeons' gloves. Portland cement is added to the sludge for absorption of free liquid.

Prior to approximately 1977, sludge waste was placed in a PVC bag and sealed with tape. The bag was then double-contained in plastic and placed in a 1-gallon metal paint can containing Portland cement. More Portland cement was added, and the paint can lid installed. Approximately 25 cans were placed into a drum, depending upon the plutonium content.

Since approximately 1977, sludge has been collected in 1-gallon polyethylene bottles. Portland cement is added in layers as the bottle fills with sludge. The sludge is capped with cement, the bottle lid installed, and the bottle then double-contained in plastic. Each bottle contains approximately 1 pound of Portland cement. An estimated 20 bottles were placed in a drum, depending upon plutonium content.

Since approximately 1972, the drums which accepted the paint cans or polyethylene bottles were prepared with one or two polyethylene drum bags inside the 90-mil rigid polyethylene liner. After the addition of the cans or bottles, approximately 1 to 2 quarts of Oil-Dri were placed on top of the outer, sealed polyethylene drum bag. Since February 1982, 3 to 12 pounds of vermiculite have been used to fill the remaining space between the outer, sealed polyethylene drum bag and the top of the 90-mil rigid liner.

The average drum weight for this waste form is 265 lbs, and the range is 111 to 522 lbs. Contact radiation doses average 1.1 mR/hr and range from 0.0 to 7.0 mR/hr. Plutonium and americium inventories average 23.6 g Pu and 0.0 g Am and range from 0.0 to 162 and 2.0 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

C-10. COMBUSTIBLES - DRY (CONTENT CODE 330)

This waste consists of dry combustibles such as paper, rags, plastics, surgeons' gloves, cloth overalls and booties, cardboard, wood, wood filter frames, polyethylene bottles, and laundry lint. Damp or moist combustible wastes may also be present. Prior to being placed in a prepared 55-gal drum (i.e., a drum with two drum bags), most of the waste was double bagged in polyethylene or PVC. Some of the waste was first placed in 1-gal polyethylene bottles before being bagged. For the pre-1975 drums, Oil-Dry may have been added to the waste drums.

The average drum weight for this waste form is 183 lbs, and the range is 82 to 561 lbs. Contact radiation doses average 0.5 mR/hr and range from 0 to 55 mR/hr. Plutonium and americium inventories average 0.5 g Pu and 0.0 g Am and range from 0.0 to 45 and 27 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

C-11. COMBUSTIBLES - MOIST (CONTENT CODE 336)

This waste consists of damp or wet line- and some nonline-generated combustible wastes such as paper, rags, and Kimwipes. Other combustibles which might be present include plastics, surgeons' gloves, canvas, wood, cardboard, polyethylene bottles, and rubber. The moisture content ranges from moist to wet, and dry combustibles may be present. Prior to the addition of waste, the drums were prepared by lining them with one or two drum bags. Prior to 1975, the waste may have been placed directly in the drum or single- or double-contained in polyethylene or PVC. Oil-Dry, ranging from none to 55 pounds may have been added to the drum to absorb moisture. If added, the Oil-Dry was usually placed in the bottom of the drum, and more was added as the drum was filled with waste. Beginning in 1975, the waste was usually double-contained in PVC and polyethylene bags. Until 1977, combustible wastes were compacted in prepared waste drums.

The average drum weight for this waste form is 197 lbs, and the range is 91 to 596 lbs. Contact radiation doses average 0.5 mR/hr and range from 0 to 52 mR/hr. Plutonium and americium inventories average 0.3 g Pu and 0.0 g Am and range from 0.0 to 55 and 45 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

C-12. METALS - UNLEACHED (CONTENT CODE 480)

This waste consists of non-stainless steel metals (such as iron, copper, aluminum) and stainless steel which may be in the form of gloveboxes, glovebox windows, furnaces, lathes, drill presses, ducting, piping, angle iron, tanks, downdraft tables, part-carriers, respirator filters, ultrasonic cleaners, control panels, electronic instrumentation, vacuum sweepers, pumps, motors, railing, stairs, metal racks and trays, hotplates, empty metal produce and paint cans, carts, power tools, hand tools, chairs, desks, tables, typewriters, filing cabinets, crushed 55-gal drums, etc. The waste may also include limited amounts of combustible wastes. The waste may or may not be double-contained in plastic before being placed in prepared (i.e., lined with two drum bags) drums.

The average drum weight for this waste form is 253 lbs, and the range is 90 to 795 lbs. Contact radiation doses average 0.5 mR/hr and range from 0 to 66 mR/hr. Plutonium and americium inventories average 3.6 g Pu and 0.0 g Am and range from 0.0 to 129 and 20 g, respectively. Uranium-235 inventories of 12 g have been reported.

C-13. METALS - LEACHED (CONTENT CODE 481)

This waste consists of non-stainless steel metals (such as iron, copper, and aluminum) and primarily stainless steel in the form of small hand tools, valves, trays, clamps, pipe, etc. The metal waste has been processed by hot-water washing for plutonium recovery. The waste was placed directly into a drum (or, beginning in 1972, the drum liner) which had been prepared by being lined with two polyethylene drum bags. Each bag was then sealed. Since approximately 1972, the drums were inspected for free liquids (corrected if found to contain free liquids), and then 2-3 pounds of Oil-Dry was added to the top of the outer, sealed polyethylene bag. Beginning in 1982, approximately 3-12 pounds of vermiculite was used to fill the gap between the outer, sealed polyethylene drum bag and the top of the 90-mil rigid liner.

The average drum weight for this waste form is 295 lbs, and the range is 103 to 658 lbs. Contact radiation doses average 1.2 mR/hr and range from 0.1 to 70 mR/hr. Plutonium and americium inventories average 21.6 g Pu and 0.0 g Am and range from 0.0 to 116 and 3.0 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

C-14. METALS - HEAVY NON-STAINLESS STEEL METALS (CONTENT CODE 320)

This waste consists primarily of tantalum components such as crucibles, funnels, funnel inserts, and pour-rods. Other metals may include tungsten, platinum, and lead.

The tantalum components, after going through a process to remove adhering plutonium and americium, are placed in double PVC bags and sealed. The bagged tantalum is then placed in Fibre-Paks, with two Fibre-Paks being placed in a prepared 55-gal. drum. The drums are prepared with two polyethylene bags. Since 1972, lead used for lead-lined drums is located between the drum and the rigid liner.

The average drum weight for this waste form is 222 lbs, and the range is 101 lbs to 576 lbs. Contact radiation doses average 2.9 mR/hr and range from 0.1 to 70 mR/hr. Plutonium and americium inventories average 30 g Pu and 0 g Am and range from 0 to 183 g and 6.3 g, respectively.No average or range for uranium has been reported. Uranium-235 inventories up to 3 g have been reported.

C-15. GLASS (CONTENT CODE 440)

This waste consists of glass in the form of sample vials and bottles, lead-taped sample vials, ion exchange columns, dissolver pots, laboratory glassware such as Pyrex flasks and beakers, glovebox windows (glass, Plexiglass, leaded glass), and crushed ground glass. The waste may also contain limited amounts of other noncombustibles (such as metal) and combustible wastes.

Glass is packaged in several different ways, depending on the area in which the waste is generated. The glass may be whole, broken into pieces, or in some instances, crushed or ground. Whole or broken glass may be packaged in the following ways: (1) in 1-gal polyethylene bottles; (2) in 13-in.-high x 15-1/2-in.-diameter Fibre-Paks (the glass being either loose or contained in plastic bags inside the Fibre-Pak); (3) double-contained in plastic bags, with the outside of the outer bag taped for protection against sharp edges; and (4) glassware such as sample vials which may be taped together. All waste is usually double-contained in plastic (PVC/polyethylene). Nonline-generated glassware, light bulbs, and fluorescent tubes are usually crushed or ground and placed directly into a prepared (i.e., lined with two drum bags) 55-gal drum.

The average drum weight for this waste form is 232 lbs, and the range is 88 to 922 lbs. Contact radiation doses average 1.1 mR/hr and range from 0.1 to 86 mR/hr. Plutonium and americium inventories average 5.2 g Pu and 0.0 g Am and range from 0.0 to 182 and 11.0 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

C-16. GLASS - UNLEACHED RASCHIG RINGS (CONTENT CODES 441)

This waste consists of borated-glass rings used to minimize neutron multiplication in liquid storage tanks. The rings are approximately 1.75 in. high x 1.50 in. diameter with a wall thickness of approximately 0.25 in. They contain 11.8 to 13.8 wt% B_2O_3, with an isotopic content of $^{10}B/^{11}B$ of not less than 0.24. Some Raschig rings have been broken up into fragments of approximately 1/4 in. diameter and the fragments placed in 4-liter polyethylene bottles. Drums containing Raschig rings from oil and carbon tetrachloride tanks may contain Oil-Dri. The Raschig rings were placed in prepared drums (i.e., drums lined with one or two drum bags). Raschig rings removed from a liquid storage tank are triple-contained in plastic (polyethylene and/or PVC) and placed in a 13-in.-high x 15-1/2-in.-diameter Fibre-Pak. Two Fibre-Paks are placed in a prepared 55-gal drum. Prior to 1972, the drums might have had a cardboard liner between the drum bags. Since approximately 1972, drums have been inspected for free liquids (the rejected ones being returned for correction), and then 1-2 quarts of Oil-Dri were placed on top of the outer, sealed polyethylene drum bag. Beginning in February 1982, 3-12 pounds of vermiculite were used to fill the space between the outer, sealed polyethylene drum bag and the top of the 90-mil rigid liner.

The average drum weight for this waste form is 194 lbs, and the range is 100 to 563 lbs. Contact radiation doses average 0.9 mR/hr and range from 0.1 to 20 mR/hr. Plutonium and americium inventories average 7.9 g Pu and 0.0 g Am and range from 0.0 to 132 and 4.0 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

C-17. GLASS - LEACHED RASCHIG RINGS (CONTENT CODE 442)

This waste is the same as content code 440 except that the Raschig rings have been leached. Raschig rings contaminated with above-discard amounts of plutonium are processed by leaching with nitric acid or water. After leaching, the rings are rinsed with water and allowed to dry before being packaged. They are then double-contained in plastic bags and placed in a Fibre-Pak. Two Fibre-Paks are placed inside a prepared 55-gal drum.

The average drum weight for this waste form is 182 lbs, and the range is 105 to 484 lbs. Contact radiation doses average 0.3 mR/hr and range from 0.1 to 6.0 mR/hr. Plutonium and americium inventories average 2.1 g Pu and 0.0 g Am and range from 0.0 to 49 and 2.0 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

C-18. GRAPHITE MOLDS (CONTENT CODE 300)

This waste consists of graphite molds used in casting plutonium metal. Flat and shaped graphite molds from the foundry operations are brushed with a wire brush to remove adhering plutonium, broken into large pieces, and placed directly into a prepared (i.e., lined with one or two drum bags) 55-gal waste drum. Graphite waste from the plutonium recovery operations are processed by scarfing (i.e., scraping) to remove adhering plutonium and then placed in 13-in.-high x 15-1/2-in.-diameter Fibre-Paks. The Fibre-Paks are single- or double-contained in plastic (PVC/polyethylene) bags, the bags are sealed, and then two Fibre-Paks are placed in a prepared drum. Prior to 1972, the inner drum bag in the drums from the foundry operations was lined with a cardboard liner. Since 1972, only one drum bag was used to line the 90-mil rigid polyethylene liner, and this drum bag was lined with a cardboard liner (bottom and side only). Prior to 1972, one or two drum bags were used for drums from the plutonium recovery operations, and cardboard liners may have been used. Since 1972, one or two drum bags have been used to line the 90-mil rigid polyethylene liner, and no cardboard was used. Since approximately 1972, drums have been inspected for free liquids (rejected drums being returned for correction), and 1-2 quarts of Oil-Dri is placed on top of the outer, sealed polyethylene drum bag. Since February 1982, 3-12 pounds of vermiculite has been used to fill the space between the outer sealed polyethylene drum bag and the top of the 90-mil rigid liner.

The average drum weight for this waste form is 254 lbs, and the range is 110 to 473 lbs. Contact radiation doses average 0.8 mR/hr and range from 0 to 100 mR/hr. Plutonium inventories average 9.9 g Pu and range from 0.0 to 61 g. No average or range for americium has been reported. This does not, however, preclude the possibility that americium may be present. Uranium-235 inventories average 3 g and range from 1 to 5 g.

C-19. GRAPHITE CORES (CONTENT CODE 301)

This waste is very similar to content code 300, since a graphite core is part of a shaped mold used in casting plutonium metal. The waste in this content code will consist of both graphite molds and cores, and it is treated and packaged the same as content code 300.

The average drum weight for this waste form is 260 lbs, and the range is 164 to 471 lbs. Contact radiation doses average 0.7 mR/hr and range from 0.5 to 4.0 mR/hr. Plutonium inventories average 12.6 g Pu and range from 0.0 to 45 g. No average or range for americium has been reported. This does not, however, preclude the possibility that americium may be present. No uranium has been reported to be present, but small amounts (up to several grams) of ^{235}U can be expected, based on the reported inventories for content codes 300 and 303.

C-20. SCARFED GRAPHITE CHUNKS (CONTENT CODE 303)

This waste is content code 300 graphite which has been scarfed (i.e., cleaned using a rotary-type sanding tool) to remove recoverable plutonium. Use of this content code began in the early 1980s. The waste in this content code was placed in a prepared 55-gal drum which contained a 50-mil fiberboard liner inside the 90-mil rigid polyethylene liner.

The average net drum weight for this waste form is 175 pounds, and the range is 29 to 214 pounds. Contact radiation doses are low, <10 mR/hr for all but a few drums. Plutonium inventories average 18.4 g Pu and range from 0.0 to 95 g. No average or range for americium has been reported. This does not, however, preclude the possibility that americium may be present. Uranium-235 inventories average 4.29 g and range from 1 to 9 g.

C-21. FIREBRICK (CONTENT CODE 371)

This waste consists of whole and broken pieces of construction bricks, cinderblocks, and firebrick. It may also contain limited amounts of other noncombustible and combustible wastes. Firebrick waste generated since 1973 from the Plutonium Recovery incinerator is a high-alumina, high strength, Class F brick manufactured by Plibrico (brick trade name: Plicast 40). Typical brick composition is: Al_2O_3 = 95.67%, SiO_2 = 0.03%, Fe_2O_3 = 0.10%, TiO_2 = 0.01%, CaO = 3.60%, MgO = 0.08%, alkalies = 0.28%. If the firebrick has been contaminated with above-discard amounts of plutonium, it was scarfed (i.e., scraped) to remove surface contamination.

During the period 1971-1973, brick waste was packaged by three different methods: (1) double-contained in plastic and then placed into Fibre-Paks, two Fibre-Paks to a prepared 55-gallon drum; (2) double-contained in plastic and placed directly into a prepared 55-gallon drum; (3) no packaging, i.e., placed directly into a prepared 55-gallon drum.

Since 1974, firebrick waste is double-bagged into PVC and polyethylene bags, each bag being sealed with tape. The bag is then placed into a 13-in.-high x 15-1/2-in.-diameter Fibre-Pak. When the Fibre-Pak is filled, its lid is replaced. Two Fibre-Paks will fit in a drum.

During the period 1970-1972, waste drums were prepared by lining them with one or two polyethylene drum bags. Cardboard liners may have been used to line the inner drum bag. Since approximately 1972, a 90-mil rigid polyethylene liner was used, with the 90-mil liner being lined with one or two polyethylene drum bags. Approximately 1 to 2 quarts of Oil-Dri were placed on top of the outer, sealed polyethylene drum bag. Since February 1982, 3 to 12 pounds of vermiculite have been used to fill the space between the outer, sealed polyethylene drum bag and the top of the 90-mil rigid liner.

The average drum weight for this waste form is 361 lbs, and the range is 102 to 770 lbs. Contact radiation doses average 0.5 mR/hr and range from 0.1 to 5.2 mR/hr. Plutonium inventories average 3.7 g Pu and range from 0.0 to 89 and 2.0 g. No averages or ranges for americium or uranium have been reported. This does not, however, preclude the possibility that americium and uranium may be present.

C-22. PLASTIC AND NONLEADED RUBBER (CONTENT CODE 337)

This waste consists of various types of plastics such as polyethylene, PVC, Teflon, and nonleaded rubber items. The waste may be in the form of bags, sample vials, bottles, sheeting, and surgeons' gloves. Other types of combustible waste, such as respirator face masks and paper, may also be included. The waste might include limited amounts of noncombustible items. The majority of this waste should be dry.

Prior to 1975, the waste was packaged by either placing items directly into a prepared 55-gallon drum or by double-containing items in plastic (PVC/polyethylene) before placing them in a prepared drum. Absorbent, such as Oil-Dri or Portland cement, was generally added to the waste if moisture was present. Absorbent, if added, was usually placed in the bottom of the drum; more was added as the container was filled with waste. In some instances, absorbent material was added to the waste itself (such as an empty polyethylene bottle). The quantity of absorbent added to the waste or placed in a waste container depended upon the individual packaging the waste.

Since 1974, content code 337 waste has been packaged as follows:

Waste from Aqueous Waste Treatment (Building 774) consists of plastic bags and polyethylene bottles. Bottles are emptied; a small amount of Portland cement is added to each bottle to absorb any residual liquid; then the bottles are recapped. The bottle and containment bags are placed directly into a prepared 55-gallon drum. Portland cement is added to each drum prior to the addition of waste bottles. An estimated 18-24 bottles will fit into a drum. The total quantity of Portland cement in a drum is estimated at 15-18 pounds.

Waste from Plutonium Recovery Operations (Building 771) consists of surgeons' gloves, plastic bags, and polyethylene bottles. Waste is double-contained in plastic (PVC/polyethylene) bags. The bags are sealed with tape and then placed in a prepared 55-gallon drum.

Waste from Chemical Operations Support Laboratory (Building 771) consists of surgeons' gloves and limited amounts of unleaded neoprene and Hypalon glovebox gloves and paper wipes. The waste is double-contained in polyethylene bags. Each bag is closed by sealing with tape and then placed in a prepared 55-gallon drum.

R&D Chemical Technology Division (Building 771) waste consists of surgeons' gloves, plastic bags, polyethylene bottles, and possible limited amounts of paper and Kimwipes. Waste is double-bagged in PVC and polyethylene bags. Each bag is sealed with tape and then placed in a prepared 55-gallon drum.

Prior to approximately 1972, drums were prepared by lining them with one or two polyethylene drum bags. Cardboard liners might have been used to line the inner drum bag. After being filled with waste, each drum bag was sealed with tape. In approximately 1972, use of the 90-mil rigid polyethylene liner began. The rigid liner is lined with one or two polyethylene drum bags. After being filled with waste packages, the bags are sealed with tape.

Since approximately 1972, 1-2 quarts of Oil-Dri were placed on top of the outer, sealed polyethylene drum bag. Beginning in February 1982, 3-12 pounds of vermiculite has been used to fill the space between the outer, sealed polyethylene drum bag and the top of the 90-mil rigid liner.

The average drum weight for this waste form is 170 lbs, and the range is 81 to 474 lbs. Contact radiation doses average 0.6 mR/hr and range from 0.0 to 24 mR/hr. Plutonium and americium inventories average 0.8 g Pu and 0.0 g Am and range from 0.0 to 49 and 10 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

C-23. BLACKTOP, CONCRETE, DIRT, AND SAND (CONTENT CODES 374/960)

This waste form consists of blacktop, concrete, reinforced concrete, cinderblocks, bricks,dirt, and sand. Waste may also include some combustibles such as surgeons' gloves and Kimwipes. Content code 960 was replaced by content code 374 in 1973.

Waste packaging and handling varied, depending on the waste-generating area. Some waste was placed directly in prepared 55-gallon drums, while other waste was first single- or double-contained in polyethylene and/or PVC plastic bags. Still other waste was packaged in Fibre-Paks before being loaded into the drums.

Prior to approximately 1972, drums were prepared by lining them with one or two polyethylene drum bags. Cardboard liners might have been used to line the inner drum bag. After being filled with waste, each drum bag was sealed with tape. In approximately 1972, use of the 90-mil rigid polyethylene liner began. The rigid liner is lined with one or two polyethylene drum bags. After being filled with waste packages, the bags are sealed with tape.

Since approximately 1972, 1-2 quarts of Oil-Dri were placed on top of the outer, sealed polyethylene drum bag. Beginning in February 1982, 3-12 pounds of vermiculite has been used to fill the space between the outer, sealed polyethylene drum bag and the top of the 90-mil rigid liner.

The average drum weight for content code 374 waste is 390 lbs, and the range is 125 to 756 lbs. Contact radiation doses average 0.4 mR/hr and range from 0.1 to 7.2 mR/hr. Plutonium and americium inventories average 0.5 g Pu and 0.0 g Am and range from 0.0 to 44 and 1.0 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

The average drum weight for content code 960 waste is 439 lbs, and the range is 131 to 796 lbs. Contact radiation doses average 0.5 mR/hr and range from 0.0 to 0.8 mR/hr. Plutonium inventories average 0.4 g Pu and range from 0.0 to 137. No averages or ranges for americium or uranium have been reported. This does not, however, preclude the possibility that americium and uranium may be present.

C-24. CEMENTED FILTER MEDIA (CONTENT CODE 376)

This waste consists primarily of filter media (pre-1979) and filter media and whole filters (since 1979). The waste also contains limited amounts of insulation waste such as asbestos gloves and fireblankets. Portland cement has been added to all waste packages in order to neutralize any residual nitric acid that may be present.

Prior to 1979, the waste was usually emptied from its original packaging into a mortar box, mixed with Portland cement, repackaged in a 15-gal plastic bag, and this bag placed in a prepared (i.e., lined with one or two polyethylene drum bags) 55-gal drum. Since 1979, received waste is repackaged in a 15-gal polyethylene bag, a small quantity of Portland cement is added to the bag, the bag is shaken to spread the cement, and then the bag is placed in a prepared 55-gal drum. The total quantity of Portland cement in a waste drum may range up to 50 lbs.

The average drum weight for this waste form is 200 lbs, and the range is 100 to 409 lbs. Contact radiation doses average 3.4 mR/hr and range from 0.0 to 180 mR/hr. Plutonium and americium inventories average 22.2 g Pu and 0.1 g Am and range from 0.0 to 189 and 16 g, respectively. Uranium-235 inventories average 14 g and range from 1 to 23 g.

C-25. CEMENTED RESINS (CONTENT CODE 432)

This waste consists of anion and cation exchange resins used in the purification and recovery of plutonium and americium. These resins are first leached and rinsed. Then a slurry consisting of 1 liter Portland cement, 500 mL of water, and 1 liter of washed resin is poured into a 1-gal polyethylene bottle containing approximately 1/2 inch of dry Portland cement and allowed to cure. Another 1/2-inch layer of dry Portland cement is placed on top of the hardened resin/cement mixture before the bottle is capped. Each bottle is double bagged (PVC and polyethylene), with each bag being sealed with tape, and placed in a prepared 55-gal drum. Approximately 15 to 20 bottles fit in a drum. Drums containing bottles of resin waste from the americium recovery line are usually lead-lined. Lead sheeting (1/16 to 1/8 in. thick) is placed between the drum and the rigid drum liner. Since approximately 1972, 1 to 2 quarts of Oil-Dri were placed on top of the outer, sealed polyethylene bag. Since February 1982 3 to 12 pounds of vermiculite has been used to fill the space between the outer, sealed polyethylene drum bag and the top of the 90-mil rigid liner.

The average drum weight for this waste form is 273 lbs, and the range is 101 to 481 lbs. Contact radiation doses average 1.3 mR/hr and range from 0.1 to 30 mR/hr. Plutonium and americium inventories average 31.2 g Pu and 0.3 g Am and range from 0.0 to 195 and 4.5 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

C-26. LEADED RUBBER GLOVES AND APRONS (CONTENT CODES 339/463)

This waste consists of leaded glovebox gloves and aprons. It may also contain limited amounts of unleaded gloves, lead bricks, and lead sheeting. Content code 463 was replaced with content code 339 in 1973.

Glovebox gloves and aprons are double-contained in plastic (PVC/polyethylene) and placed in a prepared 55-gallon drum. If any moisture is present, Oil-Dri is added to the drum.

Prior to approximately 1972, drums were prepared by lining them with one or two polyethylene drum bags. After being filled with waste, each drum bag was sealed with tape. In approximately 1972, use of the 90-mil rigid polyethylene liner began. The rigid liner is lined with one or two polyethylene drum bags. After being filled with waste packages, the bags are sealed with tape.

Since approximately 1972, 1-2 quarts of Oil-Dri were placed on top of the outer, sealed polyethylene drum bag. Beginning in February 1982, 3-12 pounds of vermiculite has been used to fill the space between the outer, sealed polyethylene drum bag and the top of the 90-mil rigid liner.

The average drum weight for content code 463 is 368 lbs, and the range is 160 to 620 lbs. Contact radiation doses average 0.5 mR/hr and range from 0.5 to 1.5 mR/hr. Plutonium inventories average 14.3 g Pu and range from 0.0 to 57 g. No averages or ranges for americium or uranium have been reported. This does not, however, preclude the possibility that americium and uranium may be present. The average drum weight for content code 339 is 339 lbs, and the range is 130 lbs to 534 lbs. Contact radiation doses average 0.6 mR/hr and range from 0.1 to 24 mR/hr. Plutonium and americium inventories average 24.5 g Pu and 0.0 g Am and range from 0 to 98 g and 4.0 g, respectively. No average or range for uranium has been reported. This does not, however, preclude the possibility that uranium may be present.

Appendix D

Descriptions of Noninterfering Matrix Drums

Appendix D

Descriptions of Noninterfering Matrix Drums

This appendix contains brief descriptions (including drawings) of the INEL noninterfering matrix drums. Currently, the INEL has five noninterfering matrix drums, three containing no matrix material (i.e., empty except for structure for the mounting of source material) and two containing an ethafoam matrix. Each drum is configured so that source material can be placed at various locations in the drum.

D-1. ZERO MATRIX DRUMS

There are three different zero matrix drum configurations, the differences between the three being in the structures used for the mounting of source material. Type 1 has three vertical source insert tubes, each 34.875-in. long and 1.53-in. inside diameter. Type 2 has three vertical source insert tubes each 32.875-in. long and 2.15-in. inside diameter. Type 3 has five horizontal rods from which source material can be hung. The Type 1 drum design is the basic design for several of the INEL surrogate drums (e.g., content codes 480/481), while the Type 2 drum design is the basic design for the PDP drums. Type 3 is a unique design, developed to give flexibility in source material placement.

D-1.1 Type 1 Zero Matrix Drum

A three-dimensional view of the internal source matrix/support assembly is shown in Figure D-1 with elevation and plan views as shown in Figures D-2 and D-3, respectively. The internal support consists of 1/8-in. thick, 22-3/8-in. diameter top and bottom aluminum support plates connected by seven 3/8-in. diameter carbon steel support rods. Aluminum (type 6061) is used in the support structure to the extent practicable to minimize neutron interactions.

A significant component of the internal matrix/source support assembly are the three aluminum source insert tubes depicted in Figures D-1, D-2, and D-3. These source insert tubes run vertically through the drum and are welded to the top and bottom aluminum support plates. The source insert tubes are 34-7/8 inches high, with internal and external dimensions of 1.53-in. and 1.625-in., respectively. One source insert tube is fixed at the drum center, one at a radius of 5.5-in., and the third at a radius of 10.25-in., corresponding to the closest point of any matrix to the interior drum package wall.

To precisely locate a source at a given location within the source insert tube, source insert fixtures were fabricated from aluminum. Source insert fixtures (see Figures D-1 and D-4) were designed to accommodate the nuclear accident dosimeter (NAD)-type source utilized for calibration measurements at the SWEPP facility. Cutouts dimensioned to accommodate the NAD-type sources are located on 3-in. centers down the length of the source insert fixture. Aluminum clips are provided to secure the NAD-type sources within a given cutout to immobilize them and ensure precise positioning. A second, similar type of source insert fixture accommodates ZPPR-type fuel plates. Note in the Figure D-4 detail that the top cap of the source insert fixture is circumferentially marked

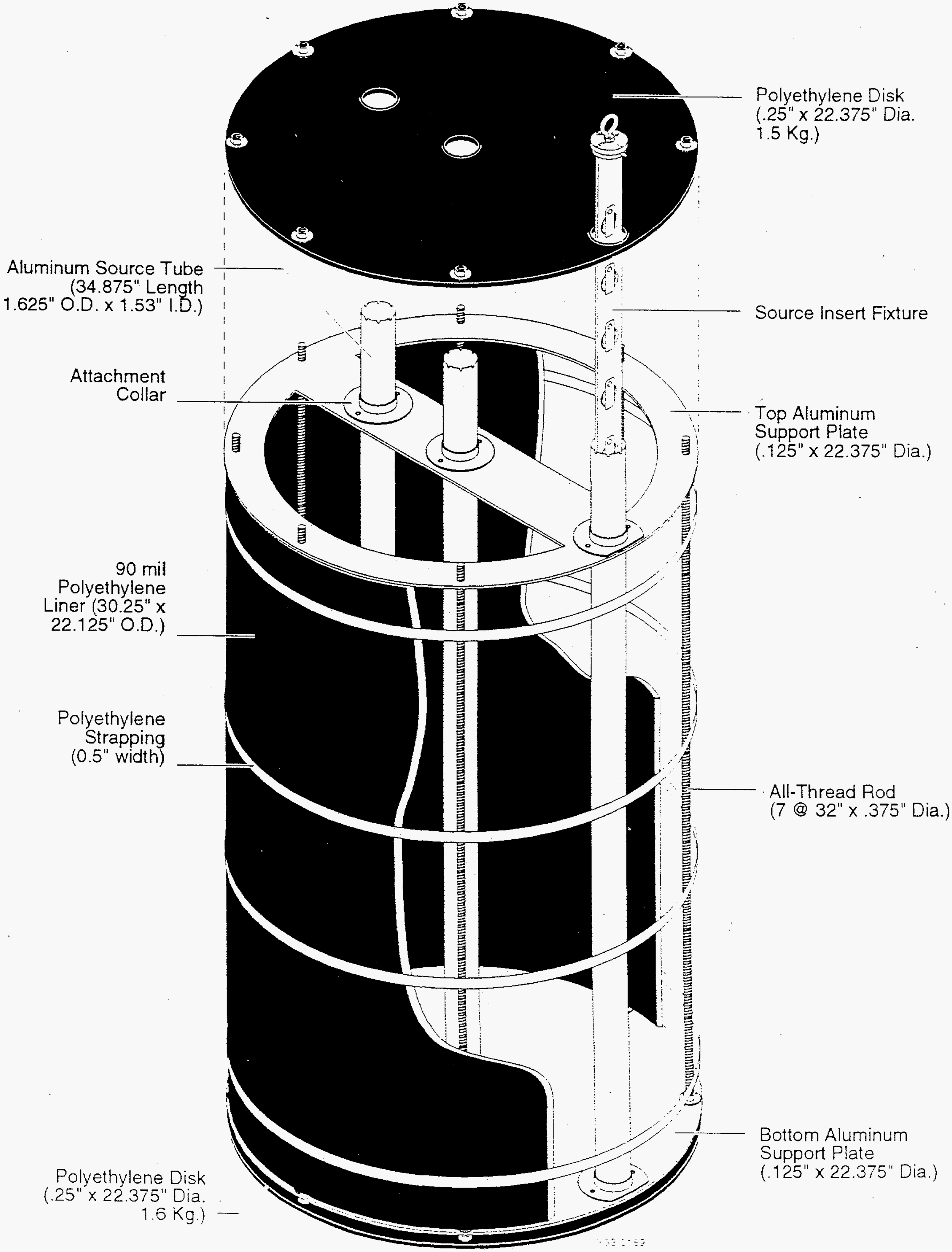


Figure D-1. Type 1 zero matrix drum internal support structure.

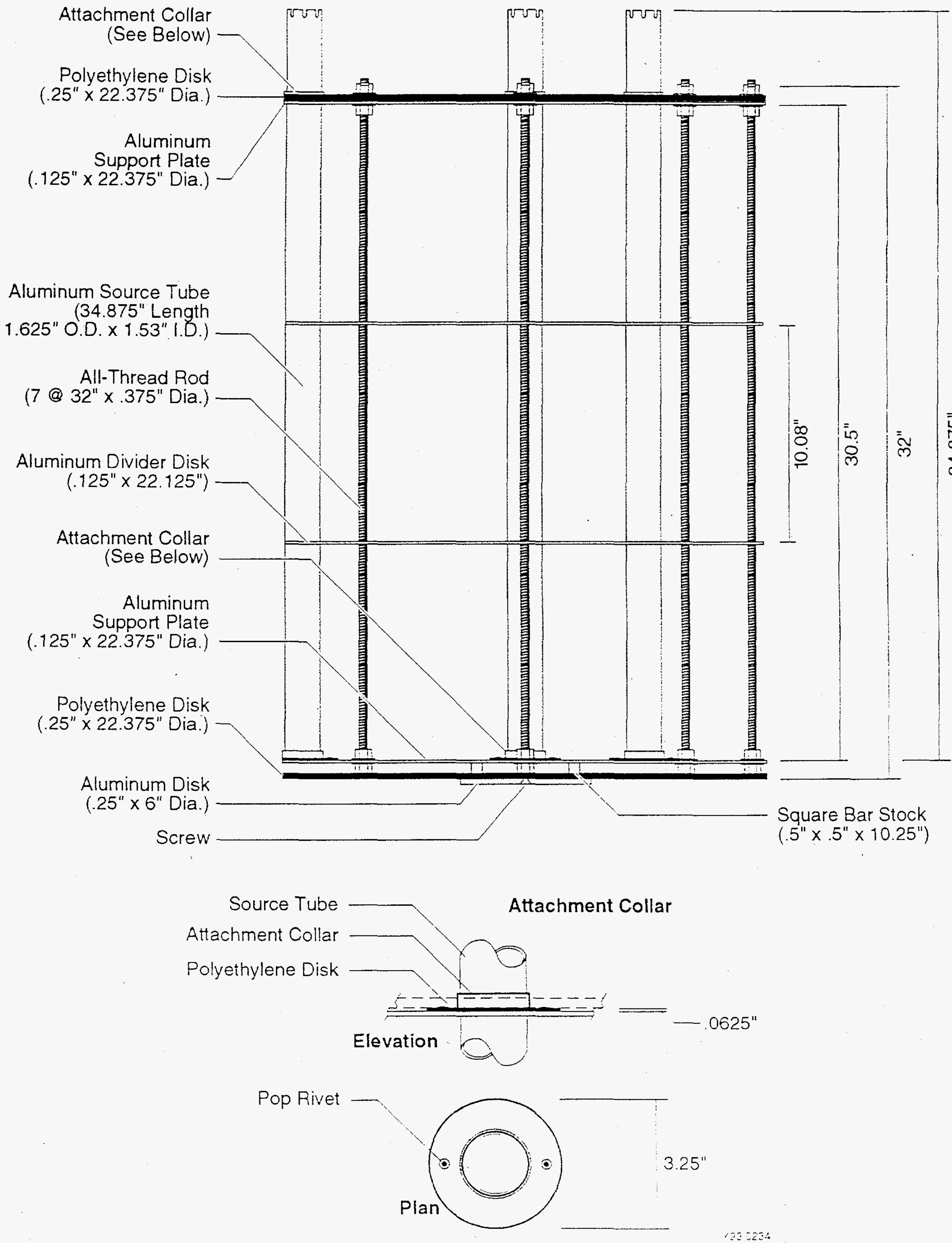
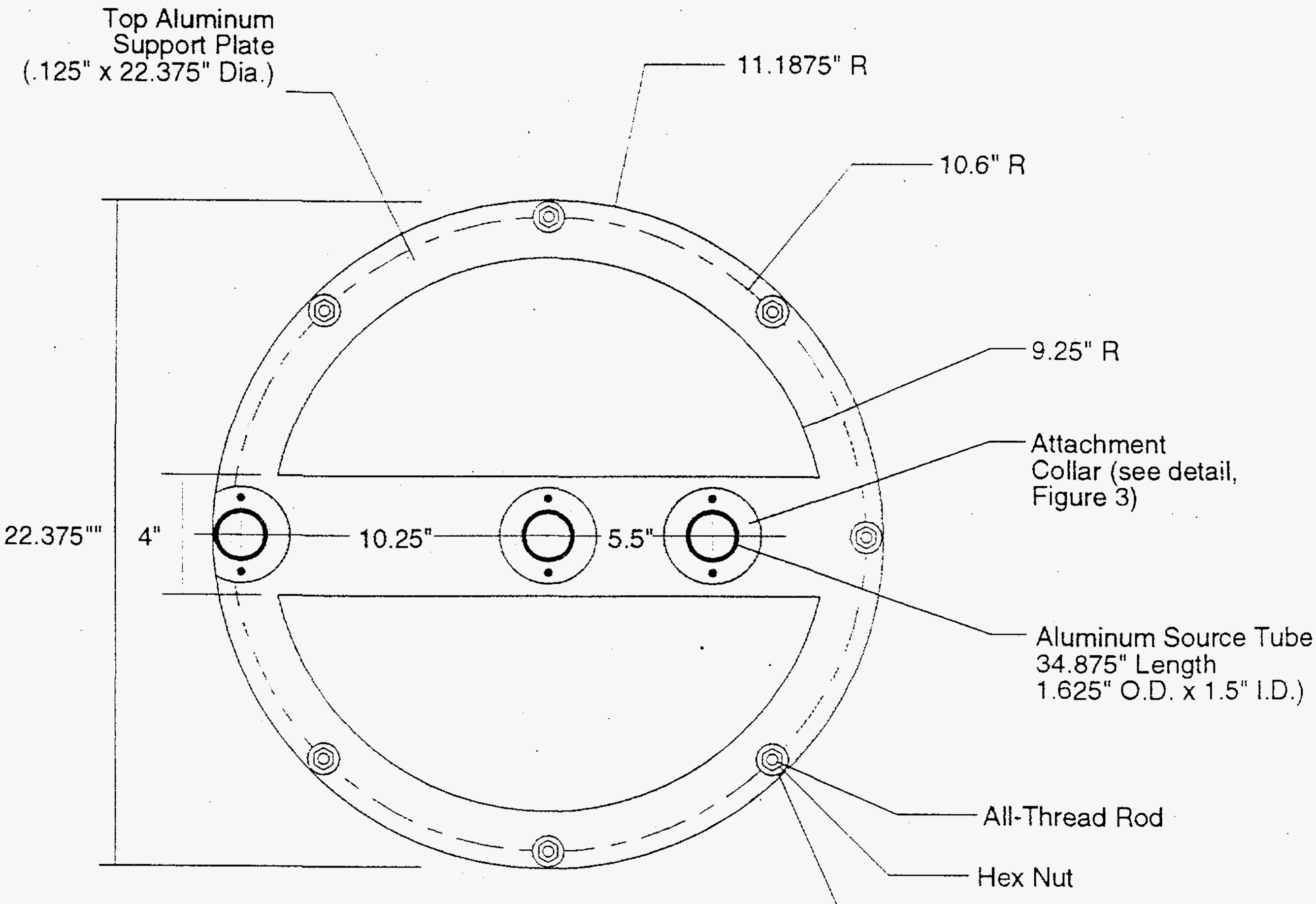
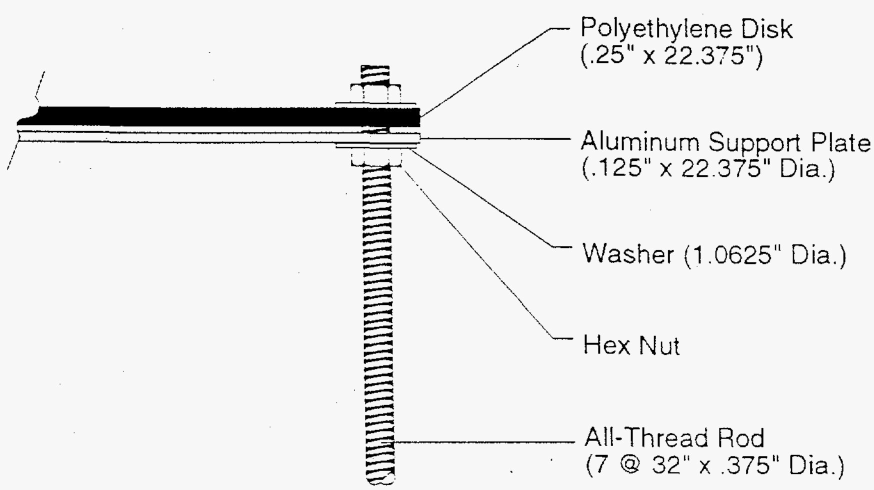


Figure D-2. Type I zero matrix drum details.



Aluminum Support Plate Plan



Attachment Detail

Figure D-3. Type 1 zero matrix drum details.

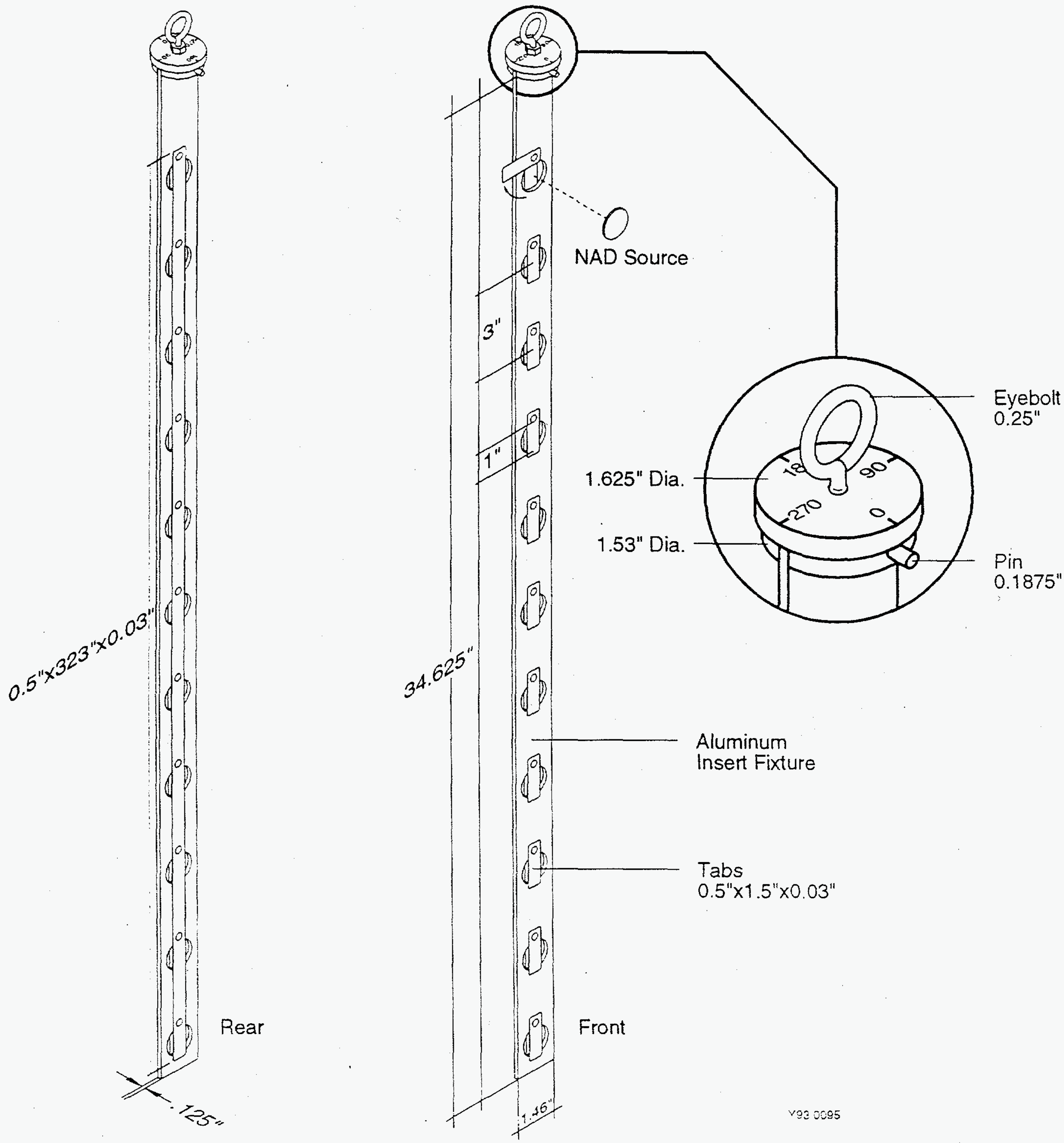


Figure D-4. Type 1 zero matrix drum source insert fixture.

from 0° to 360° in increments of 90°, and is keyed at 45° angular indexes on the circumference of the source insert tube itself, as illustrated in Figure D-5. These angular indexes are further referenced to another set of angular indexes on the drum lid (see Figure D-5). This angular demarcation arrangement allows for the quantification of drum neutron counter response as a function of space in the cylindrical coordinate system. It allows for the quantification of angular dependencies associated with the orientation of the source within the source insert tube.

Once fabricated, the entire internal matrix/source support assembly is fitted with a 90-mil rigid polyethylene liner. As the internal structure dimensions are not exactly those of the internal dimensions of a 90-mil rigid liner, the liner bottom is cut out, and the cylinder side 30.25-in. in height and weighing 5,000 grams is sliced open vertically and fit over the support assembly. The 90-mil rigid liner bottom is represented by placing a 0.25-in. thick, 22-3/8-in. diameter, 1,550 gram polyethylene plate in the drum bottom below the bottom aluminum support plate. Cutouts have been made in a similar top polyethylene plate to accept the source insert tube penetrations and simulate the liner lid. The simulated liner lid, 1050 grams, is then placed on to the support structure assembly as designed. The support structure assembly complete with 90-mil liner is lowered into a 55-gallon drum and the drum lid installed to conclude the fabrication process.

More detailed information concerning the Type 1 zero matrix drum can be found in Reference D-1.

D-1.2 Type 2 Zero Matrix Drum

A three-dimensional view of the internal source matrix/support assembly is shown in Figure D-6 with elevation and plan views as shown in Figures D-7 and D-8, respectively. The internal support consists of 1/8-in. thick, 22-3/8-in. diameter top and bottom aluminum support plates connected by seven 3/8-in. carbon steel support rods. Aluminum (type 6061) is used in the support structure to the extent practicable to minimize neutron interactions.

A significant component of the internal matrix/source support assembly are the three aluminum source insert tubes depicted in Figures D-6, D-7, and D-8. These source insert tubes run vertically through the drum and are welded to the top and bottom aluminum support plates. The source insert tubes are 32-7/8 inches high, with internal and external dimensions of 2.15-in. and 2.25-in., respectively. One source insert tube is fixed at the drum center, one at a radius of 5.5-in., and the third at a radius of 9.05-in.

To precisely locate a source at a given location within the source insert tube, source insert fixtures were fabricated from aluminum. Source insert fixtures (see Figures D-9 through D-11) were designed to accommodate a variety of source material configurations. Aluminum plunger rods are provided to secure the source material in a given location.

Once fabricated, the entire internal matrix/source support assembly is fitted with a 90-mil rigid polyethylene liner. As the internal structure dimensions are not exactly those of the internal dimensions of a 90-mil rigid liner, the liner bottom is cut out, and the cylinder side 30.25-in. in height and weighing 5,000 grams is sliced open vertically and fit over the support assembly. The 90-mil rigid liner bottom is represented by placing a 0.25-in. thick, 22.375-in. diameter, 1,550 gram polyethylene plate in the drum bottom below the bottom aluminum support plate. Cutouts have been

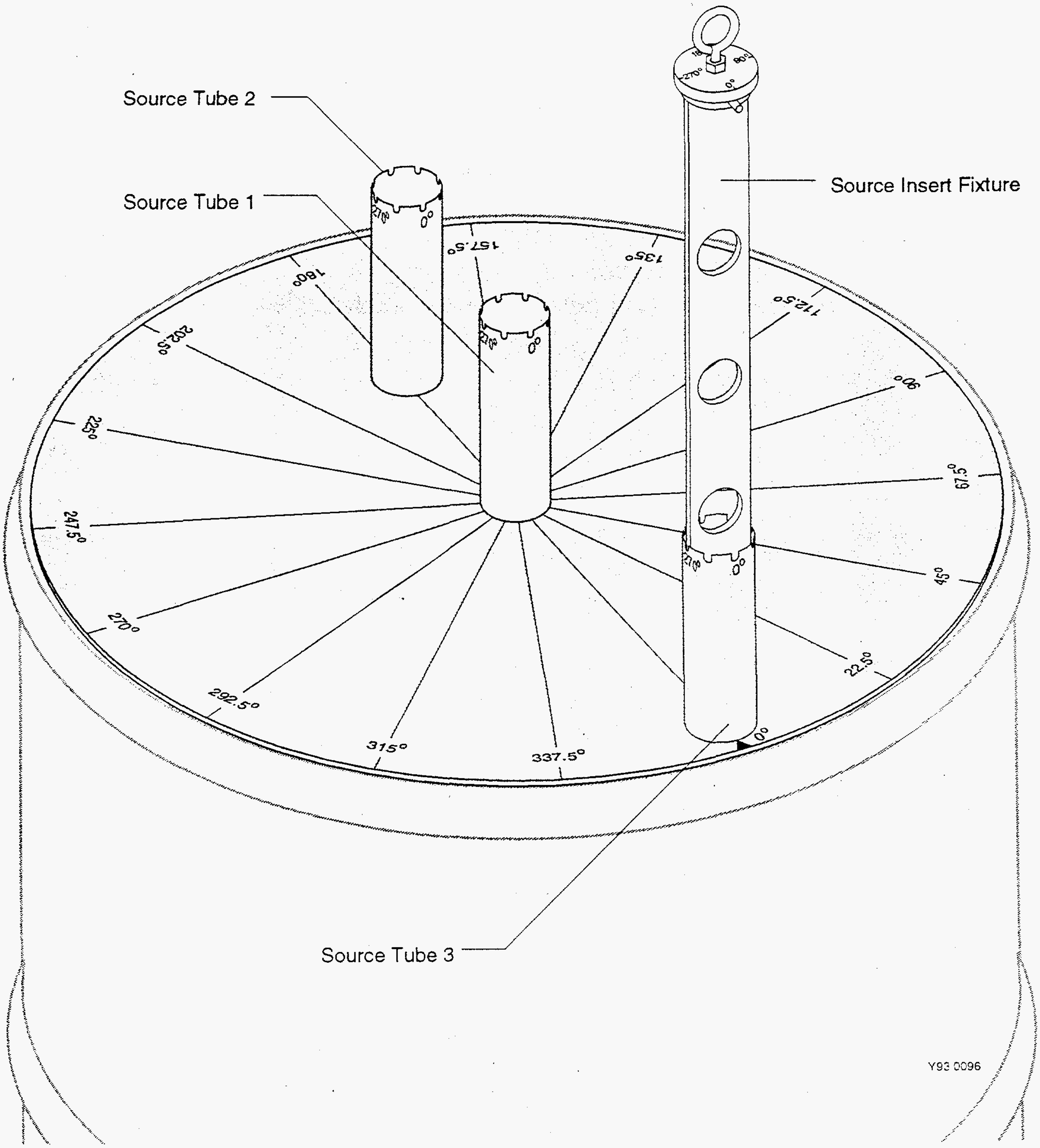


Figure D-5. Type 1 zero matrix drum steel cover angular demarcations.

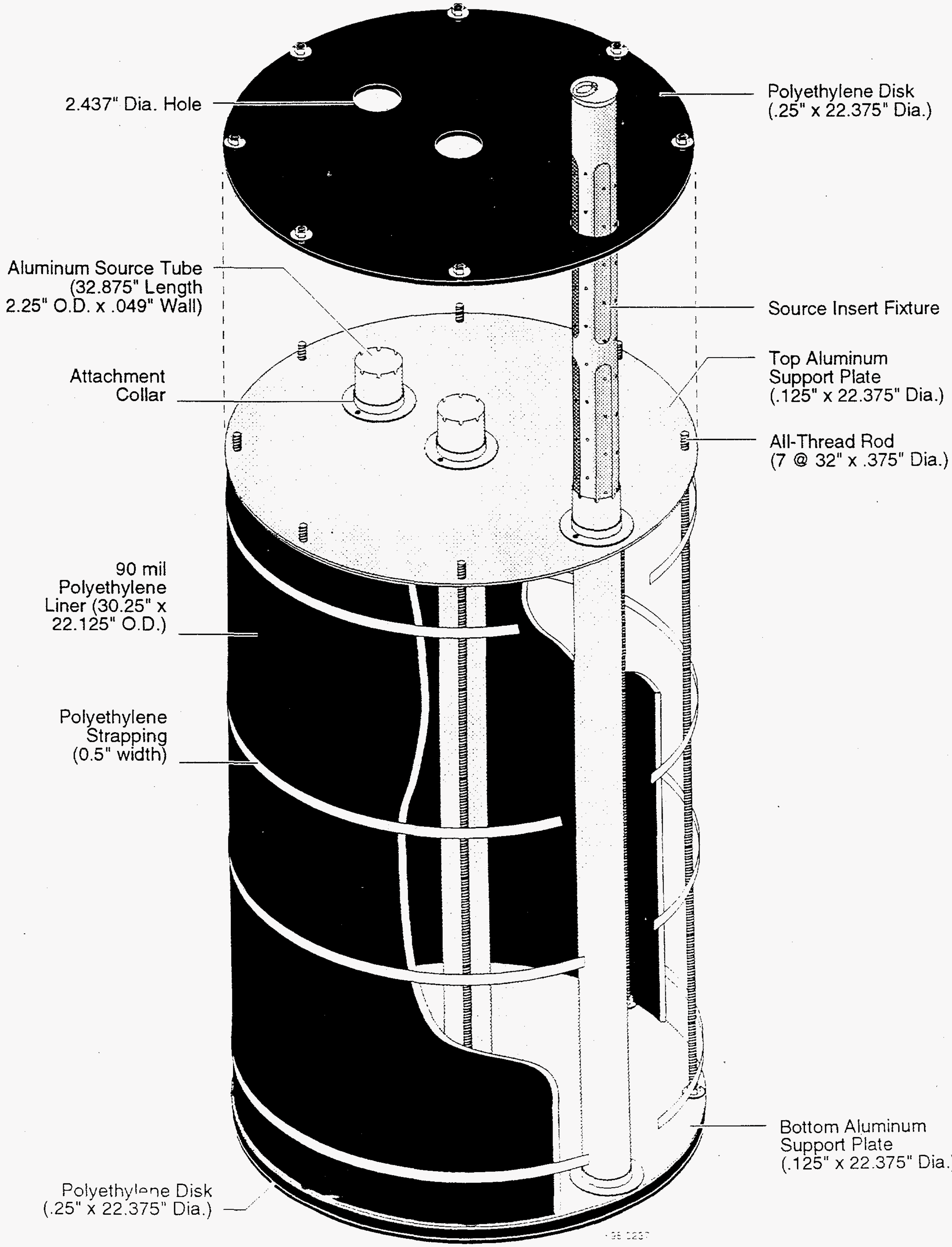


Figure D-6. Type 2 zero matrix drum internals.

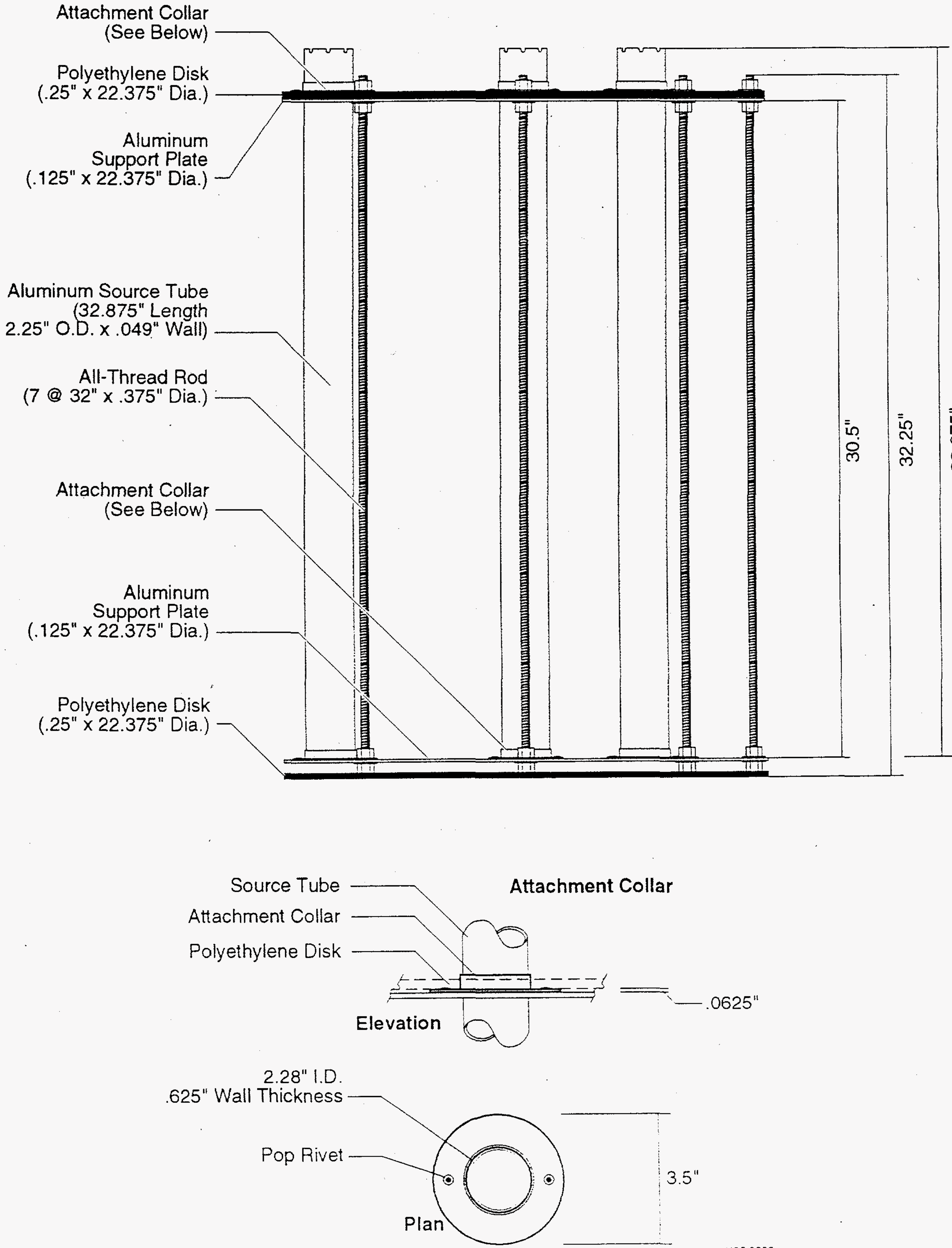
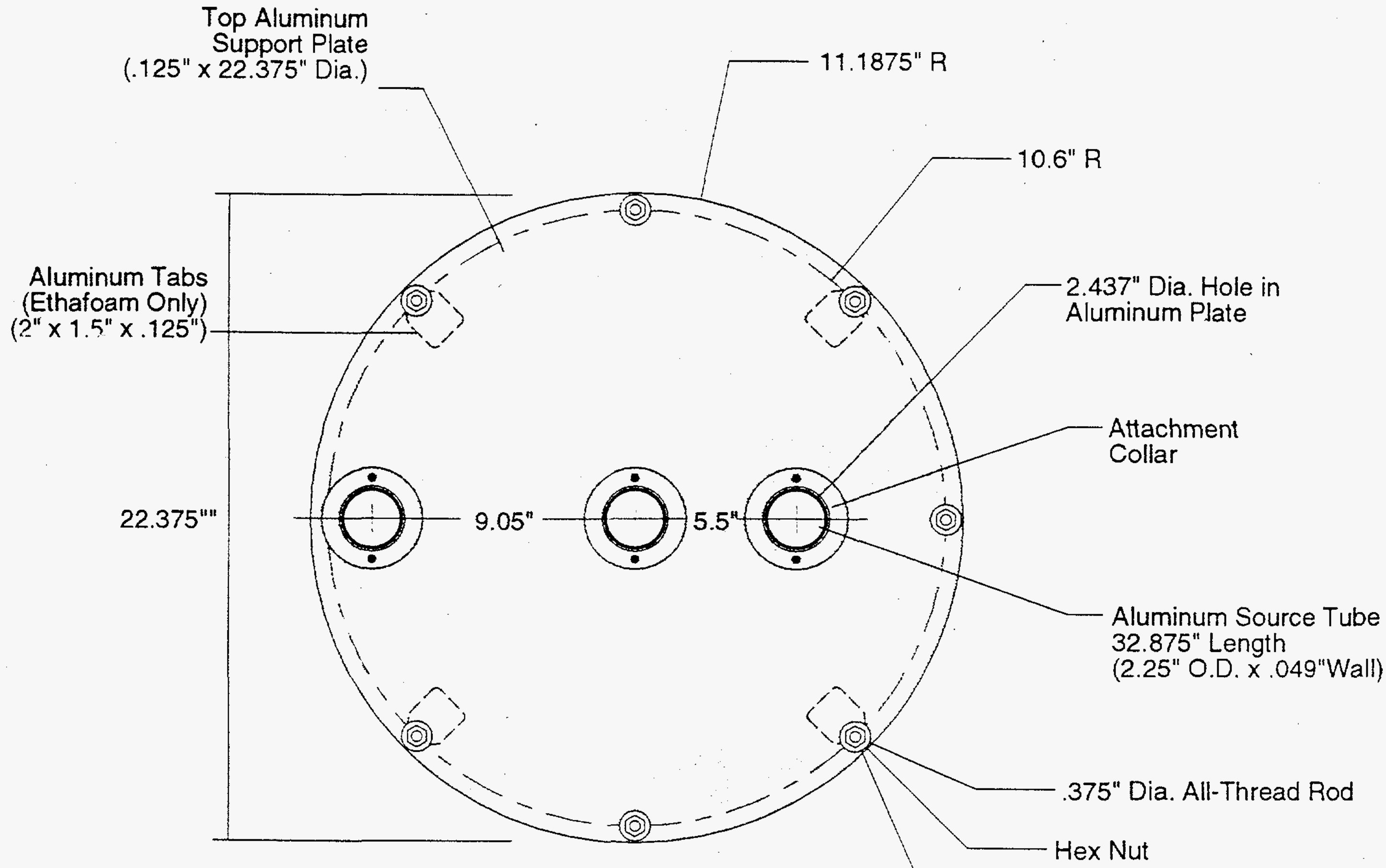
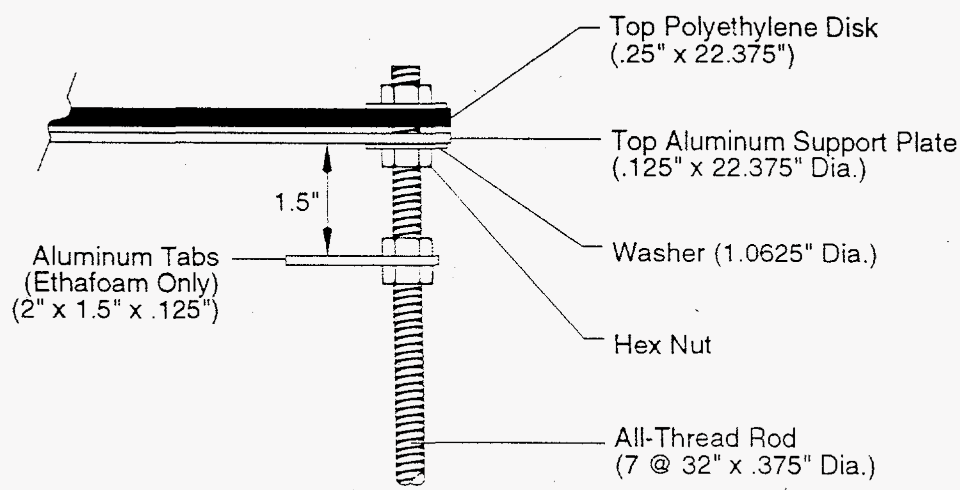


Figure D-7. Type 2 zero matrix drum support structure (elevation).



Top Aluminum Support Plate Plan



Top Plate Attachment Detail

Y95 0240

Figure D-8. Type 2 zero matrix drum support structure (plan).

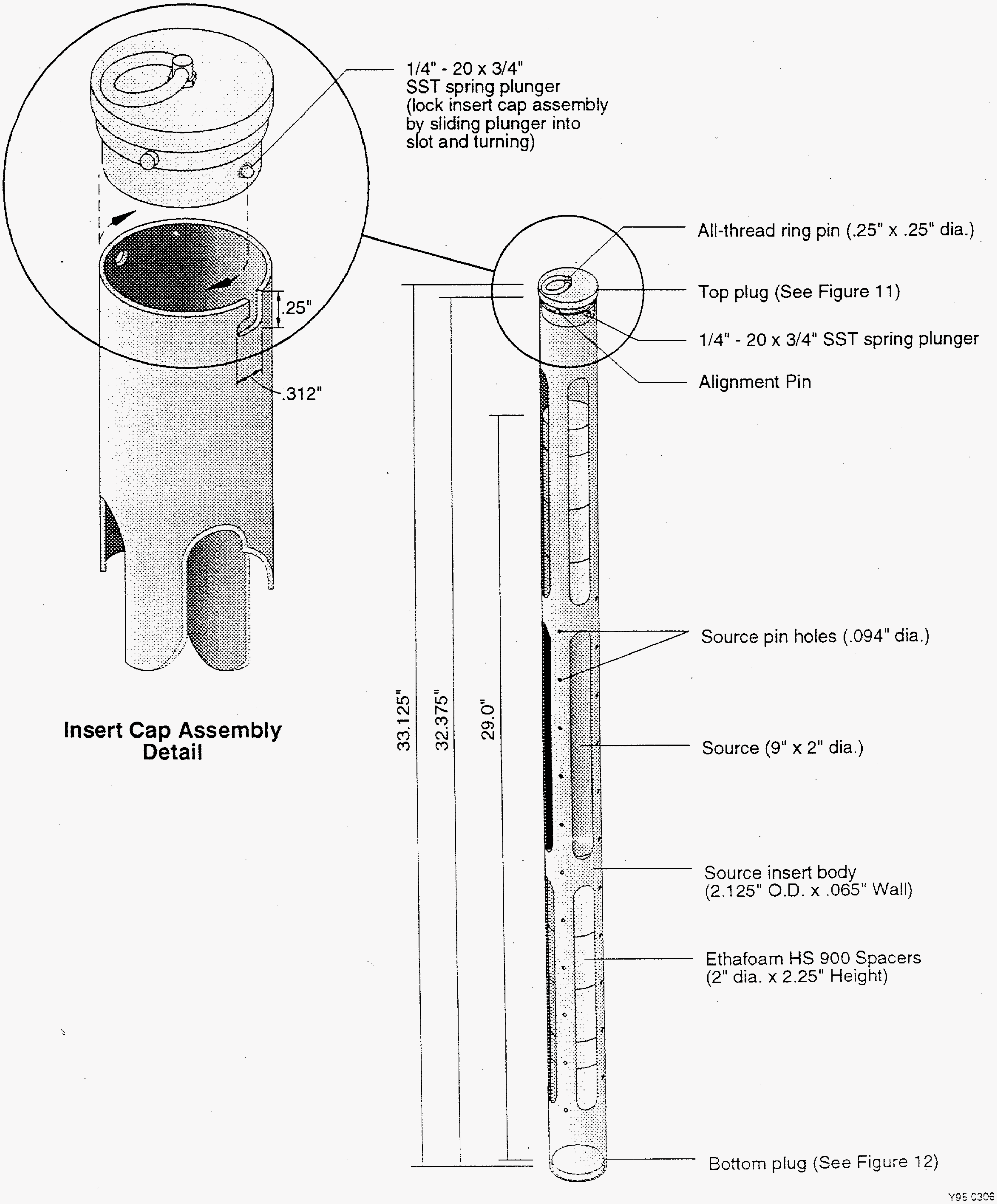


Figure D-9. Type 2 zero matrix drum source insert fixture.

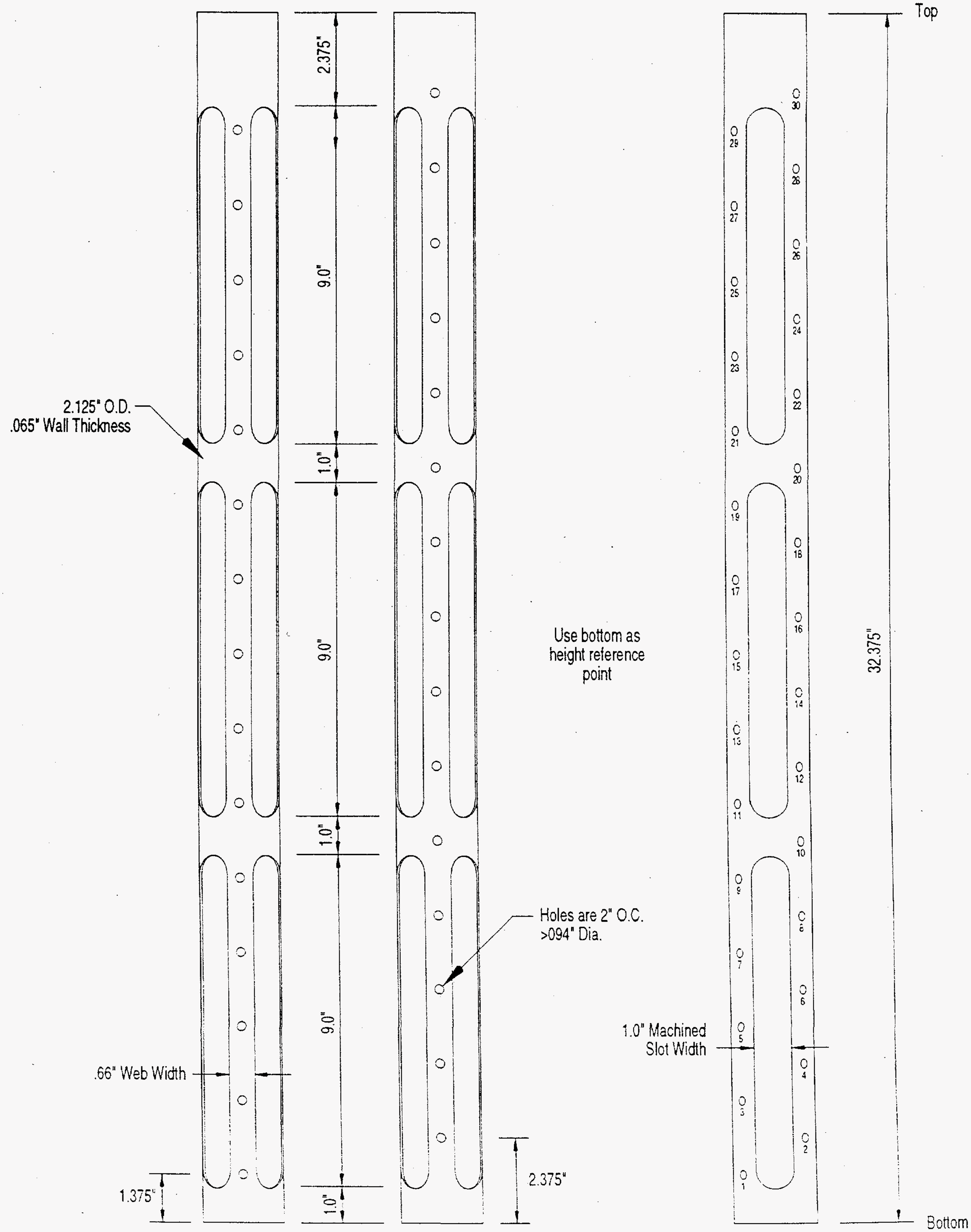


Figure D-10. Type 2 zero matrix drum source insert fixture details.

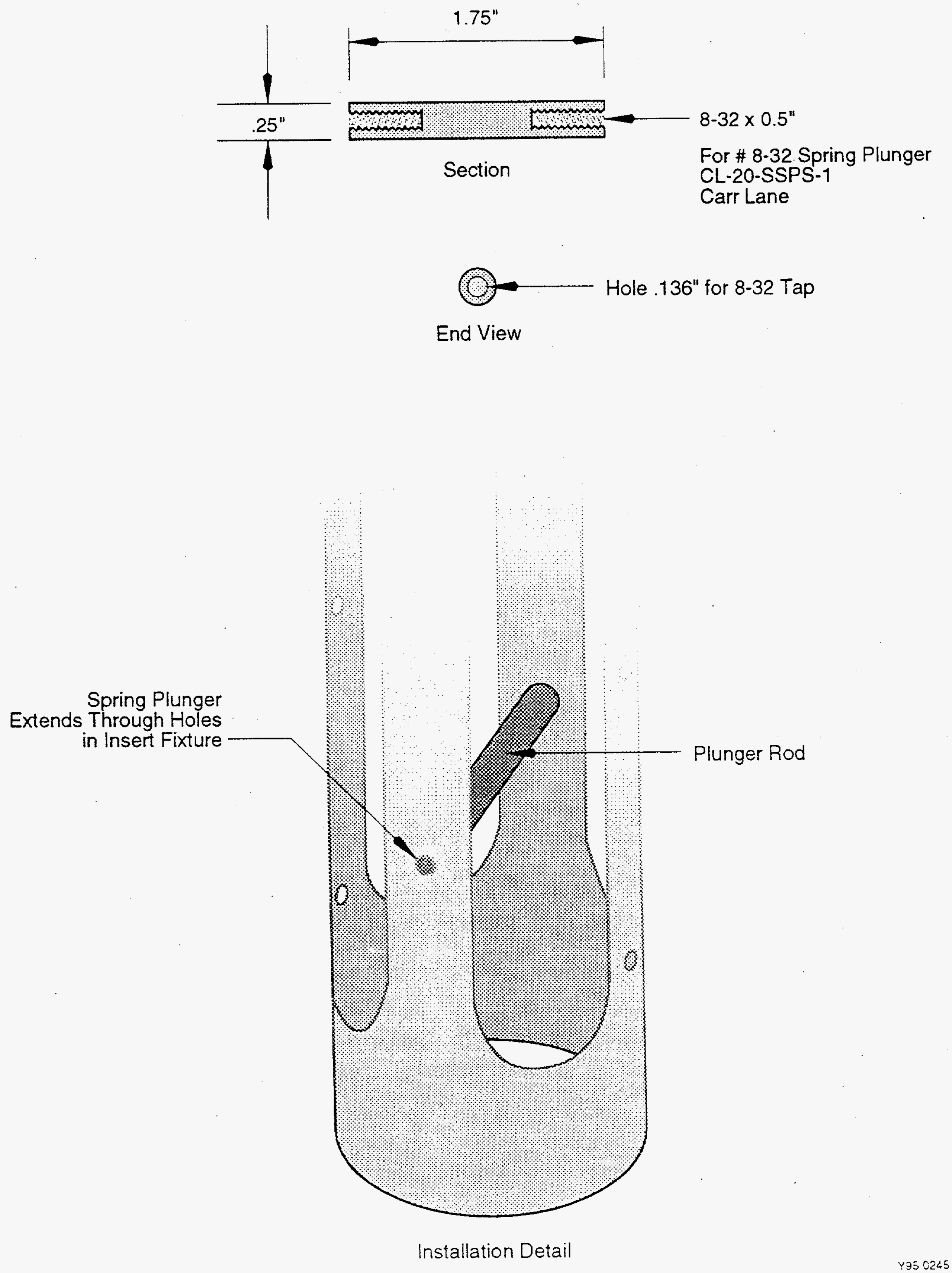


Figure D-11. Type 2 zero matrix drum source insert fixture plunger rod details.

made in a similar top polyethylene plate to accept the source insert tube penetrations and simulate the liner lid. The simulated liner lid, 1,050 grams, is then placed on to the support structure assembly as designed. The support structure assembly complete with 90-mil liner is lowered into a 55-gallon drum and the drum lid installed to conclude the fabrication process.

More detailed information concerning the Type 2 zero matrix drum can be found in Reference D-2.

D-1.3 Type 3 Zero Matrix Drum

A three-dimensional view of the internal source matrix/support assembly is shown in Figure D-12 and an elevation view in Figure D-13. The horizontal rod design is quite versatile for source material placement and has the advantage of low mass and insignificant neutron/gamma emission interference.

The rod configuration consists of five horizontal rods aligned such that they divide the drum cross-sectional circle into two equal halves. The five rods are spaced vertically at the increments shown in Figure D-13. In addition, the rods are oriented at an angle of 36° to each other as shown in Figure D-12. Several rod collar fixtures (Figures D-12 and D-14) are provided for each rod to which source material is fixed. The rod source collars can be positioned and fixed at any radius along the rod. They can be used in combination to support source material of various size and configuration.

More detailed information concerning the Type 3 zero matrix drum can be found in Reference D-3.

D-2. ETHAFOAM MATRIX DRUMS

There are two types of ethafoam matrix drums: one consists of a Type 1 zero matrix drum in which 13 2-1/4-in. by 20-3/4-in. diameter disks of ethafoam HS 900 have been stacked, and a Type 2 zero matrix drum in which 13 2-1/4-in. by 20-3/4-in. diameter disks of ethafoam have been stacked (see Figure D-15).

More detailed information concerning the Type 2 ethafoam drum can be found in Reference D-2.

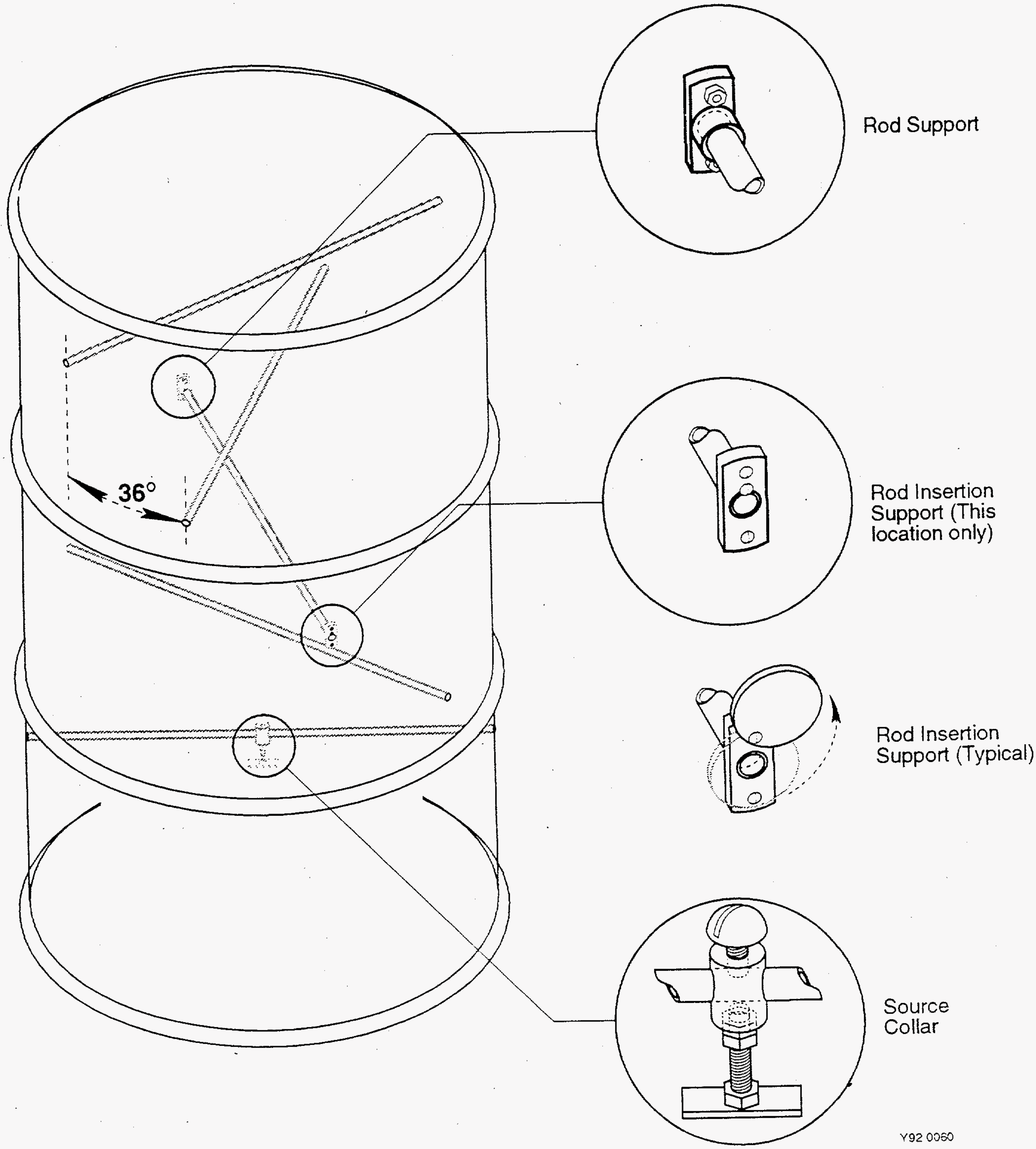


Figure D-12. Type 3 zero matrix drum.

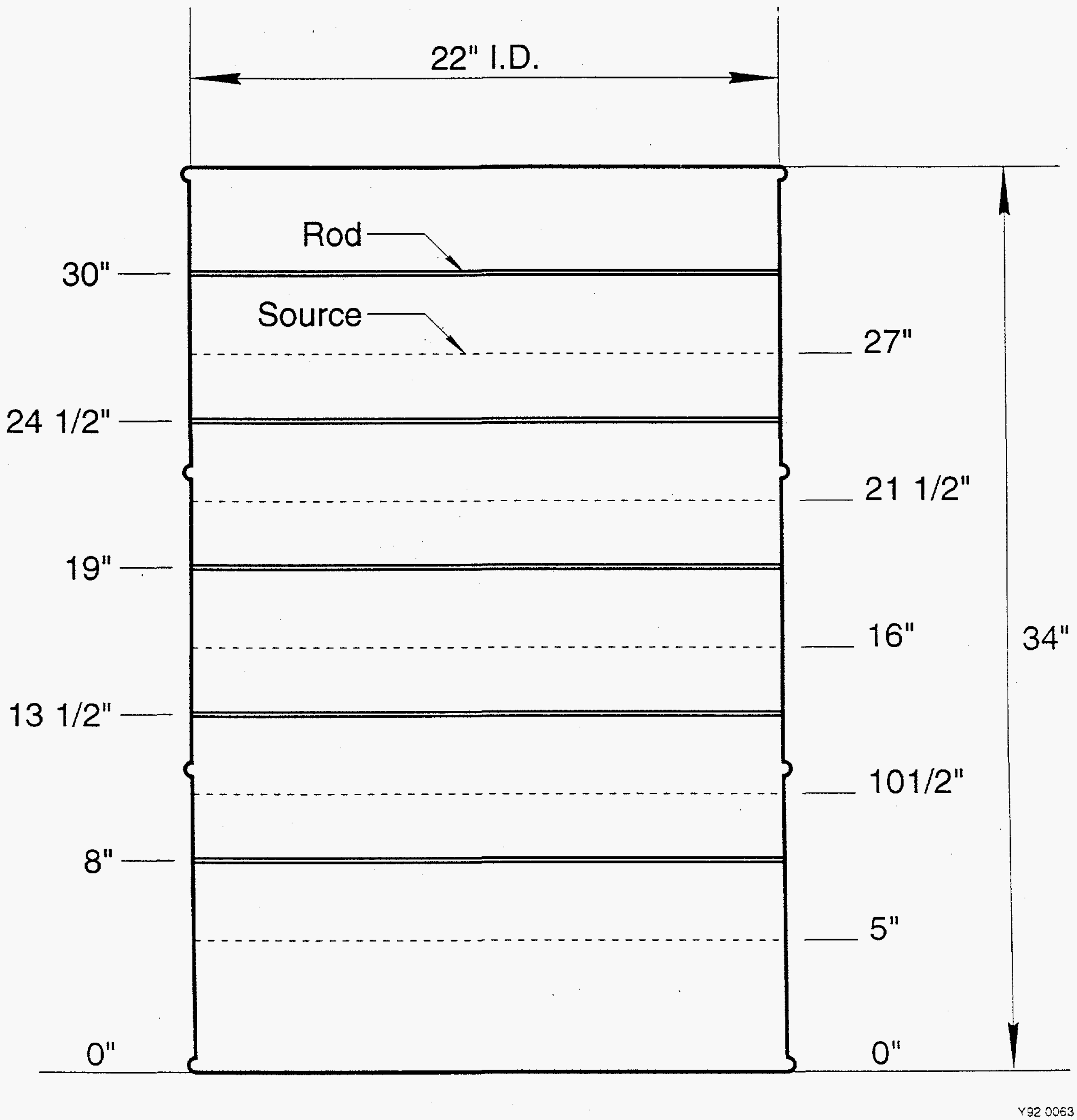


Figure D-13. Type 3 zero matrix drum rod and source locations.

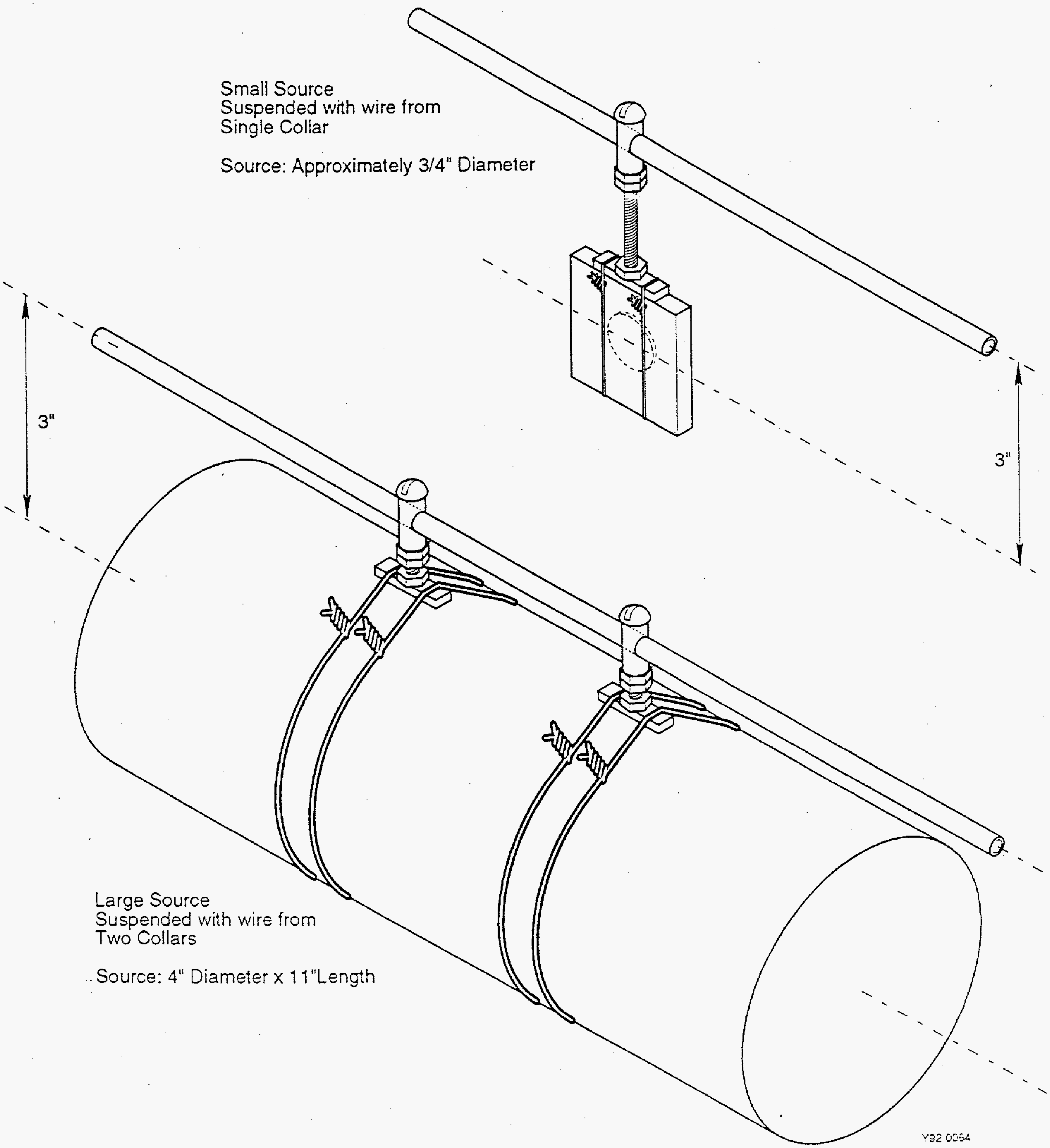


Figure D-14. Type 3 zero matrix drum collars with sources.

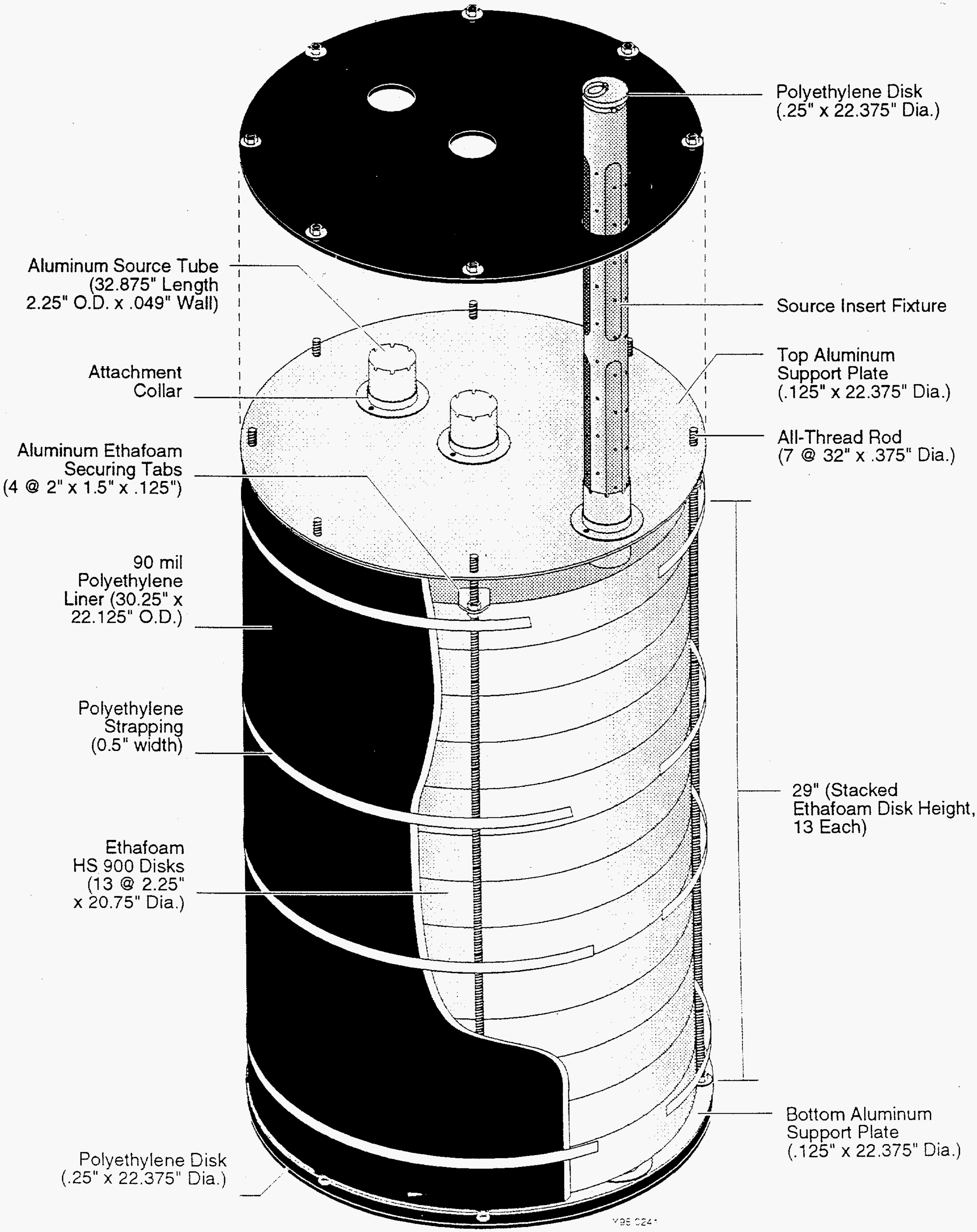


Figure D-15. Type 2 ethafoam matrix drum.

D-3. REFERENCES

D-1. Idaho National Engineering Laboratory, "Content Code 480/481 Calibration Drum Design and Sata, Heterogeneous Configuration—Mixed Metals," EDF# RWMC-598, February 22, 1993.

D-2. U.S. Department of Energy, "Performance Demonstration Program Plan for Nondestructive Assay for the TRU Waste Characterization Program," CAO-94-1045, Revision 0, March 1995.

D-3. Idaho National Engineering Laboratory, "Zero Matrix Calibration Drum Design and Data," EDF# RWMC-532, April 20, 1992.

Appendix E

Descriptions of Surrogate Drums

Appendix E

Descriptions of Surrogate Drums

This appendix contains brief descriptions (including drawings) of the INEL surrogate waste form drums. Currently, there are nine surrogate drums for content codes 001, 300, 330, 371, 440, 442, and 480/481. More detailed information about these drums can be found in the stated references.

E-1. SURROGATE FOR CONTENT CODE 001—UNCEMENTED INORGANIC SLUDGE

This surrogate was designed to simulate content code 001, uncemented inorganic sludge. This waste category comprises wet sludge precipitates generated by processing liquid wastes such as ion exchange column effluents, distillates, caustic soda solution, etc. produced by Rocky Flats plutonium recovery operations.

This surrogate is comprised of a modified Type 1 zero matrix drum (with the aluminum source tubes having been replaced with ABS plastic tubes) which has been filled to a height of approximately 20 inches with cement, sludge surrogate, and water. There is a 2-inch bottom layer of cement (15.7 pounds), a 19-inch middle layer of sludge surrogate (281.5 pounds), and approximately a 1 inch layer of water on top of the sludge. The major components of the sludge surrogate are water (79.4%), nitrate (4.00%), sulfate (2.0%), phosphate (1.23%), chloride (0.53%), silicon dioxide (1.08%), iron (1.16%), magnesium (1.04%), calcium (1.72%), aluminum (0.90%), sodium (1.93%), potassium (0.59%).

Figure E-1 shows a drawing of this surrogate drum. More detailed information can be obtained from Reference E-1.

E-2. SURROGATE FOR CONTENT CODE 300—GRAPHITE MOLDS

This surrogate was designed to simulate content code 300, graphite molds. This waste category comprises graphite molds used in casting plutonium metal.

This surrogate is comprised of a Type 1 zero matrix drum filled with seven layers of reactor grade graphite (199.4 pounds total graphite). Figures E-2 through E-5 show drawings of this surrogate drum. More detailed information can be obtained from Reference E-2.

E-3. SURROGATE FOR CONTENT CODES 330/336—COMBUSTIBLES (HETEROGENEOUS)

This surrogate is one of two that were designed to simulate content code 330, combustibles. These waste categories are comprised primarily of cellulosic materials, plastic, cloth, and vermiculite. Typical components include combustible materials such as paper, rags, plastics, surgeons' gloves cardboard, wood, polyethylene bottles, etc.

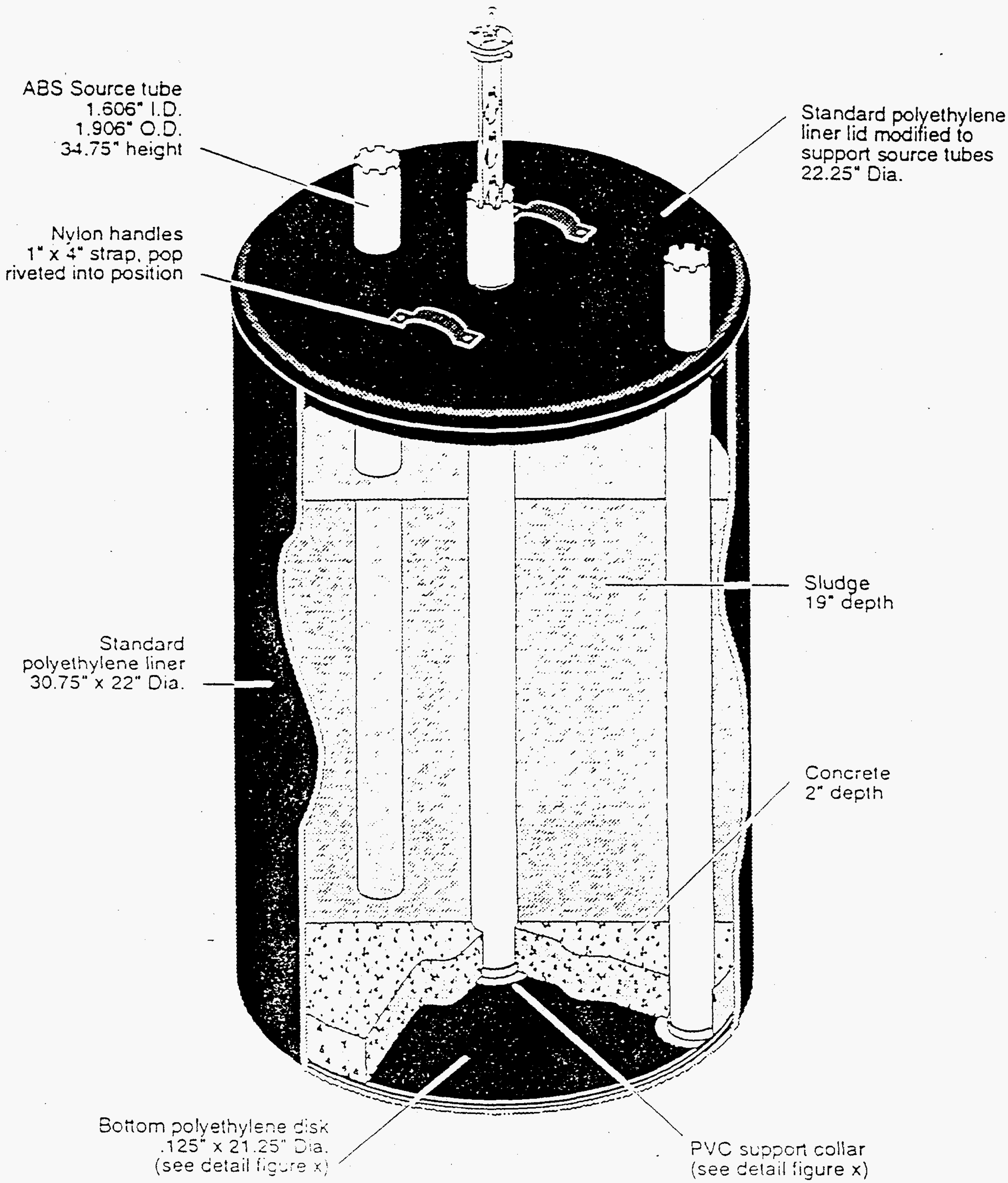


Figure E-1. Uncemented inorganic sludge surrogate drum.

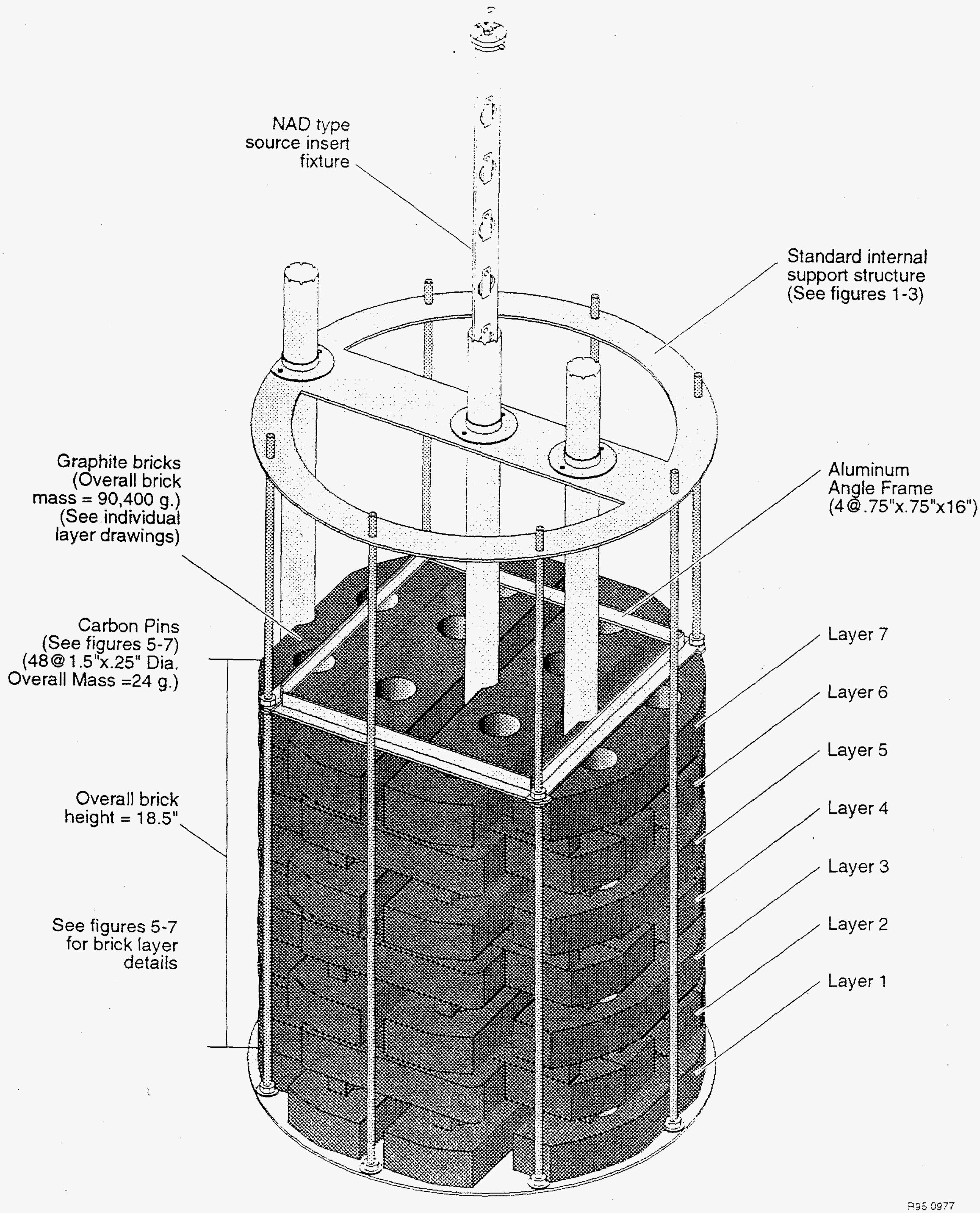


Figure E-2. Graphite surrogate drum internal configuration.

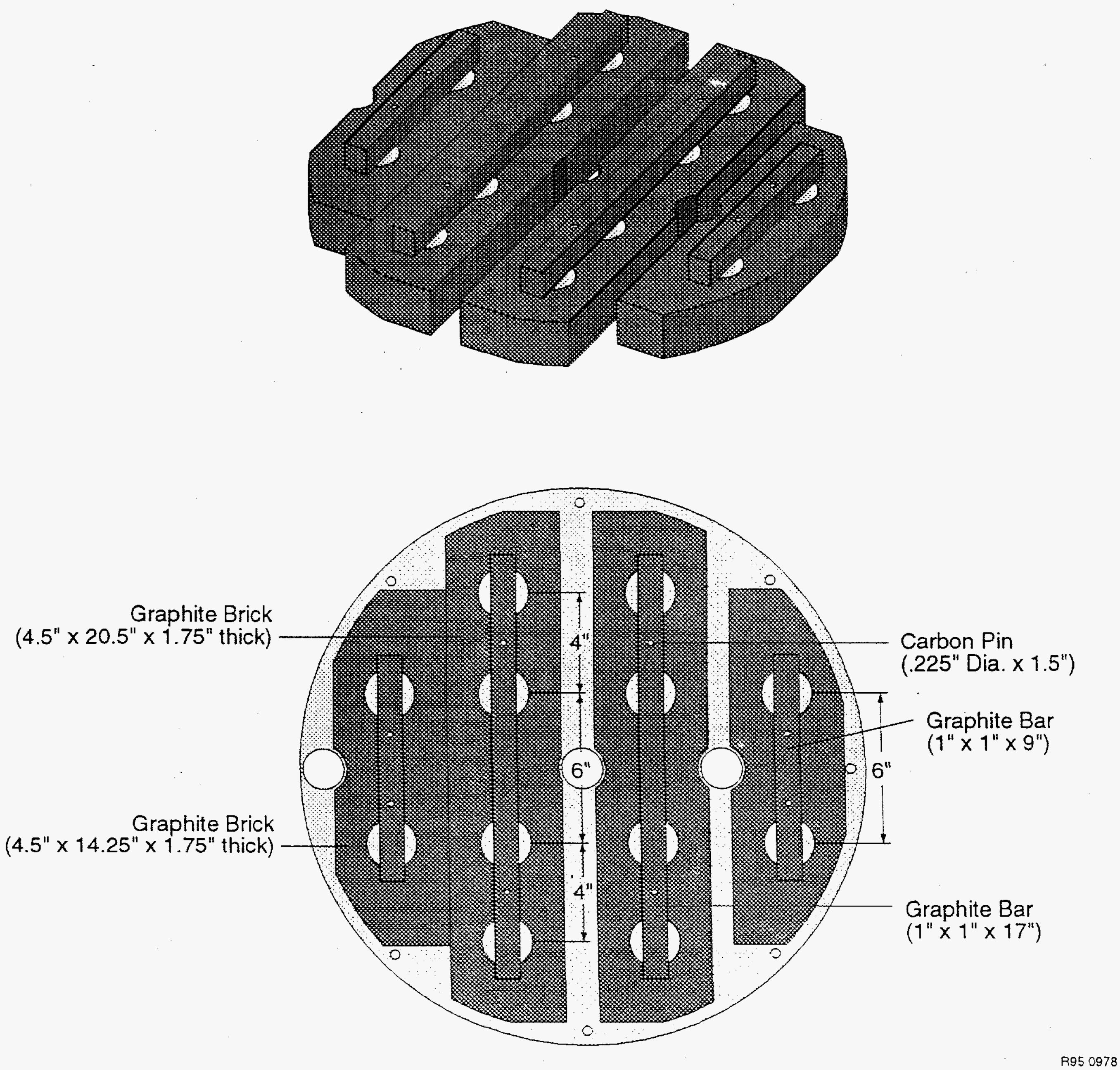


Figure E-3. Graphite layer 1, 3, and 5 detail.

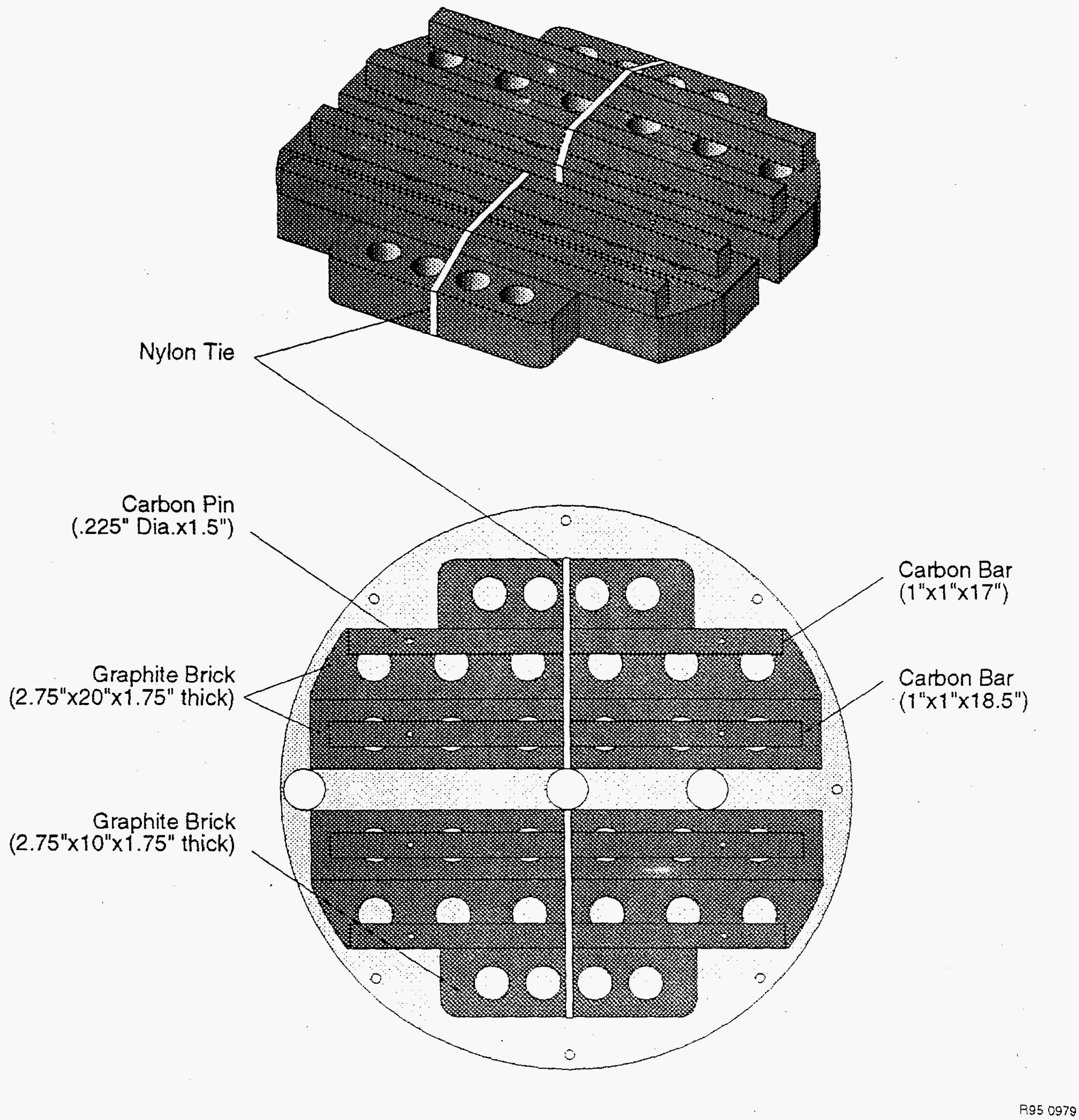


Figure E-4. Graphite layer 2, 4, and 6 detail.

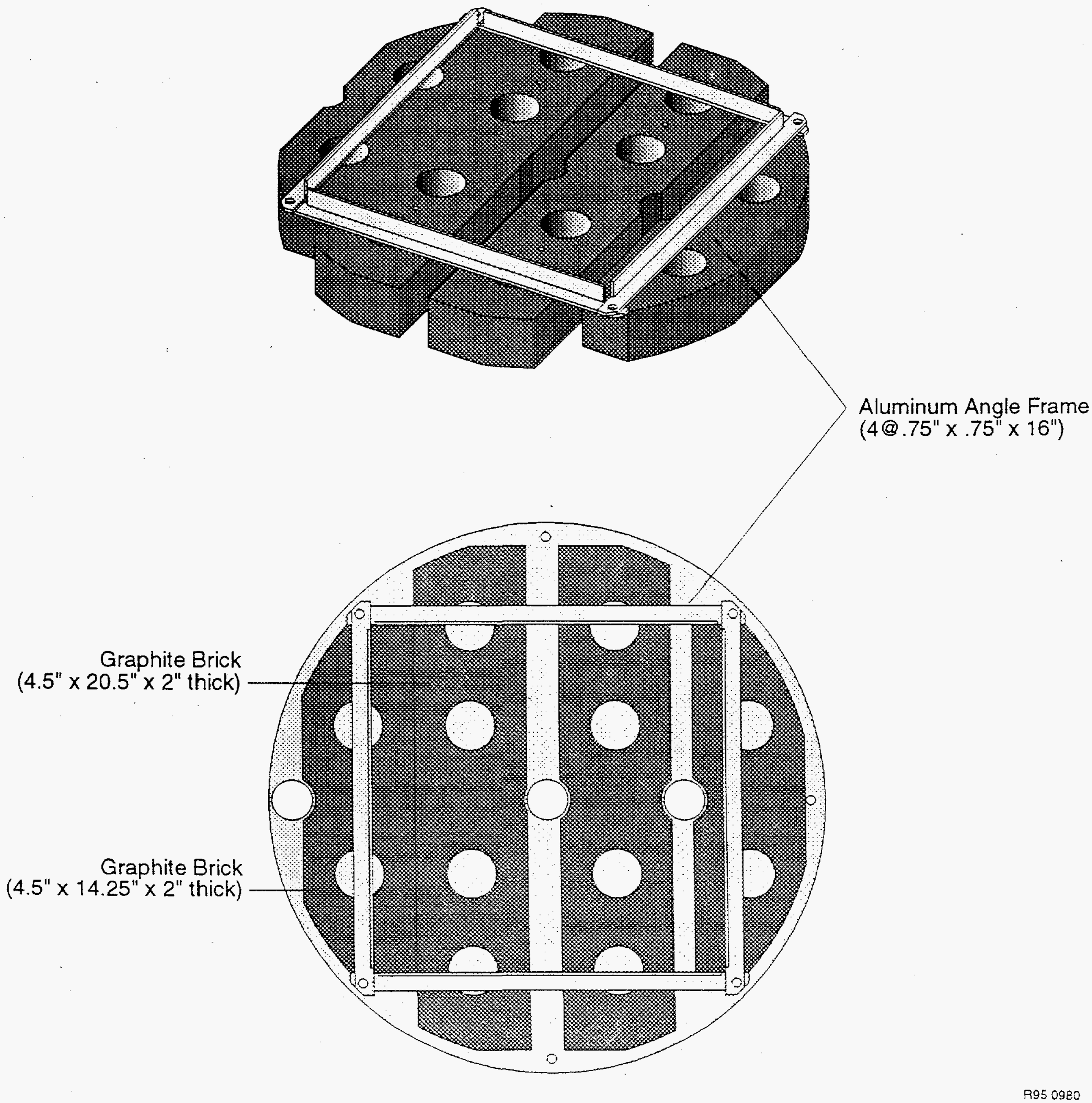


Figure E-5. Graphite layer 7 detail.

This surrogate drum attempts to simulate the heterogeneous mixed combustibles. It is comprised of cotton glove liners (10.2 pounds), KimWipes (4.8 pounds), latex gloves (7.4 pounds), Tyvek coveralls (2.5 pounds), Rag-on a-Roll (13.7 pounds), KimTex Wipers (3.1 pounds), cotton shoe covers (4.1 pounds), and Rag-on-a-Roll/plastic sheets (19.4 pounds). Figure E-6 shows a drawing of this surrogate drum. More detailed information can be obtained from Reference E-3.

E-4. SURROGATE FOR CONTENT CODES 330/336—COMBUSTIBLES (VERMICULITE AND PLASTICS)

This surrogate is the second of two that were designed to simulate content codes 330/336, combustibles. These waste categories are comprised primarily of cellulosic materials, plastic, cloth, and vermiculite. Typical components include combustible materials such as paper, rags, plastics, surgeons' gloves cardboard, wood, polyethylene bottles, etc. This surrogate drum attempts to simulate waste form drums containing vermiculite and plastics.

The surrogate is comprised of a Type 1 zero matrix drum which has been filled with 2.7 kg of polyolefin and 30 kg of vermiculite. Figure E-7 shows a drawing of this surrogate drum. More detailed information can be obtained from Reference E-4.

E-5. SURROGATE FOR CONTENT CODE 371—FIREBRICK

This surrogate was designed to simulate content code 371, firebrick. This waste category currently comprises waste consisting primarily of incinerator firebrick (although prior to 1974, construction bricks and cinderblocks were also included in this category).

The surrogate is comprised of a Type 1 zero matrix drum which has been modified using aluminum structural material to hold firebrick (see Figures E-8 to E-10). It contains three tiers (397 lbs total) of firebrick (95.7% Al_2O_3, 3.6% CaO, 0.1% SiO_2, 0.1% Fe_2O_3, 0.1% MgO). More detailed information can be obtained from Reference E-5.

E-6. SURROGATE FOR CONTENT CODE 440—GLASS

This surrogate was designed to simulate content code 440, glass. This waste category comprises glass in the form of sample vials and bottles, ion exchange columns, dissolver pots, laboratory glassware such as Pyrex flasks and beakers, glovebox windows (glass, Plexiglass, leaded glass), and crushed and ground glass.

This surrogate was constructed by filling a standard drum (with 90-mil rigid liner) with polyethylene bottles containing various glass and Pyrex glass vials. The total mass of glass in the matrix is 53.2 lbs. Figures E-11 to E-15 show drawings of this surrogate drum. More detailed information can be obtained from Reference E-6.

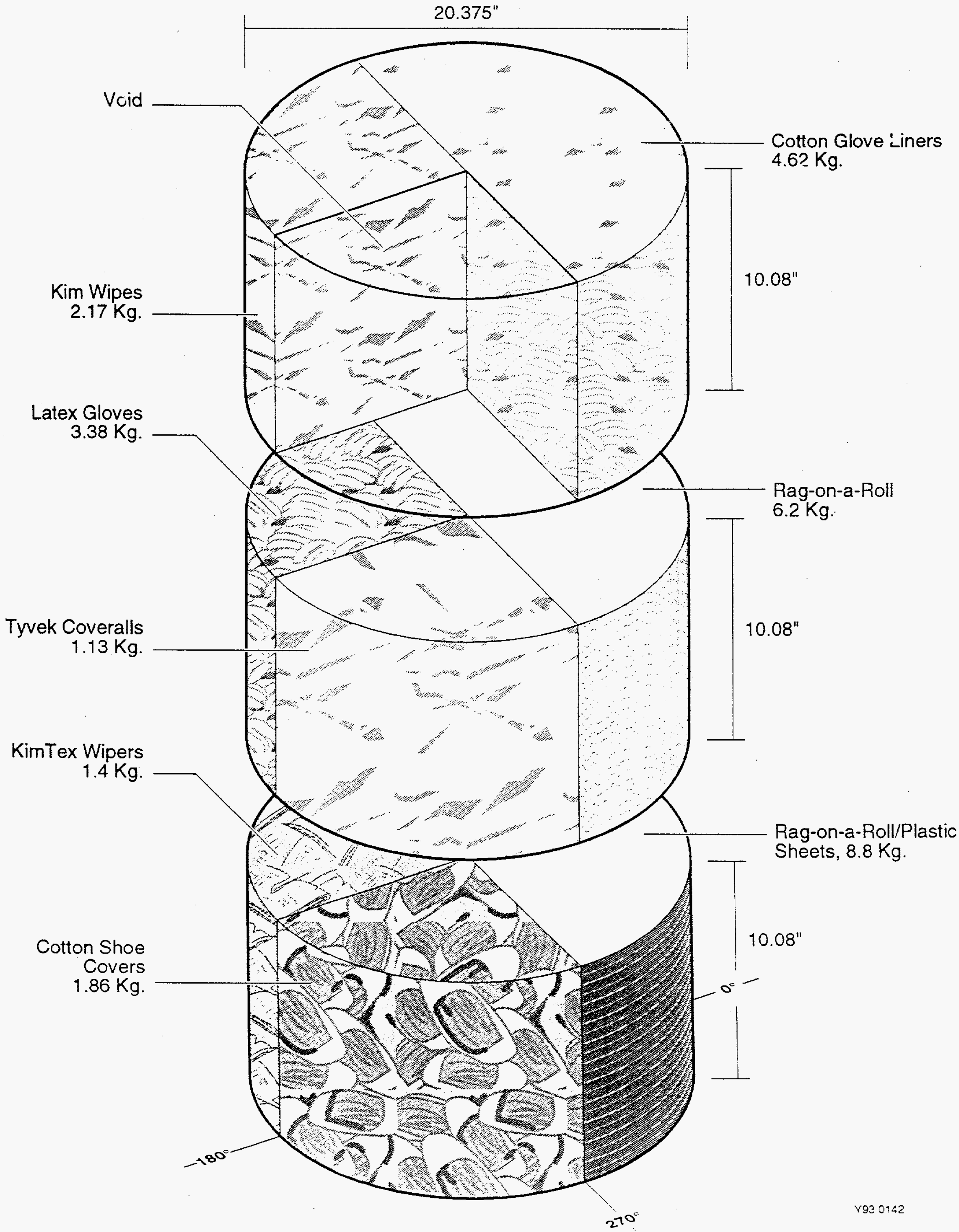


Figure E-6. Content code 330 matrix materials.

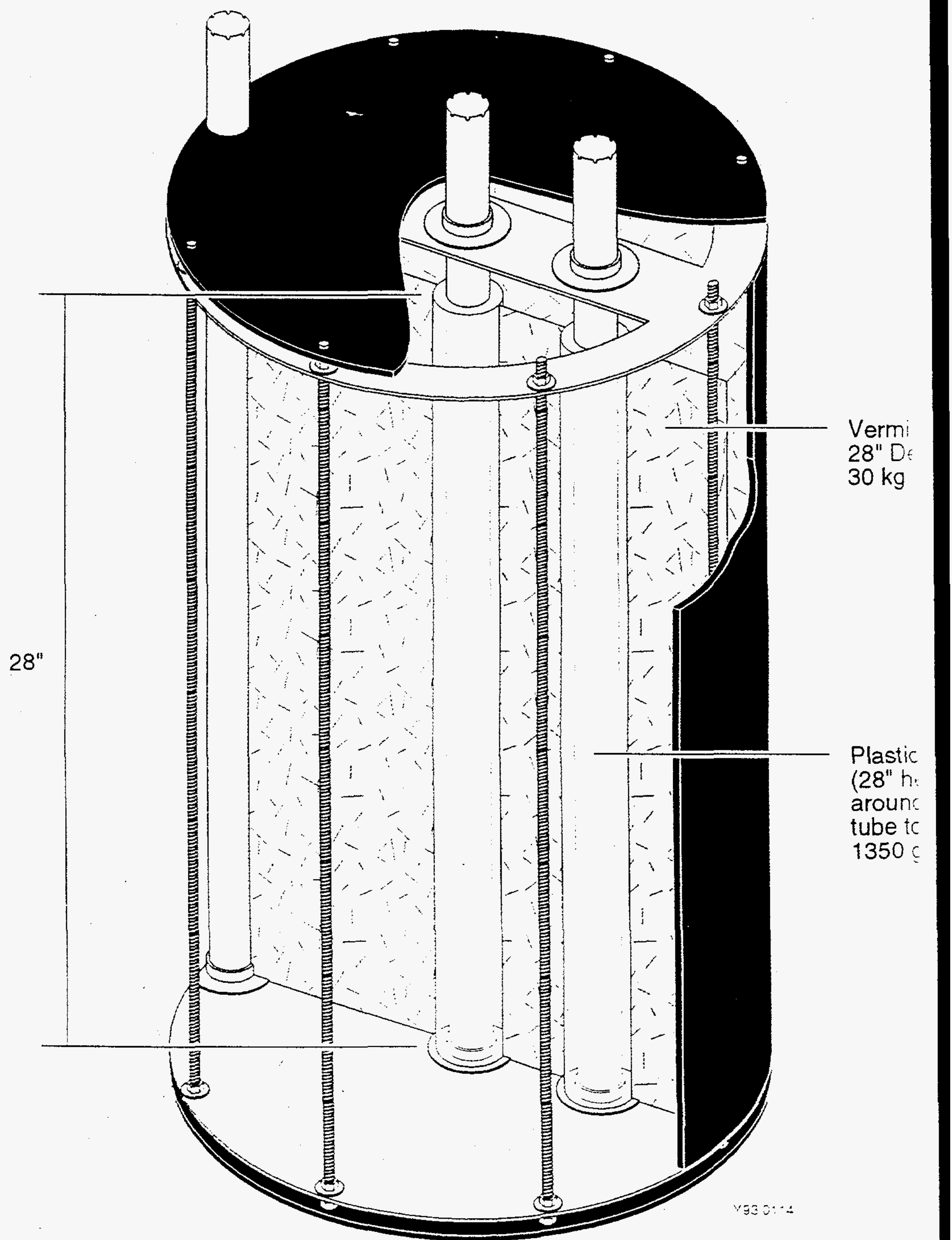


Figure E-7. Combustibles surrogate drum (vermiculite matrix).

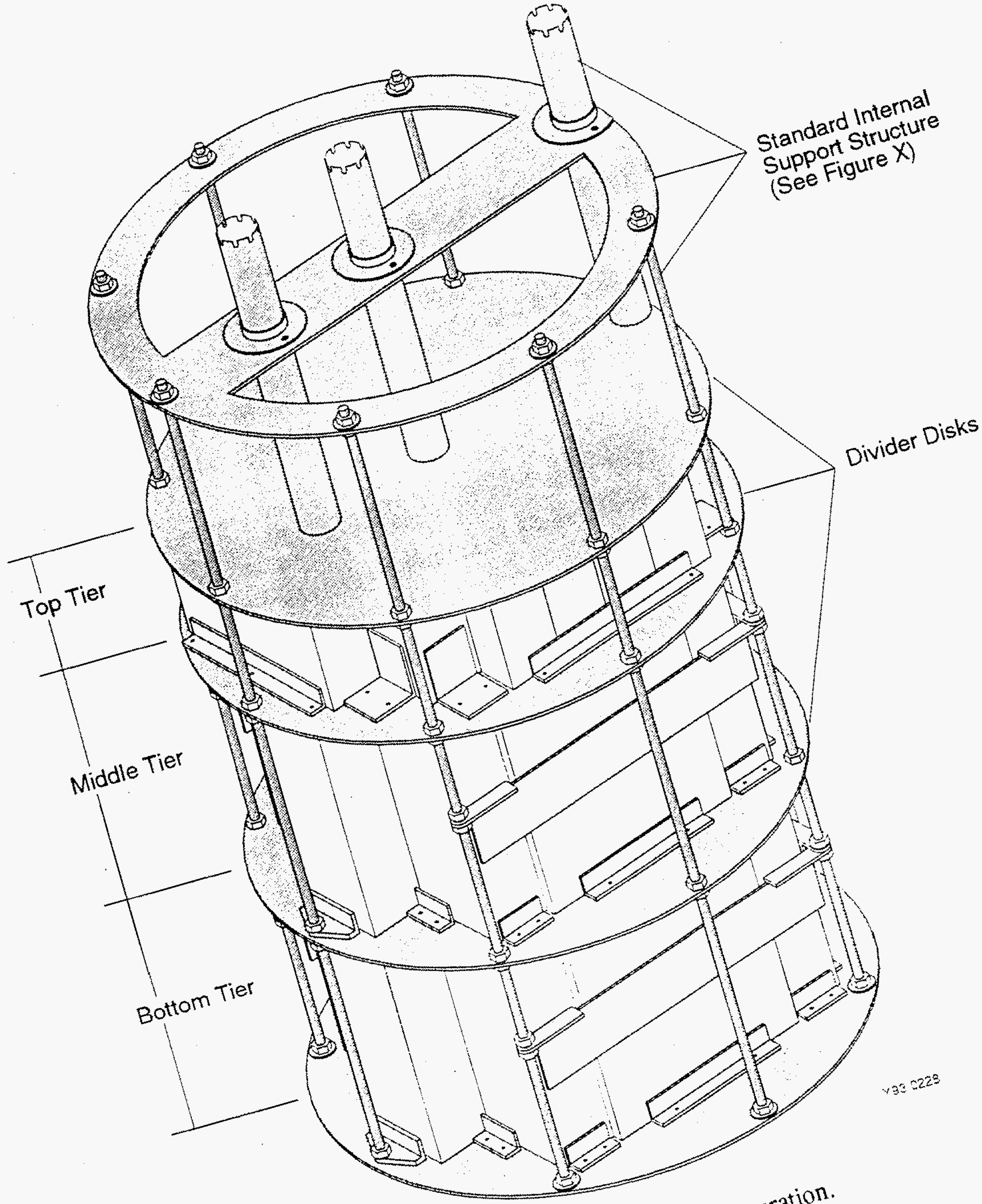


Figure E-8. Fire brick surrogate drum internal configuration.

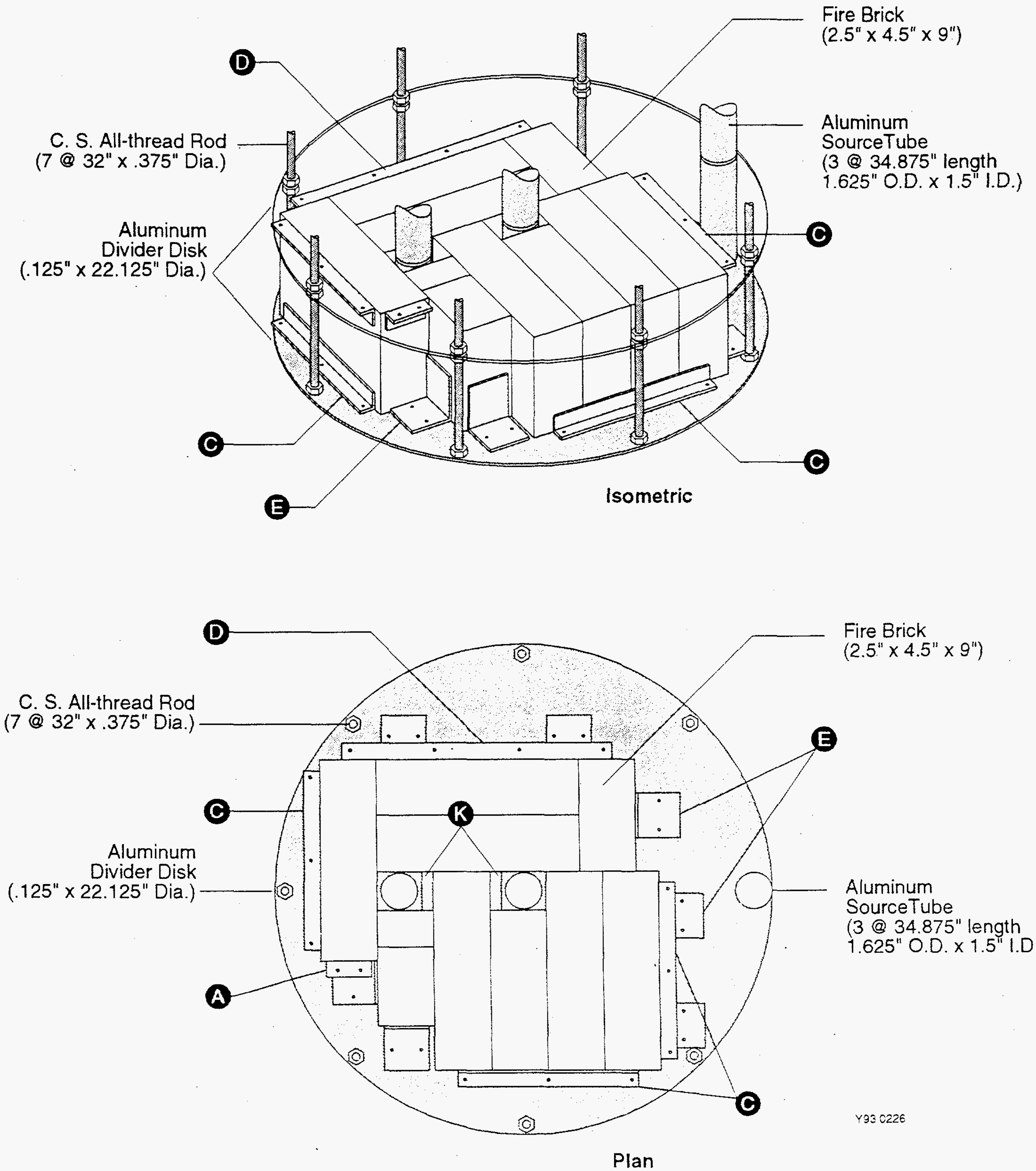


Figure E-9. Fire brick surrogate drum (top tier arrangement).

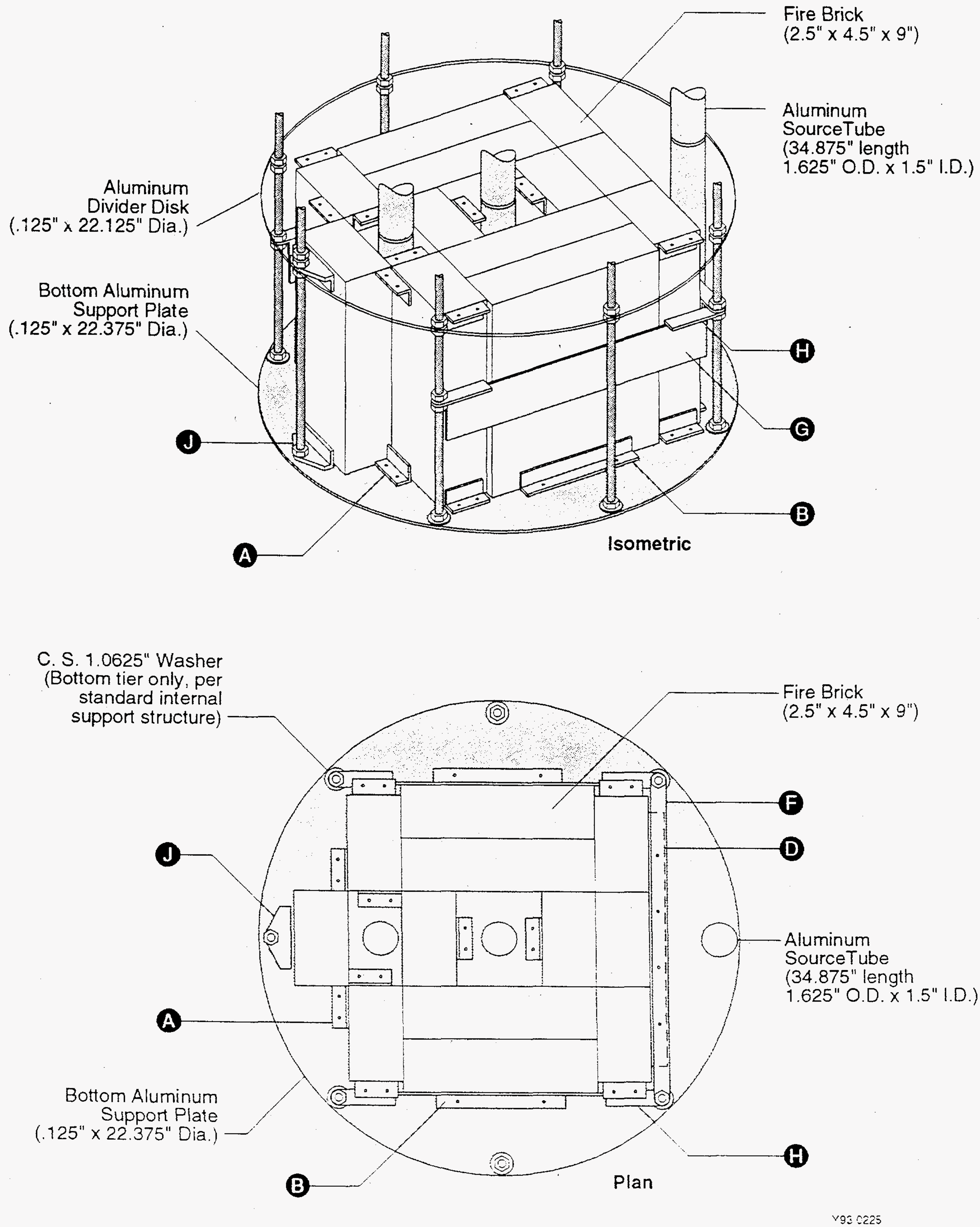


Figure E-10. Fire brick surrogate drum (bottom and middle tier arrangements).

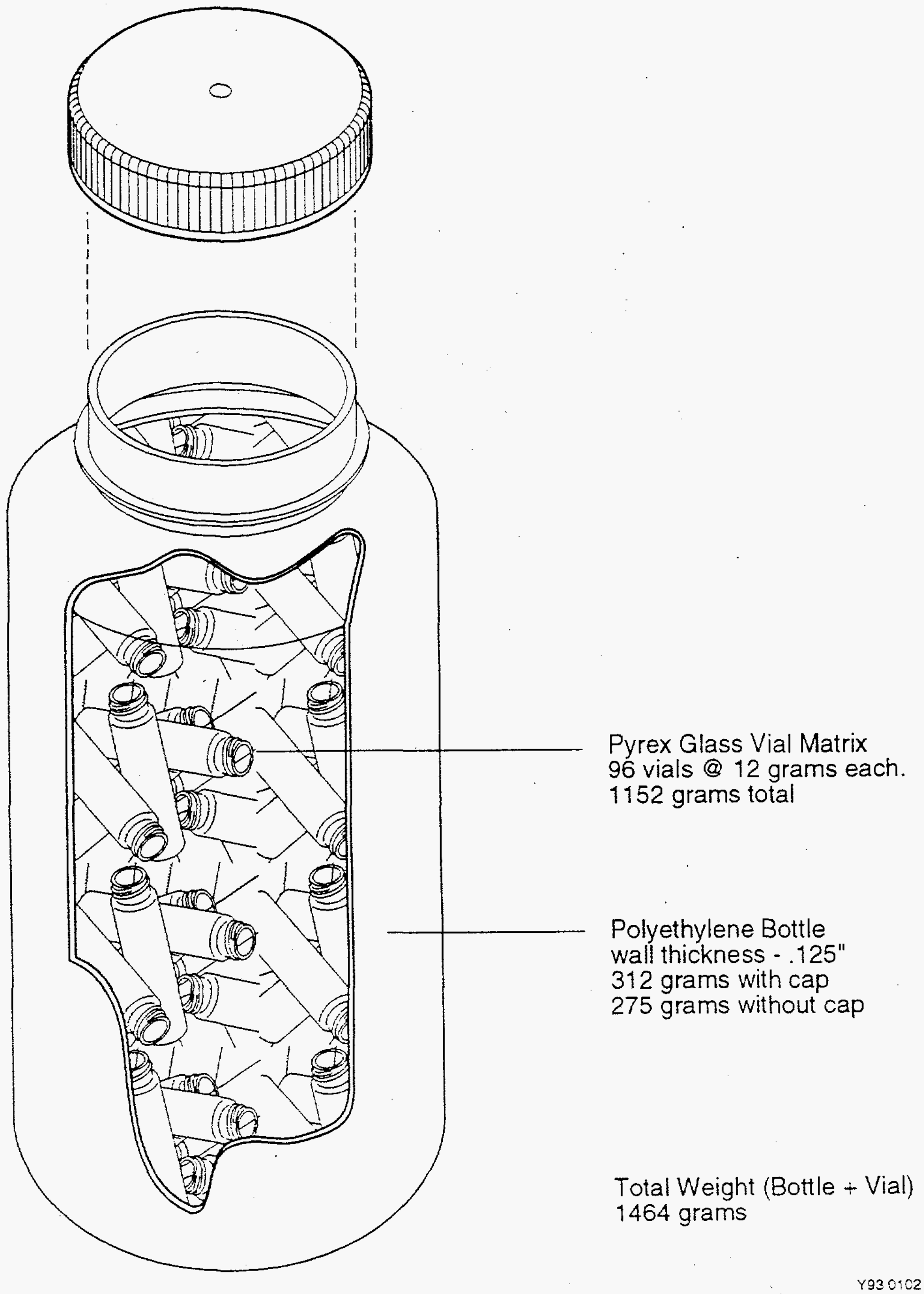


Figure E-11. Polyethylene/pyrex vial matrix (1 gallon).

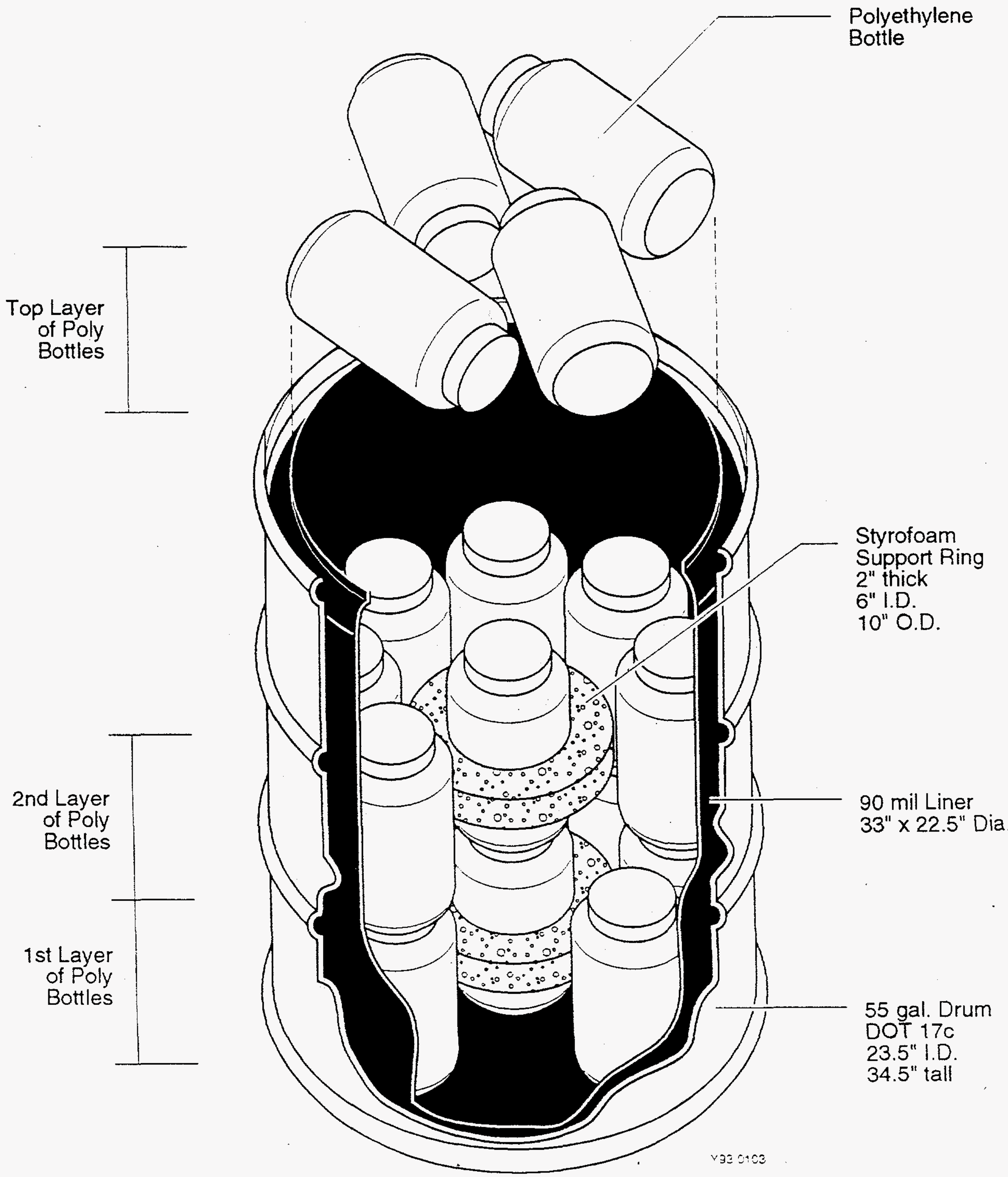


Figure E-12. Polyethylene bottle drum configuration.

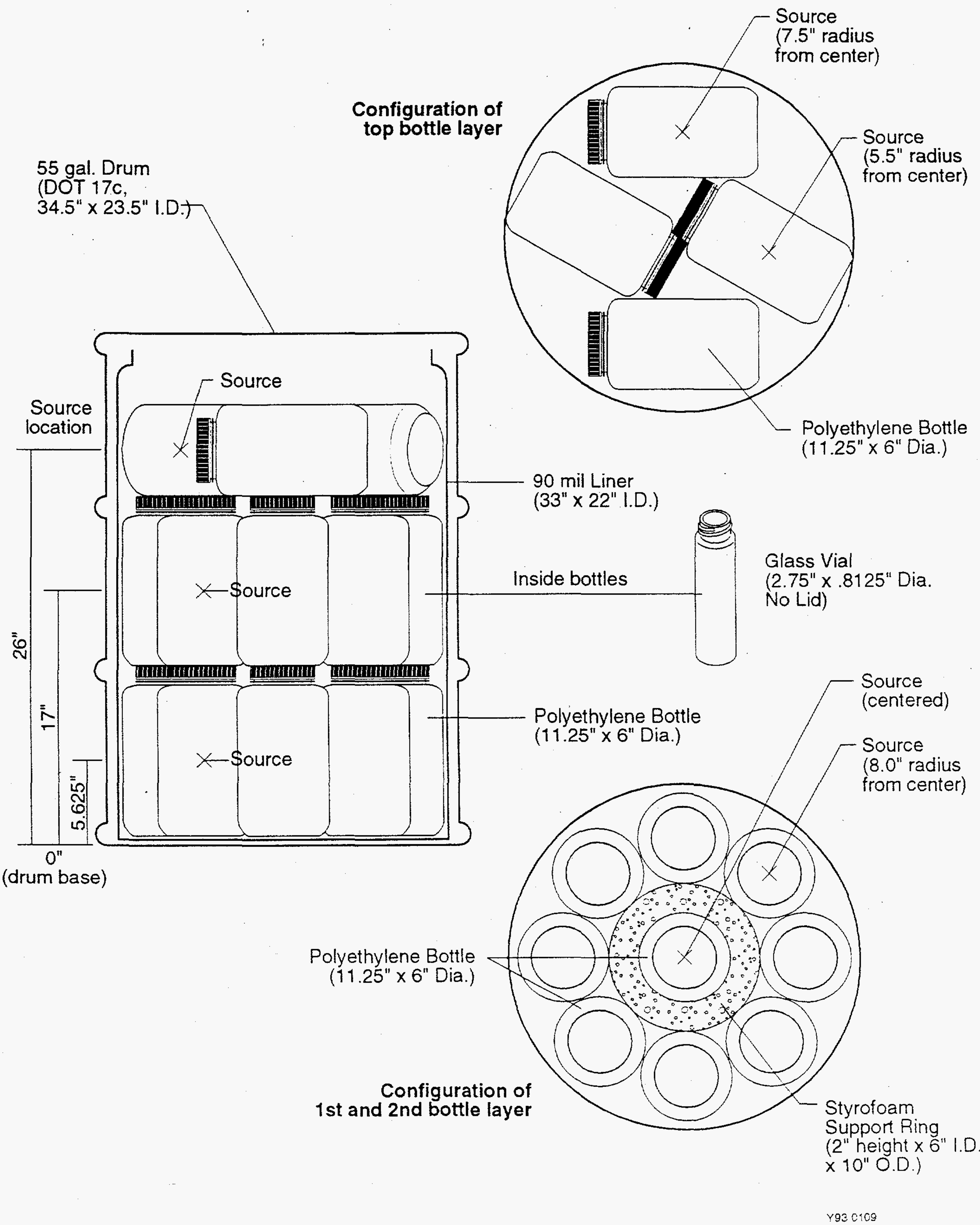


Figure E-13. Polyethylene bottle drum plans and section.

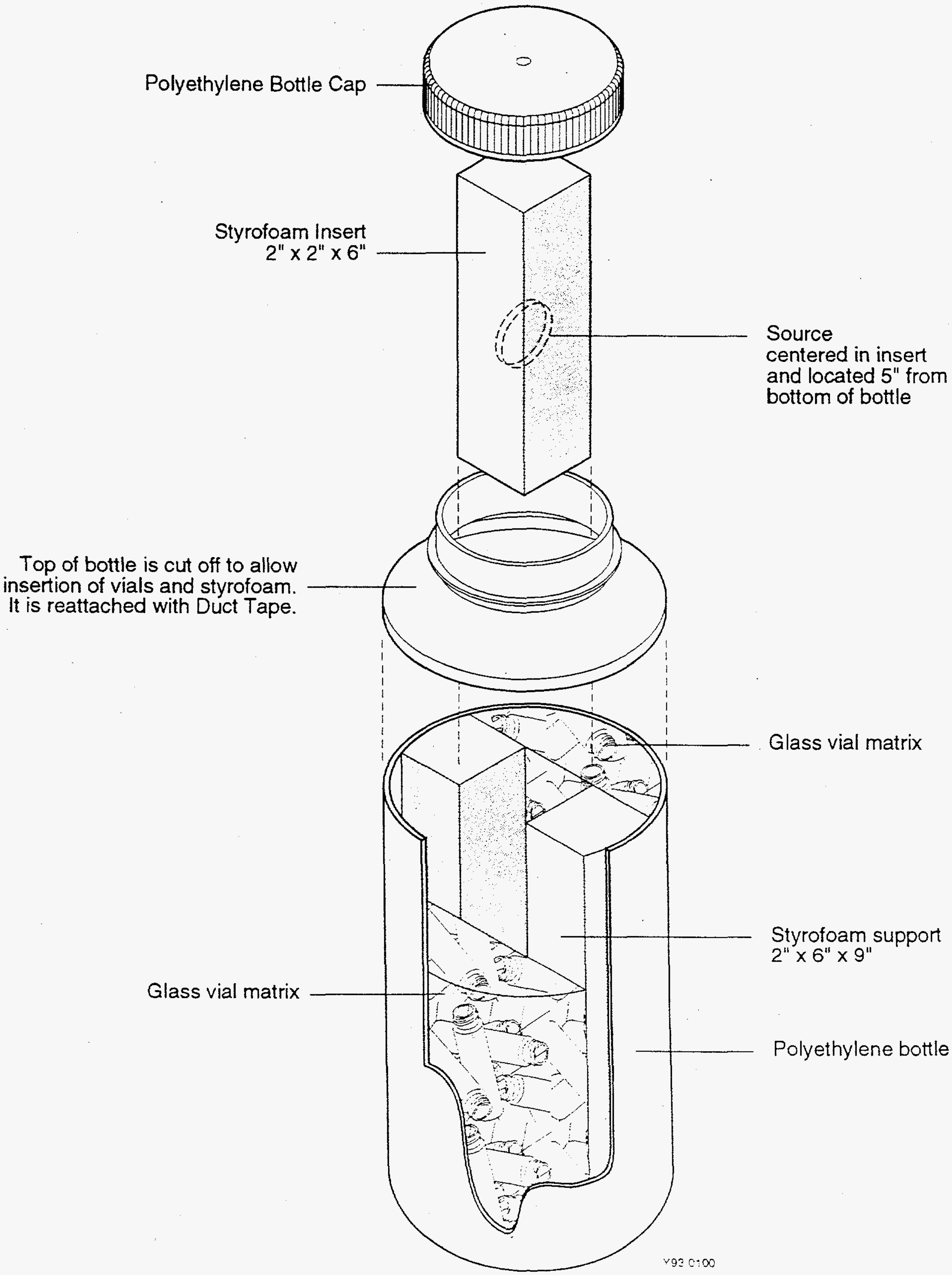


Figure E-14. Polyethylene/pyrex glass source bottle.

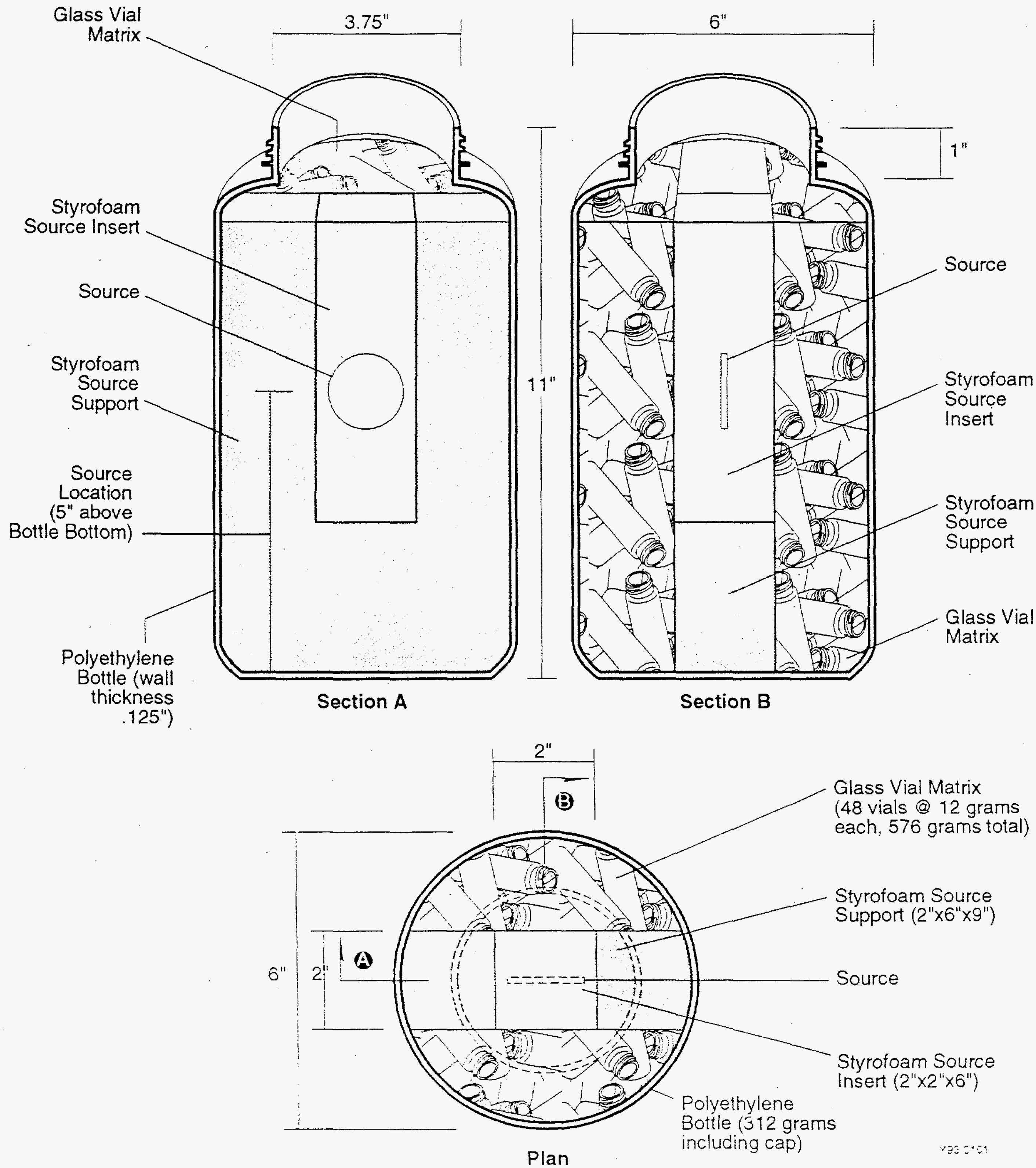


Figure E-15. Polyethylene/pyrex glass source bottle plan and sections.

E-7. SURROGATE FOR CONTENT CODE 442—GLASS (RASCHIG RINGS)

This surrogate was designed to simulate content code 442, leached and unleached Raschig rings. This waste category comprises borated-glass rings used in liquid storage tanks in the plutonium production and recovery areas of Rocky Flats.

This surrogate was constructed by placing two tiers of Fibre Paks (2 Fibre Paks per tier) containing polyethylene bags filled with Raschig rings into a drum (with a 90-mil rigid liner) which contains three vertical source insertion boxes (see Figures E-16 to E-20). The total mass of Raschig rings is 93 lbs. More detailed information can be obtained from Reference E-7.

E-8. SURROGATE FOR CONTENT CODES 480/481—HETEROGENEOUS MIXED METALS

This surrogate was designed to simulate content codes 480 and 480, leached and unleached heterogeneous mixed metals. This category comprises metal waste such as iron, copper, aluminum, stainless steel, etc.

Figures E-21 and E-22 show the internal mixed metals matrix configuration. There are ten discrete metal assemblies or bundles, each comprising a different type and elemental composition typical of mixed metal waste. Each bundle is assembled and fixed into a simple geometrical form such as a parallel piped or cylinder. The compositions and masses of these bundles are shown on Figure E-20. More detailed information can be obtained from Reference E-8.

E-9. SURROGATE FOR CONTENT CODES 480/481—METALS (VALRATH CANS)

This surrogate was designed to simulate content codes 480 and 481, leached and unleached metals. Although these categories comprise both stainless steel and non-stainless steel, this particular surrogate was prepared to simulate only the stainless steel component of these content codes.

The surrogate is comprised of a Type 1 zero matrix drum which has been filled with four tiers of stainless steel valrath cans (14 cans/tier). The total mass of the stainless steel matrix is 58 lbs. Figures E-23 to E-25 show drawings of this surrogate drum. More detailed information concerning this surrogate drum can be found in Reference E-9.

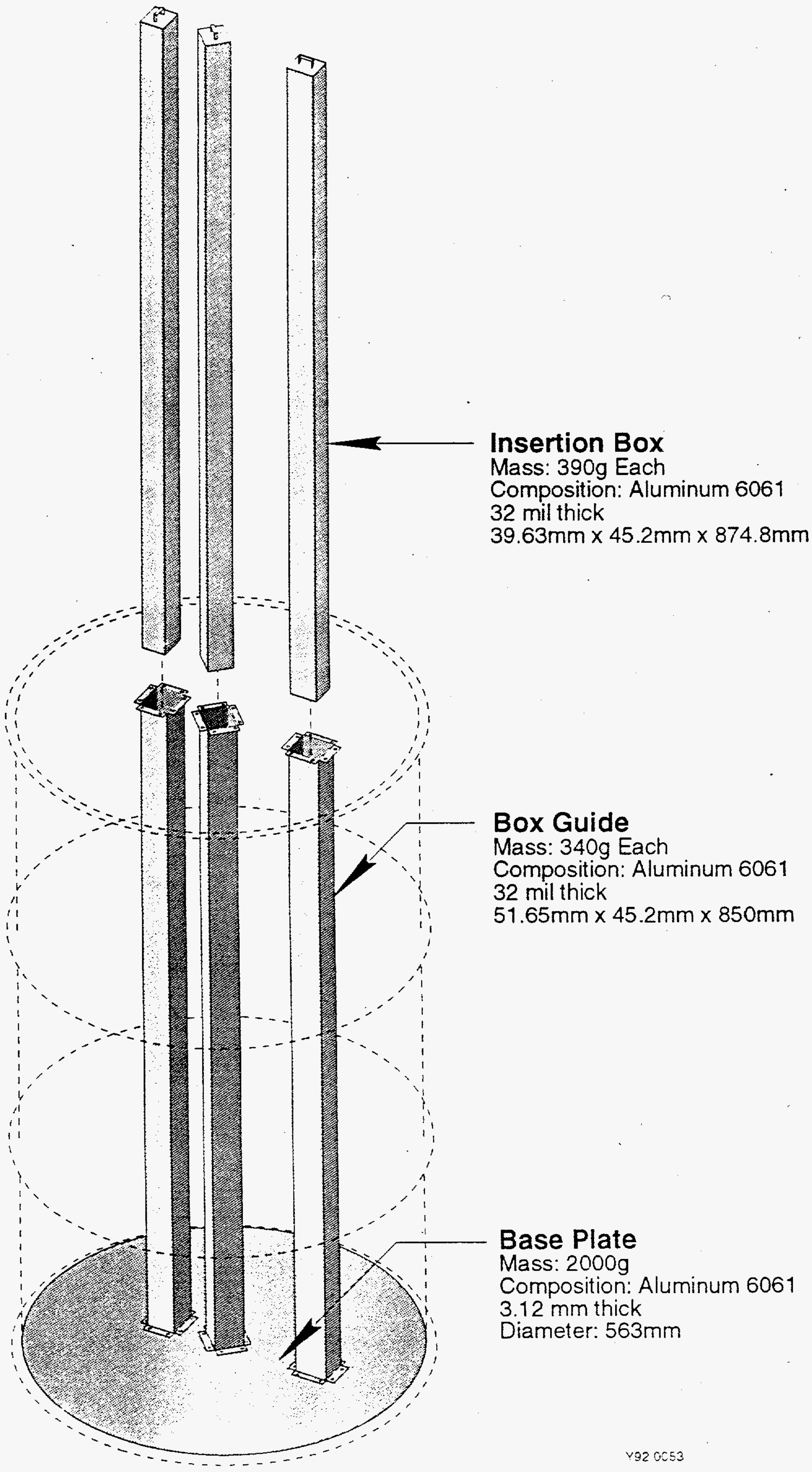


Figure E-16. Aluminum structure.

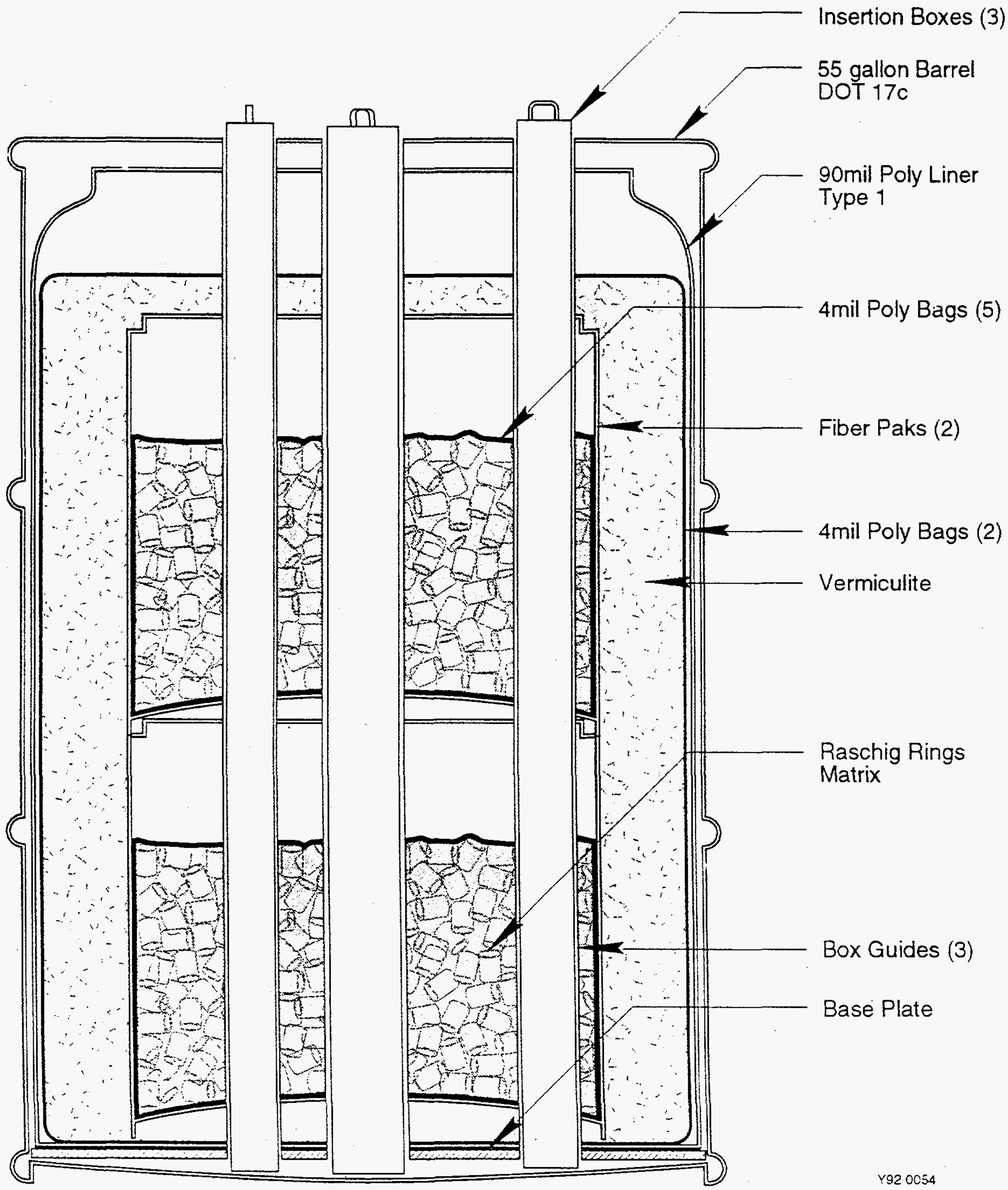


Figure E-17. Barrel assembly.

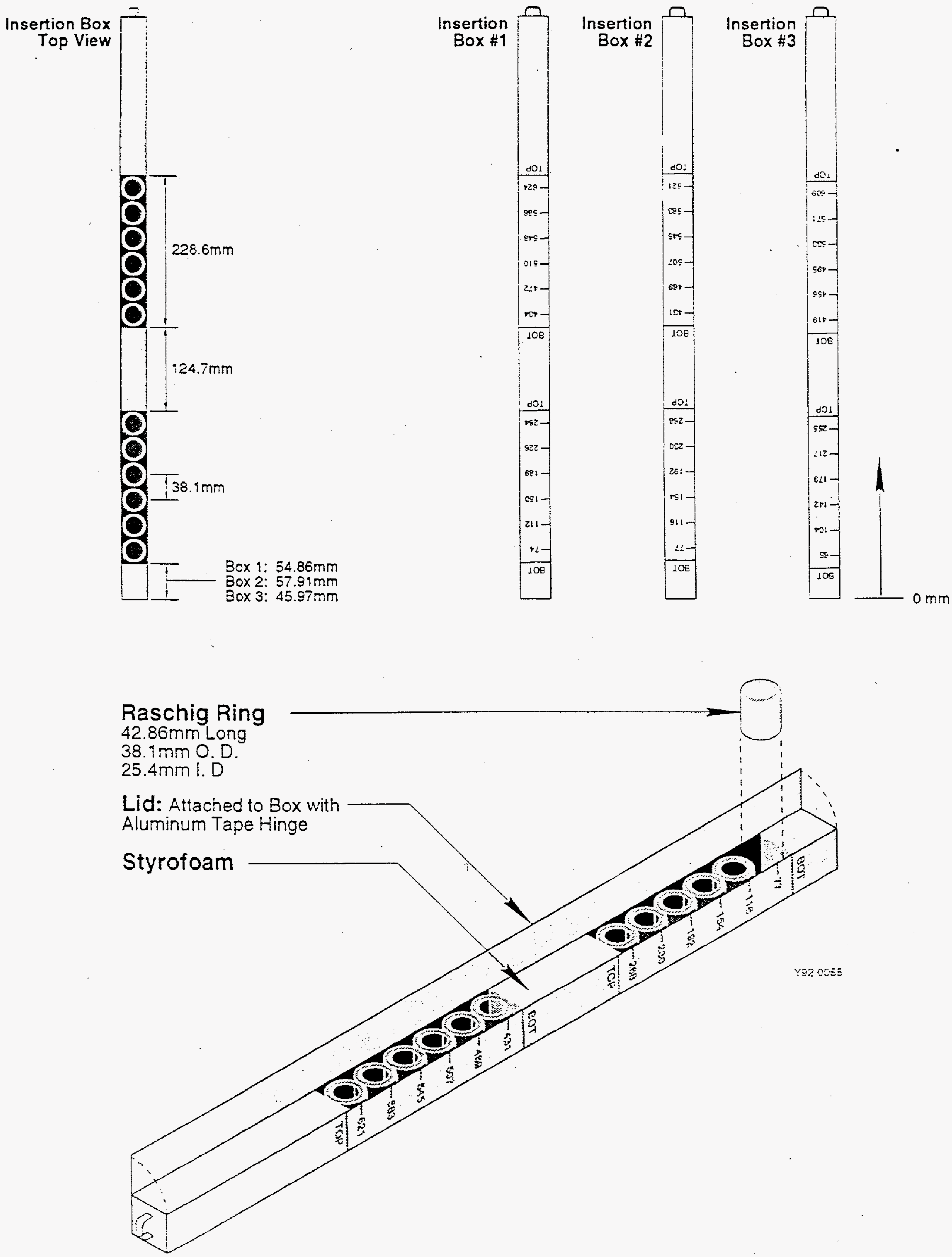


Figure E-18. Insertion boxes and raschig rings.

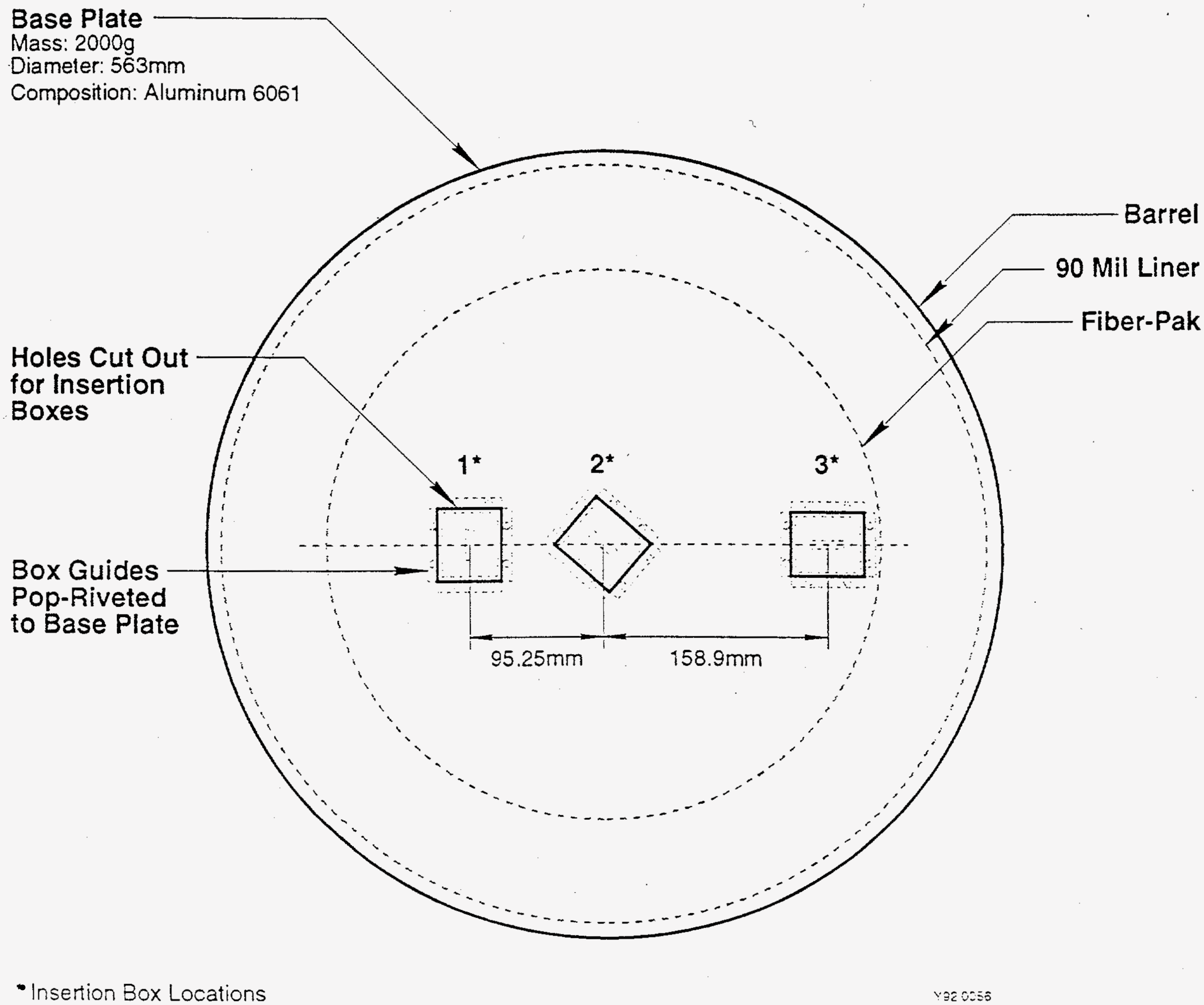


Figure E-19. Base plate.

Mass: Metal Clamp = 319.3g
Metal Lid = 545.3g
Metal Ring = estimated 400g each

Metal = 1664.6g (Lid, 2 Rings, Clamp)
Cellulose = 1000g each

Composition: Metal = Low Carbon Mild Steel
Cellulose

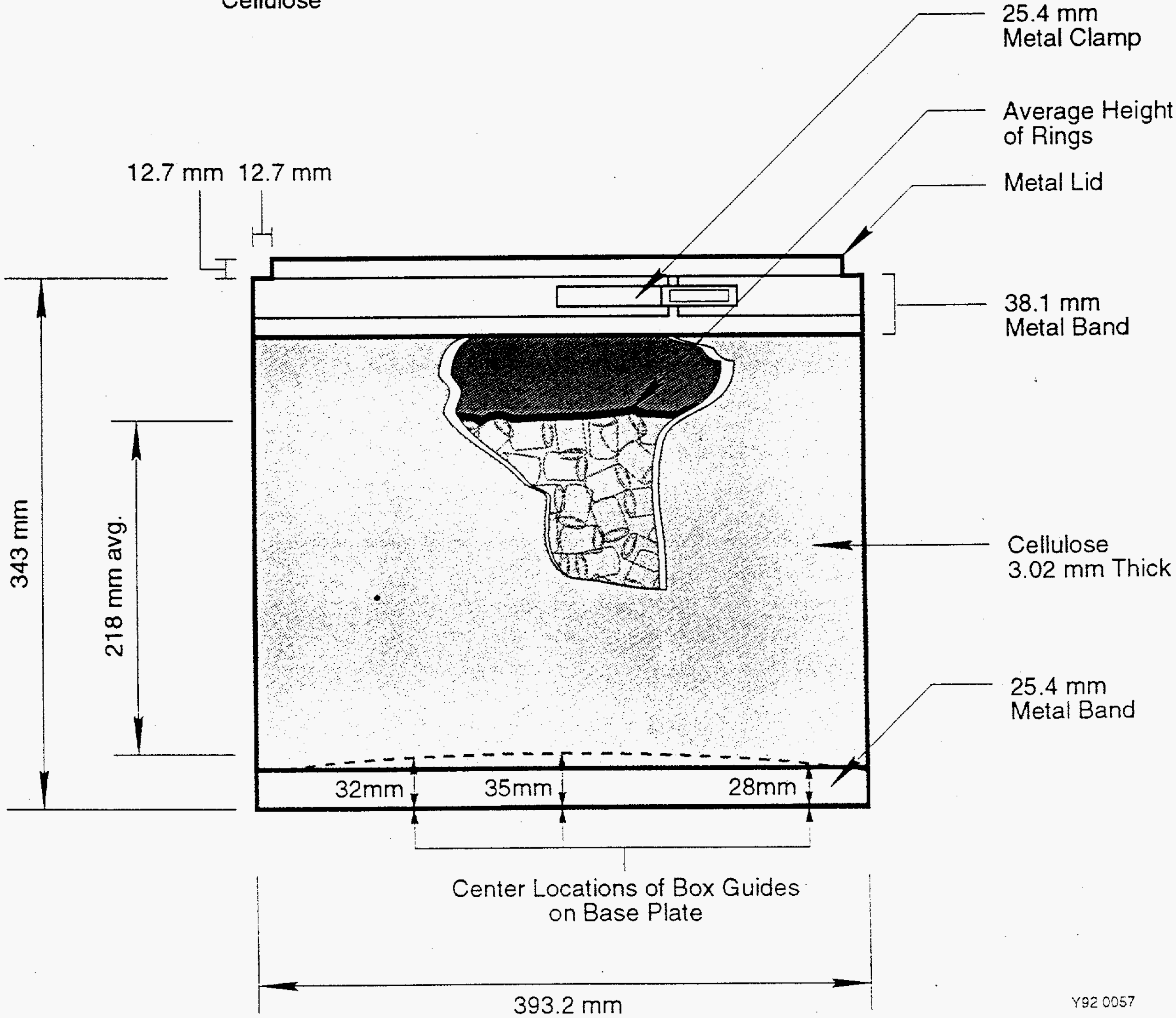


Figure E-20. Fiber-pak.

A. Copper Tubing Bundle

B. Lead Roll

C. Carbon Steel Plate Bundle

D. Carbon Steel Flat Bar Bundle

E. Carbon Steel Pipe Bundle

F. Aluminum Sheet Bundle

G. Stainless Steel Solid Rod Bundle

H. Copper Tubing Bundle

J. Aluminum Sheet Bundle

K. Stainless Steel Angle Iron Bundle

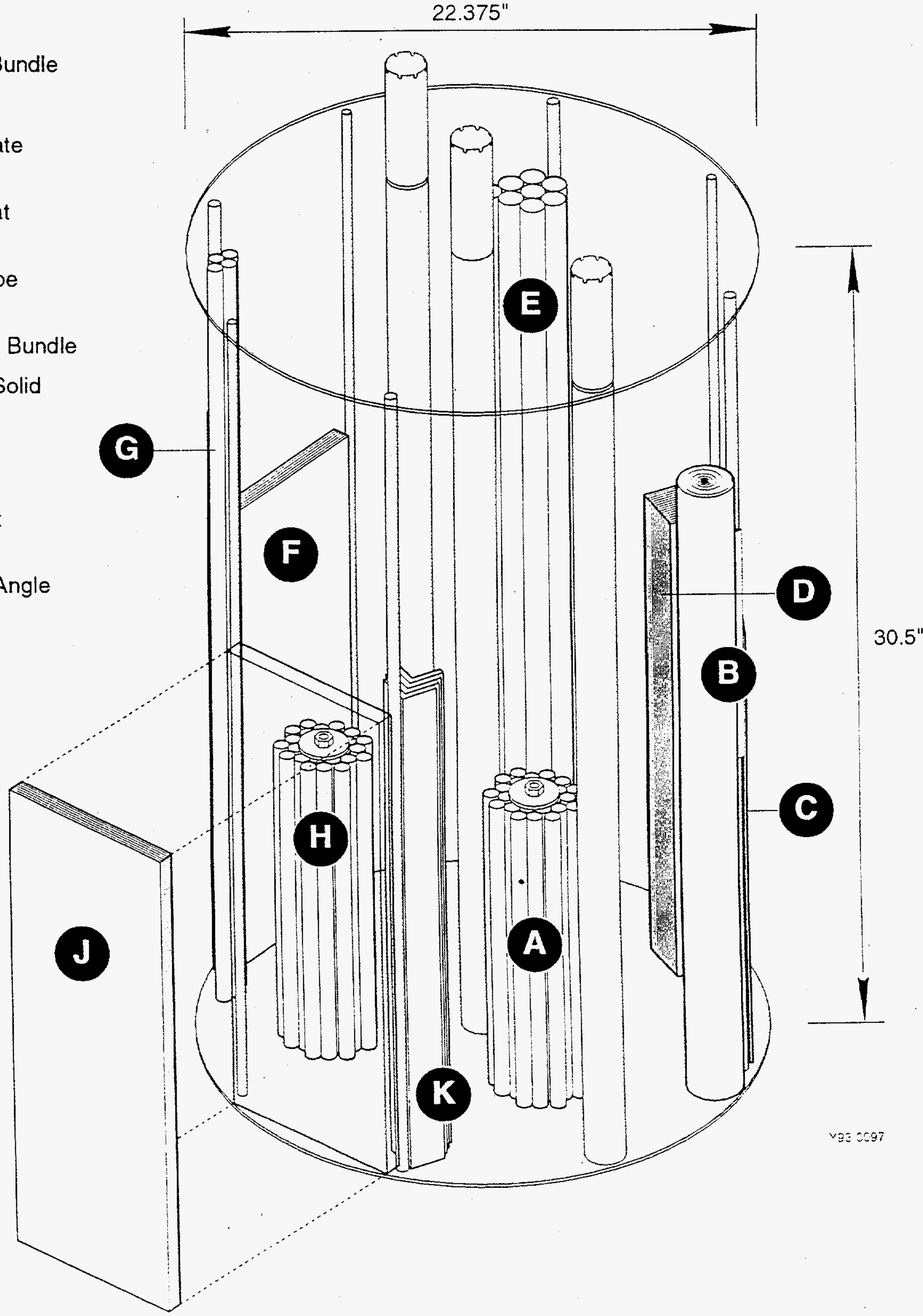
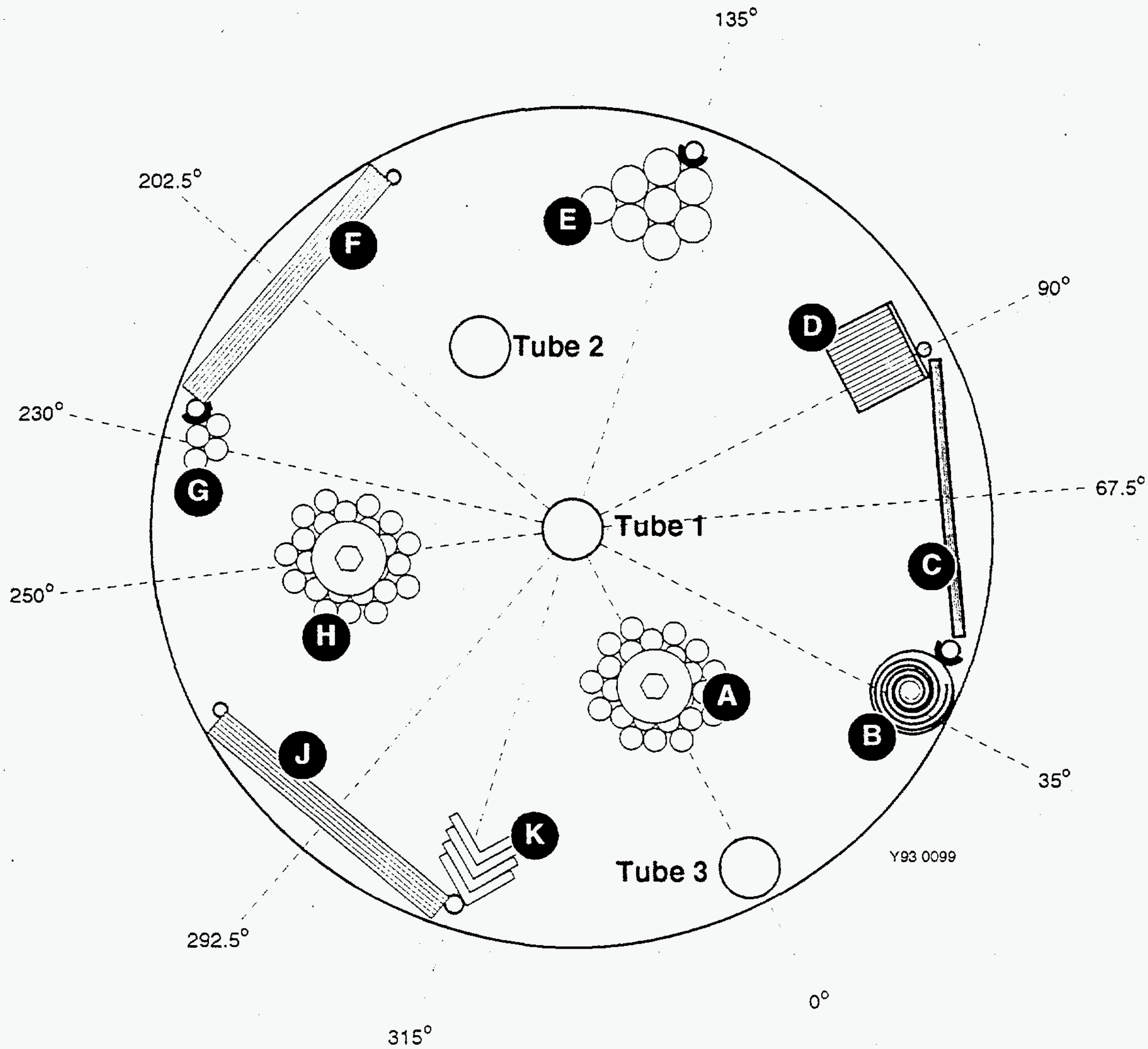


Figure E-21. Mixed metals materials insert.



A. Copper Tubing Bundle - 0° and 6.5" from outside
28 Tubes - .625" x 11.5" - wt. 3.4 kg.
12" circum. x 11.5" high
+ 1 ea. Mounting Bolt - 357.1 g.

B. Lead Roll - 35°
.125" x 2.25" dia. x 24" - wt. 13.4 kg.

C. Carbon Steel Plate Bundle - 67.5°
6 ea. Plates - .045" x 7.5" x 17" - wt. 5.0 kg.
Bundle - .3125" x 7.5" x 17"

D. Carbon Steel Flat Bar Bundle - 90°
16 ea. Bars (welded) .125" x 2" x 18" - wt. 10.0 kg.
Bundle - 2.25" x 2" x 18"

E. Carbon Steel Pipe Bundle - 135°
8 ea. Pipe Sections - 1" x 28" - wt. 13.8 kg.
Bundle - 14" circum. x 28" (welded)

F. Aluminum Sheet Bundle - 202.5°
6 ea. Sheets .125" x 7.75" x 18" - wt. 4.7 kg.
Bundle - .75" x 7.75" x 18"

G. Stainless Steel Solid Rod Bundle - 230°
4ea. Solid Rods .625" x 29" - wt. 4.8 kg.
Bundle - 4.75" circum. x 29"

H. Copper Tubing Bundle - 250° and 3.5" from outside
28 Tubes - .625" x 11.5" - wt. 3.4 kg.
12" circum. x 11.5" high + 1 ea. Mounting Bolt - 357.1 g.

J. Aluminum Sheet Bundle - 292.5°
6 ea. Sheets .125" x 7.75" x 18" - wt. 4.7 kg.
Bundle - .75" x 7.75" x 18"

K. Stainless Steel Angle Iron Bundle - 315°
4 ea. Angle Sections .25" x 1.5" x 18.75" - wt. 6.6 kg.
(welded)

Figure E-22. Mixed metals materials insert plan and material values.

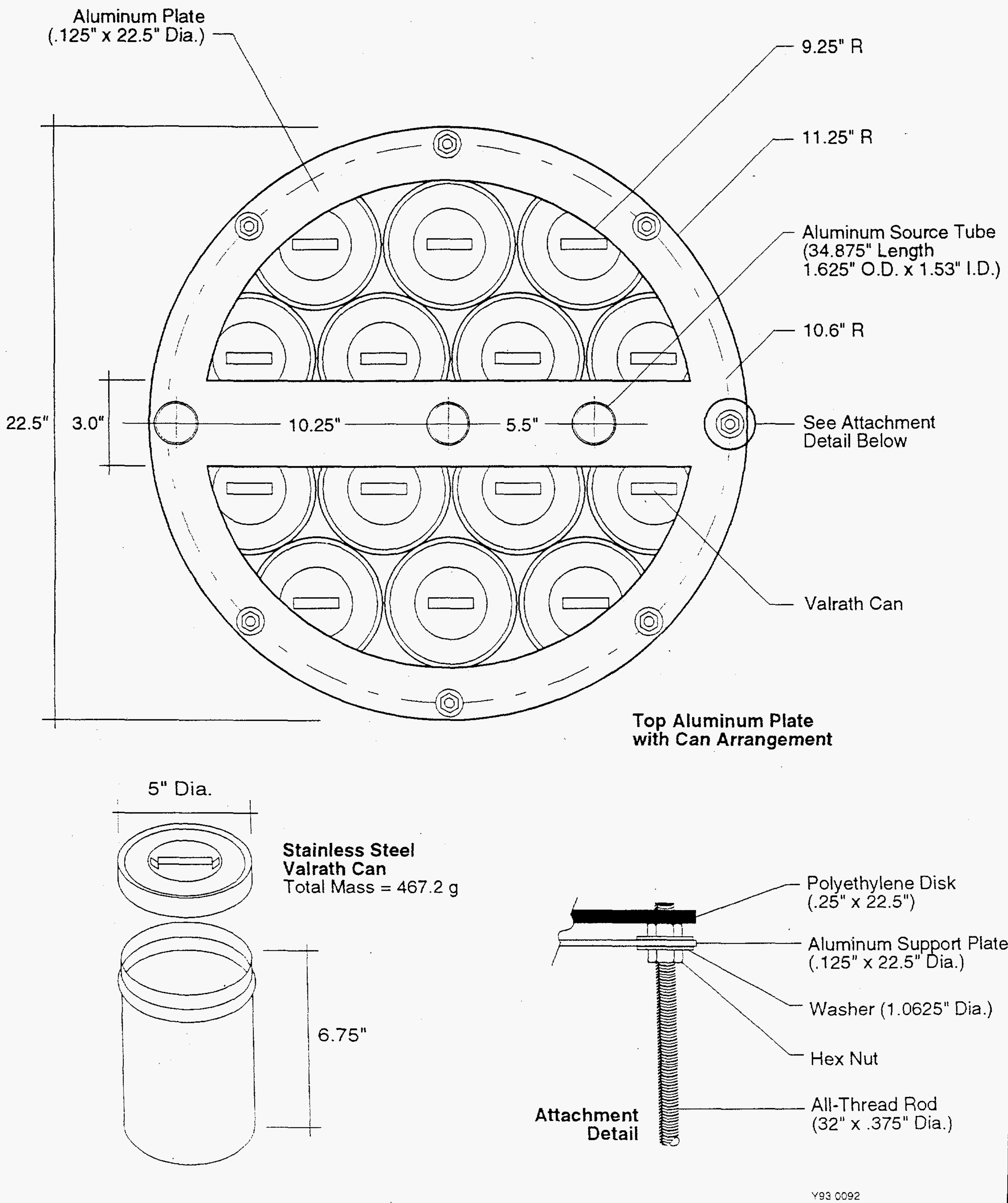


Figure E-23. Valrath can drum plan and details.

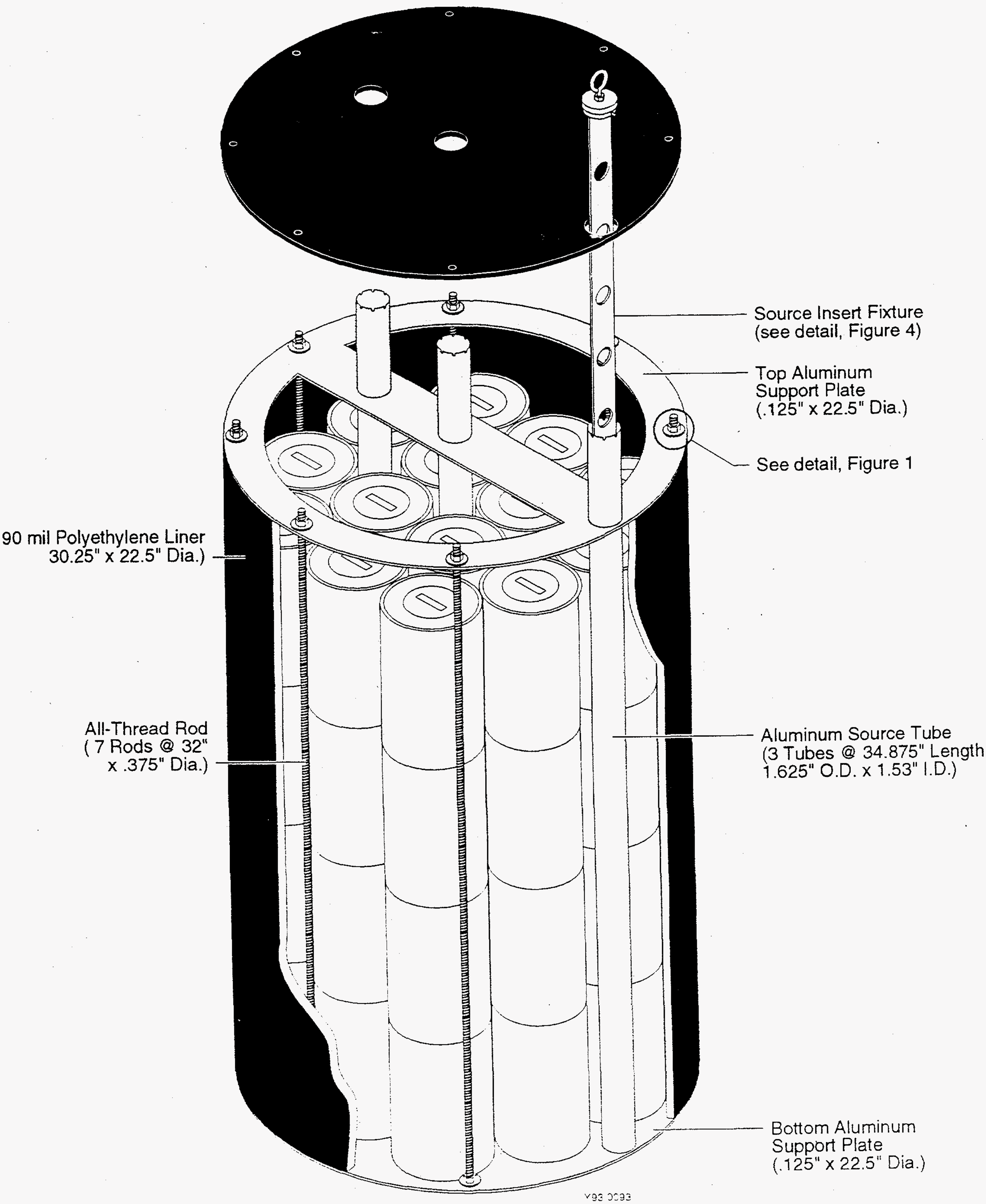


Figure E-24. Valrath can drum matrix configuration.

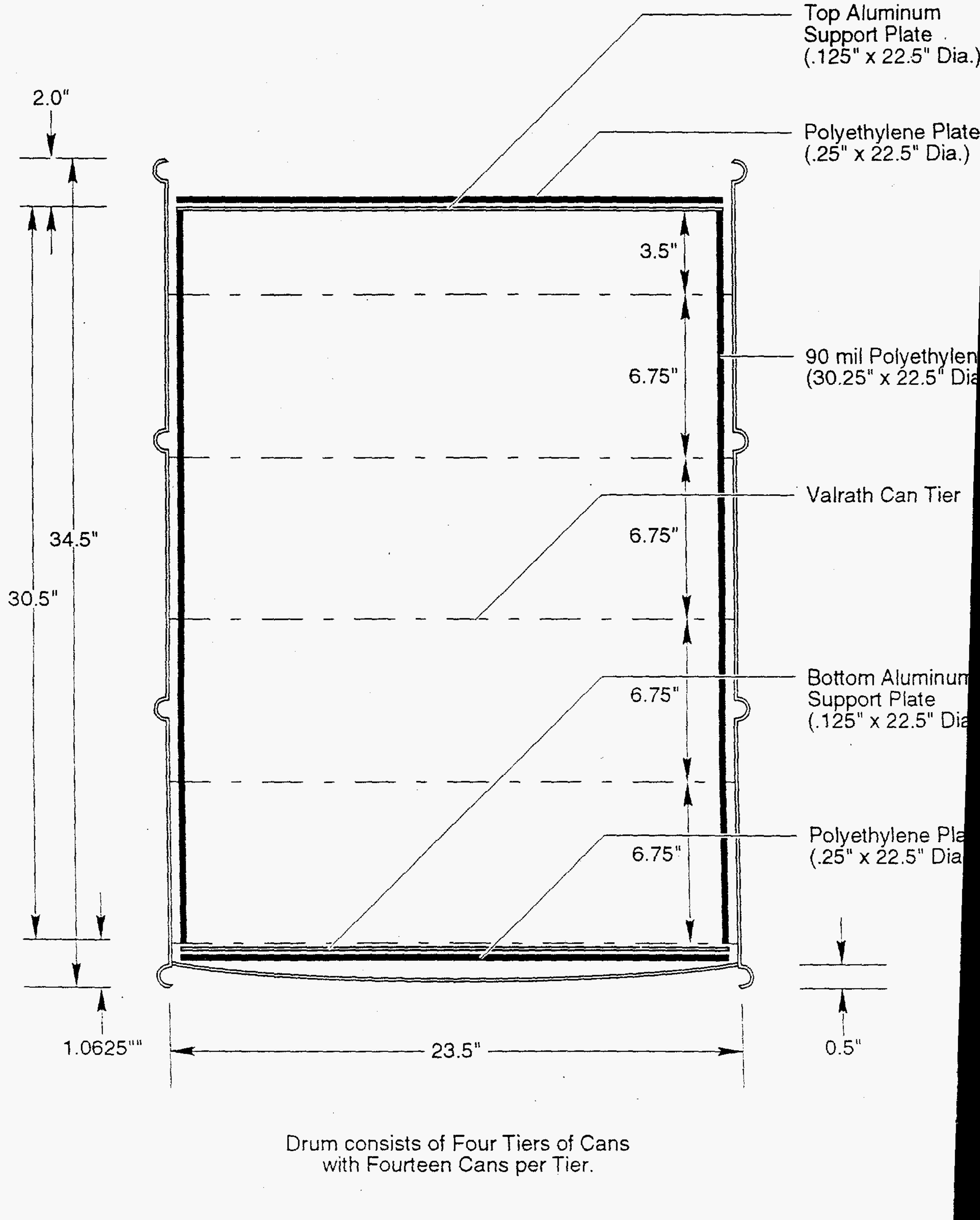


Figure E-25. Valrath can drum section.

E-10. REFERENCES

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