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ORNL - TM - 289

cy. 4

DATE - July 18, 1962

POWER REACTOR FUEL REPROCESSING PILOT PLANT:

PROPOSED ACCOUNTABILITY PROCEDURES

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Classification changed to: Unclassified (level and category)

By authority of: CG-DAR-2 (classification guide)

9/10/99 Date

ADC or ADD signature (first reviewer)

Ted Lewis 9-10-99 Date

ADD signature (final reviewer)

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SUBJECT: Power Reactor Fuel Reprocessing Pilot Plant: Proposed Accountability Procedures

1.0 INTRODUCTION

The assignment of spent reactor fuels from private power reactor operators to Oak Ridge National Laboratory for chemical processing and recovery of fissile and fertile materials requires that the power reactor fuel reprocessing pilot plant have procedures, methods and equipment suitable for strict accountability of Source and Special Nuclear Materials (SSNM). Based on accountability measurements, private power reactor operators will be credited payment for the value of these materials sold to the Atomic Energy Commission. Also, the reprocessing pilot plant will require an accurate accounting program to aid in criticality and process control.

From the standpoint of SSNM accountability, the recovery of valuable material begins with discharge of fuel from the reactor and ends with return of fuel to the reactor (Fig. 1). The pilot plant, however, is concerned with accountability of SSNM materials received from the reactor site. This discussion of SSNM accountability in the PRFR pilot plant is concerned principally with: (1) recent developments made in improving accountability measurements of pilot plant input; and (2) how these developments will be applied to future plant practice. Specifically, this discussion includes measuring-vessel design, instrumentation, analytical procedures, and a suggested accountability program. Arguments pro and con are included for taking measured plant input as a basis for contract payment.

The spent reactor fuels assigned to ORNL will in general be clad with stainless steel, except for the Foreign Research Reactor fuel which will be made from an aluminum alloy. After discharge and a suitable decay period these fuel assemblies may be reduced to sizes suitable for shipment by shearing, sawing or other machining operations, producing variable quantities of debris in the form of sawdust and fragments. Ruptured elements will also contribute

to the debris. Similarly, when the bulk material is received at the pilot plant, further storage and disassembly will probably result in additional debris. Unless this debris is accounted for by both the shipper and receiver, depending on where the debris is generated, there may be a considerable disagreement between shipper and receiver estimates of SNM contents in the fuels.

The physical combination and chemical composition of the fuels will vary with the reactor. Owing to the variety of fuels expected, a number of head-end processes have been developed to convert the fuels into aqueous-nitrate solutions for solvent extraction processing and subsequent recovery. The head-end processes being developed are of two types: (1) a two-step process in which the clad is removed first by either mechanical or chemical means followed by an independent dissolution of the fertile-fissile core material; (2) a one-step co-dissolution of the cladding and core in aqua regia with subsequent dechlorination. The aqueous solutions from either process will have to be clarified to remove solids which will interfere with solvent extraction operations. It is expected that the precision of SNM measurement in dissolver solutions will be affected considerably by the composition of the fuels, the dissolving media and the presence of solids.

The PRFR pilot plant has begun a study of these problems concerning an accountability program with a goal of devising an accountability program acceptable to receiver and shipper alike.

2.0 RECENT DEVELOPMENTS IN IMPROVING ACCOUNTABILITY MEASUREMENTS

Prior to startup of the Power Reactor Fuel Reprocessing Pilot Plant facilities in Buildings 3505 and 3019, a program was begun to obtain sufficient information to make a statistical analysis of feed input measurements. Until actual power reactor fuels are processed in 1962-3, conventional, aluminum-clad, natural uranium fuels are being processed for recovery of uranium and plutonium. These fuels serve to check out the processing equipment. The information obtained during this period will furnish a basis for design of process vessels to serve as accountability measurement vessels in the head-end facilities to be constructed

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and will be helpful in developing plant procedures and methods for accountability of power reactor fuels.

In the present process facilities housed in two separate buildings which are connected by underground piping for inter-transfer of process solutions, fuel may be dissolved in either facility (Fig. 2). When the 3505 Building dissolver is used, the metal is dissolved continuously. The dissolved metal solution is pumped continuously to the 3019 Building dissolver where it is collected and jetted batchwise to a feed measurement vessel (S-2 tank). When the 3019 Building dissolver is used for dissolution of fuel metal, the dissolvings are made batchwise and jetted to S-2 tank. Feed solution is prepared, processed through one cycle of solvent extraction in 3019 Building and transferred to the 3505 Building for further solvent extraction processing and product recovery.

Since all dissolved metal solution must pass through S-2 tank in the 3019 Building facility before being pumped to solvent extraction, this tank was selected as the feed input measurement vessel. This is a stainless-steel, 550-gal jacketed vessel having a reverse dish bottom and a standard dish top.

2.1 Preliminary Work

Before plant startup operations were begun, new instrumentation was installed for S-2 tank, the vessel was calibrated, and solution mixing and sampling tests were made in preparation for accountability studies.

2.1.1 S-2 Instrumentation

The instrumentation for measuring and recording liquid level and specific gravity of feed solution in S-2 is shown in Fig. 3. This is a 550-gal vessel with a reverse dish bottom and a straight dish top. The capacity of the vessel is about 1600 liters to the top of the straight-wall section.

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Four air-purged probes extend into the vessel, two for specific gravity and two for liquid level measurement. The ends of the specific gravity probes are separated vertically 10.2 inches at the bottom of the vessel. One liquid level probe extends to the bottom of the vessel while the low pressure probe is just inside the tank top. Purge air is metered at 0.2 scfh through air rotameters and introduced into the probes just above the tank top. The vessel is also equipped with a thermocouple and panelboard-mounted recorder for temperature measurement. All volume measurements are corrected to 20°C.

Pressure differential across each set of probes is sensed by dual transmitters, Foxboro Model 13A. Two panelboard-mounted Foxboro M-53 miniature recorders receive the output signals from the transmitters. Each recorder receives liquid level and specific gravity measurement.

In the event the readings on the two recorders vary more than one chart division, a mercury-filled manometer mounted behind the panelboard can be plugged into the rear of each recorder to measure the output signal from each transmitter. Also, two manometers mounted adjacent to the transmitters can be plugged into the transmitter input to take a direct measurement across either the liquid level or specific gravity probes. These manometers allow comparison of the pressure differential measured across the probes with the recorder values to determine which recorder is in error.

2.1.2 Calibration

The feed-input vessel was calibrated with water and nonradioactive uranyl nitrate solution of approximate feed concentration before the Power Reactor Fuel Reprocessing Program was begun. This was done by adding weighed amounts of these fluids to the vessel and recording liquid level and specific gravity after each increment. After the vessel was filled, the fluid was then withdrawn in weighed increments.

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These calibration runs resulted in two calibration curves, one for filling the tank and one for emptying the tank (Fig. 4). The coefficients for the curves were determined by a least squares fit. The curves were computed from 46 calibration points, 20 with water and 26 with uranyl nitrate points. At the top of the curves, at a true liquid level of 40%, both curves give essentially the same volume. The filling curve gives 1663 liters while the emptying curve gives 1660 liters. However, at 4% liquid level, the difference in volume is 18.56 liters (150.32 for filling and 131.76 for emptying). When solution is transferred to S-2, the filling curve is used. When solution is removed from S-2, the emptying curve is used.

More scatter was obtained in the data taken when calibrating by emptying the vessel than when filling the vessel. The limits of error curves for both calibration curves (Fig. 5) show almost twice as much error for emptying the vessel. At 400 liters the limit of error is 0.7% for filling and 1.3% for emptying, while at 1400 liters the limit of error is 0.2% for filling and 0.3% for emptying. The limit of error for a typical transfer of 1000 liters is 0.5%.

The operating procedure for feed measurement is to fill S-2 to about 1400 liters volume. The solution is air sparged for two hours, sampled and the data are recorded. About 1000 liters of solution is removed from the vessel, leaving a measureable heel of about 400 liters which is combined with the next feed batch.

2.1.3 Solution Mixing

Since the solution in the feed-input vessel is mixed by air sparging, a series of tests was made to establish the length of time required to insure a homogeneous solution. To test under more severe conditions than ordinarily expected during feed preparation, we added a uranyl nitrate solution of two different concentrations to the feed-input vessel and added a top layer of water. Approximately 900 liters of uranyl nitrate solution containing about 200 grams uranium per liter was added to the vessel. To this, 250 liters of

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uranyl nitrate solution containing about 360 grams uranium per liter was added. Finally, 240 liters of water was added.

The solution was sparged for 15-minute intervals and sampled after each sparge period. After 45 minutesof sparging, one-fourth of the solution was removed from the vessel and the remaining three-fourths of solution was sparged and sampled as described for the full tank. These tests were repeated until only one-fourth of the original volume of solution remained in the vessel. The tests were conducted this way because we felt that if stratification existed in the vessel, a new surface would be presented to the sampler and sparger at the bottom of the tank each time solution was removed from the tank. Therefore, if stratification or inhomogeneity existed, they could be detected by changes in specific gravity of the samples.

The specific gravity of the samples was measured in the analytical laboratory with a Westphal balance. The specific gravities varied from 1.258 to 1.260 (corresponding to 1.5 grams U/liter) with half of the values being 1.259. It is apparent that little, if any, stratification or inhomogeneity existed. During feed preparation each batch of feed solution is sparged two hours before sampling.

Also, samples were taken before and after transfer of solution from S-2 to continue to look for inhomogeneity. This will be discussed later.

2.1.4 Sampling

Solution is sampled from S-2 tank by air lifting the solution to a sample bottle in a sample blister outside the cell. The solution passes through the bottle and returns to the vessel. About 5 ml of sample is retained in the bottle. A test was made to determine the sampler recirculation time required to produce samples of uniform composition.

The sampler was recirculated for 5-minute intervals for a total time of 30 minutes. At the end of each period the bottle was removed and a

second bottle was placed on the sampler tips long enough to obtain 5 ml of solution.

The samples were analyzed for uranium concentration by the ammonium thiocyanate colorimetric method. The range of the analyses was from 304 to 300 grams uranium per liter over the 30-minute recirculation period, indicating no significant effects of recirculation period vs. concentration. Since there is sufficient time during feed preparation, the sampler is recirculated 20 minutes for each set of samples taken. The recirculation sample bottle is discarded to eliminate the chance of submitting a sample containing either dilute or concentrated solution flushed from the sampler supply line.

It is emphasized that the range of 304 to 300 grams/liter of uranium was not obtained with run-of-the-mill samples, but rather with clear, clean nonradioactive solutions permitting bench-top analysis.

2.2 Studies with Reactor Fuels

At the present time the process facility is in its sixth reactor fuel campaign: SCRUP, NRX booster rod, SRP-E, SNAP and BNL-1,-2. One hundred batches of feed solution were prepared for processing in these six campaigns.

Samples from the feed-input vessel (S-2) were analyzed by the analytical control unit in Building 3019 by conventional analytical procedures; uranium by the colorimetric thiocyanate method and plutonium by a TTA extraction procedure. When the samples emitted less than 1 roentgen/hr, bench-top methods were used, whereas for higher radiation levels, analysis was made by remote manipulators in a hot cell.

Both to decrease the radiation level and to reduce the concentration of uranium and plutonium to the range of the analytical procedures, a dilution was prepared for each sample. Then the uranium and plutonium content of each dilution was determined twice. These two results for each constituent were averaged and reported as a single analysis; one value for uranium, and another for plutonium.

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The initial plans for sampling and for analysis were changed as the program developed to yield information pertinent to the reproducibility, or precision, of the analytical work. However, the desire for this information had to be counterbalanced with cost of analyses, radiation exposure to personnel and sample load in the laboratory. These sampling and analysis plans were designed to yield the following statistical information for the uranium and plutonium analyses:

1. Sample variance
2. Difference between analyses made by the control unit at different times on the same sample
3. Difference between analyses made on the same sample by two different ORNL analytical units
4. Difference between analyses of the samples taken before and after transfer of feed solution from feed-input vessel (S-2).

This information enters into the calculation of the limits of error for the feed-input measurement.

The following is a brief discussion of each item. Initially four samples of feed solution were taken prior to transfer of the feed solution from S-2 tank to the process. To eliminate real "fliers" from consideration, it was required that the range of the four uranium analyses be less than or equal to 7.25% of the average of the four analyses. If the analyses satisfied this requirement they were accepted; if not, four more samples were taken for analysis. Out of 28 sets of four samples taken from 21 batches of feed only two sets were rejected for exceeding the range criterion without having a definite operational reason for rejecting the samples. From the 21 sets accepted, it was established from these tests that the average range for the uranium samples is about 2% and the limit of error is about 2%. Similarly, for plutonium the average range is 3% and the limit of error is 3%.

Six sets of four samples each were re-submitted to the control unit for re-analysis of uranium and plutonium. From these data it was evident that the precision was good; however, there was a difference between means for the first analysis and the second analysis. In other words the range for four samples was

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a satisfactory percentage of the average, but the averages were shifting up and down from the first to the second time that the samples were analyzed. This variability was larger than the sample variance.

Six sets of four samples each were submitted to other analytical units at ORNL to determine the bias between laboratories for uranium and plutonium analyses. At the present time there is insufficient information to determine if a significant difference exists. For uranium the average difference between laboratories was 2.7 ± 6 grams/liter and for plutonium was 0.0006 ± 0.0012 grams/liter.

It was previously mentioned (Sect. 2.1.3) that inhomogeneity of solution in the feed-input vessel would be studied further. This was done by sampling before and after transfer of solution from S-2 vessel as follows:

1. Before transfer of feed solution two samples were taken. One sample was analyzed for uranium and plutonium and one was stored.
2. After transfer of feed solution two samples were taken. One sample was analyzed and one was stored.
3. If the range of both uranium and plutonium analyses was less than 5% of the average, the analyses were accepted and the stored samples were discarded. If the range was too great, the stored samples were analyzed and the first set of analyzed samples were discarded.
4. The two samples accepted were re-analyzed by the control unit and then stored for analysis by another laboratory.

Twenty-eight batches of feed solution were prepared using this sampling and analysis plan. The results were not greatly different from those previously given. The pooled variance for sets of two samples was 11.9 for uranium compared to 7.6 for sets of four. The variance for the difference between the first and second time of uranium analysis for sets of two samples was 162 compared to 110 for sets of four.

It was disturbing, however, to notice that there was a definite difference in analyses of samples taken before transfer and after transfer. For uranium, the average difference was 8 ± 7.3 grams/liter and for plutonium was 0.004 ± 0.004 grams/liter. In fact, in checking with the operations group it was found that a slight leakage of either water or steam was diluting the feed solution. With such a slight leakage it is difficult for operations to determine whether or not the leakage has been stopped within the time available. Renewed efforts are being made to eliminate the source of dilution.

3.0 POWER REACTOR FUEL RECOVERY PROCESSES

All currently considered fuel recovery processes include a head-end step for converting the spent fuel to an aqueous-nitrate solution required as feed for solvent extraction. Two general head-end treatments have been developed.

1. The Darex Process in which the clad and core are dissolved together with 5 M HNO_3 - 2 M HCl. After the dissolution is completed, the chlorides are removed by distillation.

2. The Sulfex Process in which the clad is dissolved with 6 M H_2SO_4 and removed from the dissolver. The core is dissolved separately with 13 M HNO_3.

A flow diagram of a scheme including either or both of these two approaches is shown schematically in Fig. 6; the zone where accountability measurements of plant input are to be made is emphasized.

With power reactor fuels, undissolved solids will present a problem to accountability measurement. These solids may be present in either the clad solution or in the core solution and must be removed before the solution is introduced to the solvent extraction system because the mechanical operability of the system will be poorer with solids present and the concentration of valuable material in the solids will be markedly different from the solution. In any event for accountability purposes the solids must be separated, sampled and the valuable material content determined.

Entrainment in the vapor evolved from boiling feed solutions results in carryover of valuable material to the offgas scrubber bottoms and the waste acid

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fractions. It therefore becomes necessary to conduct accountability measurements on these solutions.

Summarizing, the plant input will consist of a summation of the valuable material measured in core and clad solutions, neutralized waste acid, scrubber bottoms and waste solids.

4.0 ACCOUNTABILITY MEASUREMENTS IN PROPOSED POWER REACTOR FUEL REPROCESSING PLANT

Two approaches to an accountability program may be taken. One is to determine plant input from volume measurements and chemical analyses of dissolver solution, correcting for losses occurring during the decladding and core dissolution steps; the other is to determine the plant output from volume measurements and chemical analyses of all product and waste streams leaving the plant. In all cases volume measurements and chemical analyses contribute to errors in measuring material processed.

4.1 Volume Measurement

Two methods of making volume measurements have been considered: (1) from solution weight and density and, (2) from solution liquid-level and density as determined with pneumatic-type instruments. In general the first method is not applicable where highly radioactive solutions are involved owing to the difficult remote maintenance and calibration of weigh tanks. The first method, however is capable of greater accuracy, (limit of error about ± 0.4 per cent) and it is most frequently used in making accountability measurements of finished products The second method is better adapted for remote operation with radioactive solutions but is less accurate; i.e., the limit of error is ± 0.6 to 0.7 per cent.

It is felt that volume measurements based on liquid-level density data determined with pneumatic-type instruments will be satisfactory. However, a few comments as to the accountability vessel design, instrument array and calibratio

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measurement procedures seems pertinent.

Vessel Design

1. Length to diameter ratio of the vessel should be as high as practical.
2. Equip the vessel with an effective sparging device and a recirculating type sampler.
3. Equip the vessel for heating or cooling.
4. Design to drain the vessel completely.
5. Positively eliminate leakage both to and from the vessel.
6. Provide means for introducing and withdrawing calibrating solutions in weighed amounts directly to or from the fessel. Location of the vessel will be an important item in this regard.
7. The vessel should be isolated from other process equipment (unit shielded) so that it may be quickly decontaminated for maintenance repair, inspection, and recalibration.

Instrumentation

The instrument array should include the following:

1. Temperature recorder.
2. Pneumatic-type liquid level and density instruments of highest practical sensitivity and reliability. Dual installations supported with sensitive manometers should be installed for detecting drift from zero and for rapid calibration. Transmitters of highest degree of linearity should be used. Instrument output should be recorded using the largest charts practical. Chart drives should be positive, and the chart paper chosen to eliminate error due to pen drag. Charts should be easily read.
3. The instruments should be rugged and accurate.

Calibration-Measurement Procedure

1. Instruments should be calibrated and zeroed frequently.
2. Vessel calibration should include calibration with water and with solution having the composition and physical characteristics of that expected in the process. The effect of temperature should be determined. Any hysteresis effects should be measured.

3. The procedure should consist of introducing weighed quantities of solution to the vessel and noting the liquid level and density. Similar data should be determined starting with a full vessel and weighing quantities withdrawn from the vessel.

4. Temperature effects should be measured and the coefficients determined.

5. The minimum sparging time necessary to insure solution homogeneity should be determined. This consists of adding a quantity of concentrated solution to the vessel and then a quantity of water. Air sparge the mixture for intervals, sampling after each interval until uniform solution composition is obtained.

6. The minimum sample recirculation time necessary to insure representative samples should be determined by comparing samples of the solution withdrawn from the sampler with those withdrawn simultaneously from the vessel directly.

4.2 Chemical Analyses

The accountable materials occurring in reactor fuels of current interest are uranium, plutonium, and thorium. In the future possibly neptunium, americium, curium and other members of the heavy elements may be included. The analytical procedures for these elements of current interest are as follows:

Uranium

a. Potentiometric Method - currently is used to assay product solutions decontaminated from ionic impurities and fission products. The limit of error with such solutions is ± 0.5%. If the method is to be applied to radioactive dissolver solutions it must be modified for remote operation and include provisions to eliminate interfering ionic material typical of such solutions; both modifications will tend to increase the limit of error.

b. Ammonium Thiocyanate Colorimetric Method - currently used with radioactive dissolver solutions. The limit of error is about ± 3%. In general this method is satisfactory for process control but is not as accurate as is desired for accounta-

bility measurements particularly where enriched fuels are involved.

c. Coulimetric Method - currently under development and is reported to compare favorably with the potentiometric method in accuracy; it is adaptable to remote operation with radioactive dissolver solutions and is not subject to interferences by ionic materials typical of dissolver solutions. If predictions as to the accuracy of this method are borne out in practice, it will be preferred over all others.

d. Fluorimetric Method - currently used to assay waste streams where the uranium concentration is very low. The accuracy is about ± 15% at uranium concentrations < 0.01 g/liter.

Plutonium

a. Potentiometric Method - currently used with clean product solutions decontaminated from ionic contaminants and fission products. The limit of error is ± 0.5%. If the method is to be adapted for use with radioactive dissolver solutions, the procedure will have to be modified to remote operation and include a preparation step that increases the plutonium concentration and removes interfering ionic material. The limits of error will increase as a result.

b. Thenoyltrifluoracetone Extraction Method - currently used with radioactive solutions. The limit of error is about ± 5% and requires that the specific activity of the particular lot of plutonium be known. The specific activity is calculated either (1) from an isotope assay of the particular lot of plutonium or (2) from gross α counting data and potentiometric analyses of a sample of the particular lot of plutonium solution. The method provides sufficiently accurate results for process control but is not recommended for accountability measurements other than for waste stream assay.

c. Coulimetric Method - under development; adaptable to remote operation and compares favorably with the potentiometric method in accuracy. Currently, only mercury is known to interfere.

Thorium

Thorium is determined in all solutions by the Spectrophotometric Thoron method. The limit of error is 3%.

4.3 Isotopic Assay

The isotopic composition of uranium and plutonium is determined using the thermal emission mass spectrometer. The limit of error is about ± 0.5%.

5.0 SAMPLING

5.1 Solutions

Solutions must be thoroughly mixed prior to sampling to insure homogeneity. Air sparging for ~ 1 hr is usually satisfactory, provided the depth of solution in the vessel is sufficient. Samplers should be designed for reliable remote operation with radioactive solutions and should not alter the chemical composition of the concentration or the solution. In general, the sampler will be of the recirculating type and the generation of mists and aerosols will be minimized.

5.2 Solids

No clear-cut, reliable method of directly sampling radioactive solids or slurries is currently available. The best approach at present appears to be to suspend the solids and sample the slurry. The analytical laboratory would then decant the supernate, rinse the solids on to a calibrated glass filter so that the volume or solids could be estimated and then an analyses could be conducted on a known (within ± 10%) volume of solids. These measurements are crude but probably would be satisfactory provided that (1) all solids are exhaustively leached of valuable material prior to sampling and (2) the solids account for such a small fraction of the total valuable material that an error of ± 100% in the solids accountability measurements will not be significant.

6.0 ACCOUNTABILITY MEASUREMENTS OF PLANT INPUT AS A BASIS FOR CONTRACT PAYMENTS

Contract payments will be made on the basis of accountability measurements. The payments may be based on the plant input measurements (more or less an instantaneous basis) or on the other hand, may be based on the combined plant output plus losses. Accountability measurement of plant input is convenient in that an early negotiation of contract payment can be made. However, this consideration appears to be trivial in that any time saved here is small compared to that of decay requirements. Only a few additional analytical measurements are required to determine limits of error in measuring plant input. The additional costs are more likely to be reflected in the cost of extra tankage and the more costly construction of the accountability vessels and their accessories than in additional analytical service costs. In general the limit of error will be larger, perhaps a factor of two based on current volume measurement and analytical methods, than that possible in using recovery measurements (product plus loss method).

The limits of error for the first four PRFR spent reactor fuel programs have been determined as shown in Table 1. The results of two other programs have been included for comparison; i.e., Clementine and Thorex, in which no special effort to minimize feed input measurement errors was made. The calculations are based on precision and bias data obtained experimentally for both volume measurement and analysis. In general the limit of error for uranium is 3-4% and about 2-3% for plutonium. The ± 9.4% obtained for the Pu-Al program is inordinately large and requires some explanation. This feed solution contained 5-10 times more plutonium per unit volume than contained in the first program. There is some evidence that the first analyses made may have been invalid. However, all analyses reported were combined, resulting in a higher limit of error than would have been obtained by using selected values.

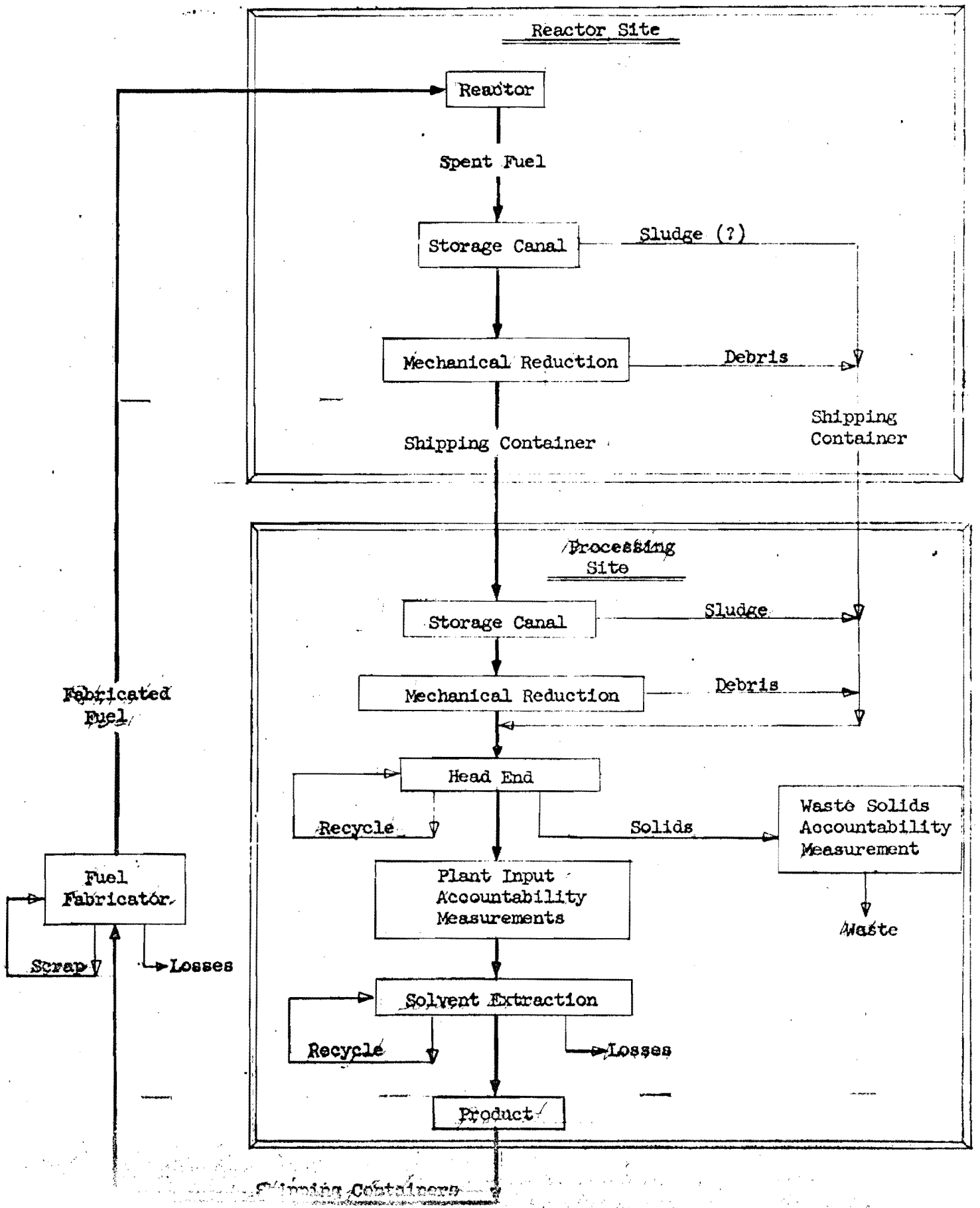
These measurements agree numerically very well with product plus loss measurements. Material balances for both uranium and plutonium have been about 99% for the first four programs. It, therefore, appears that contract payment based on recovery measurements will be associated with units of error of the order of 0.5 to 1 per cent as compared to about 3 to 6 per cent (or perhaps greater as more data are accumulated) based on the current methods of volume and chemical measurement.

Since instrumentation, measuring-vessel design and sampling procedure have been optimized to reduce the contribution to the limit of error of these to a practical minimum, future efforts should emphasize minimizing that of the analytical procedures.

Shipper-receiver differences are observed to be small in those programs in which intact fuel elements were processed (Clementine, Thorex, SRP-E), the feed input measurements spanning those of the reactor operator's estimate based on reactor loading and calculations. The differences are larger in the BNL-1 and Pu-Al programs in which the fuel elements had been sheared and exposed to storage canal water.

WTMcD:GSS:vm

Fig. 1. Schematic Fuel Cycle



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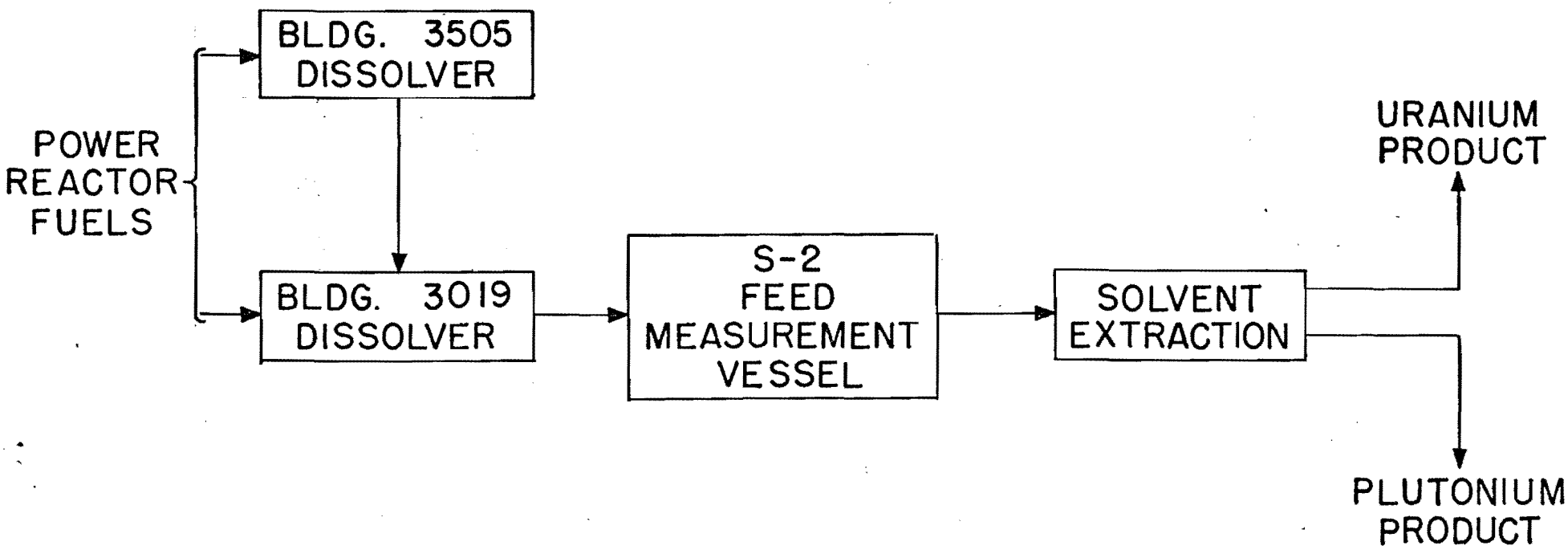

SCHEMATIC REPRESENTATION OF THE ORNL POWER REACTOR FUEL REPROCESSING PILOT PLANT

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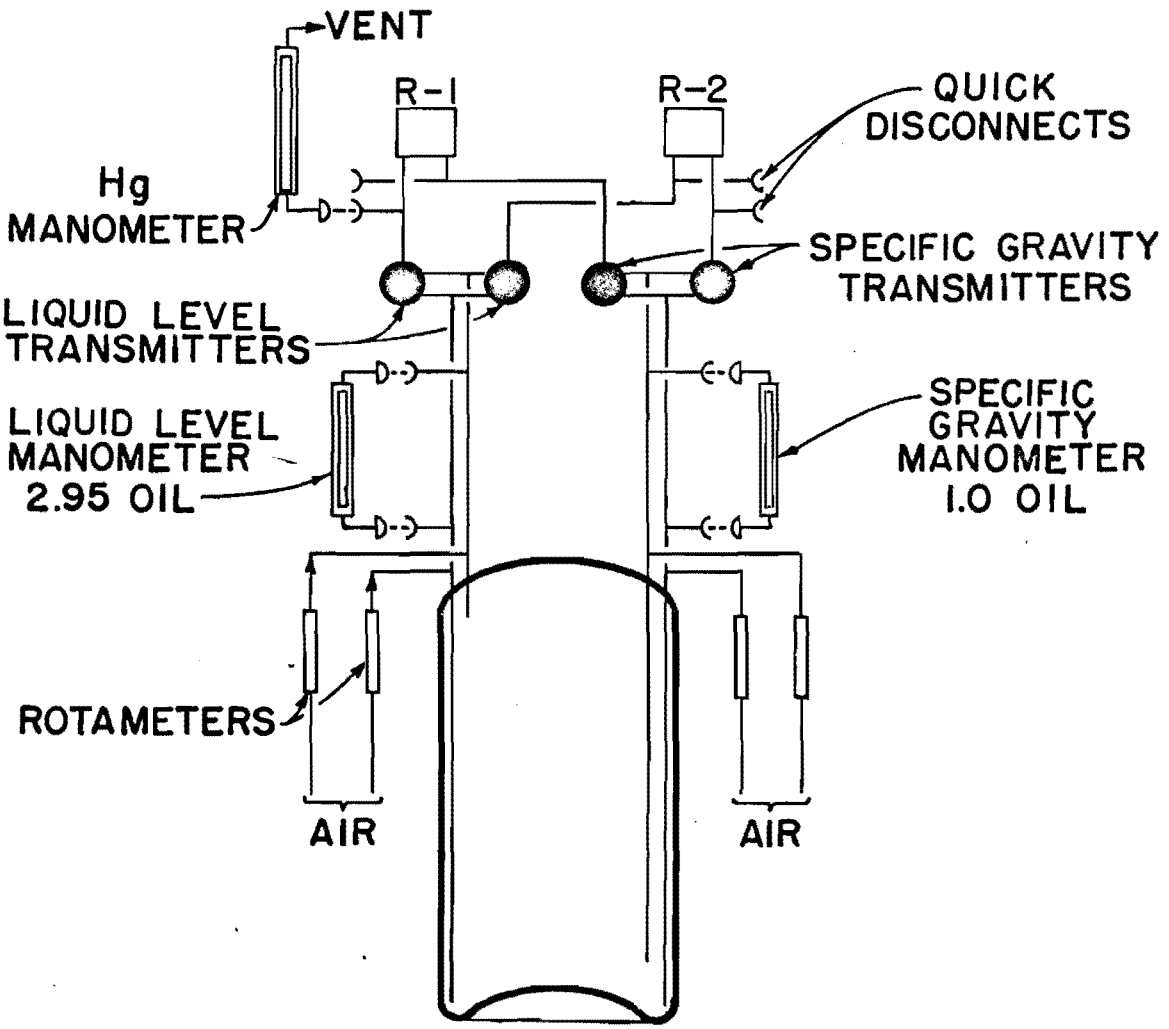

SCHEMATIC VIEW OF S-2 INSTRUMENTATION

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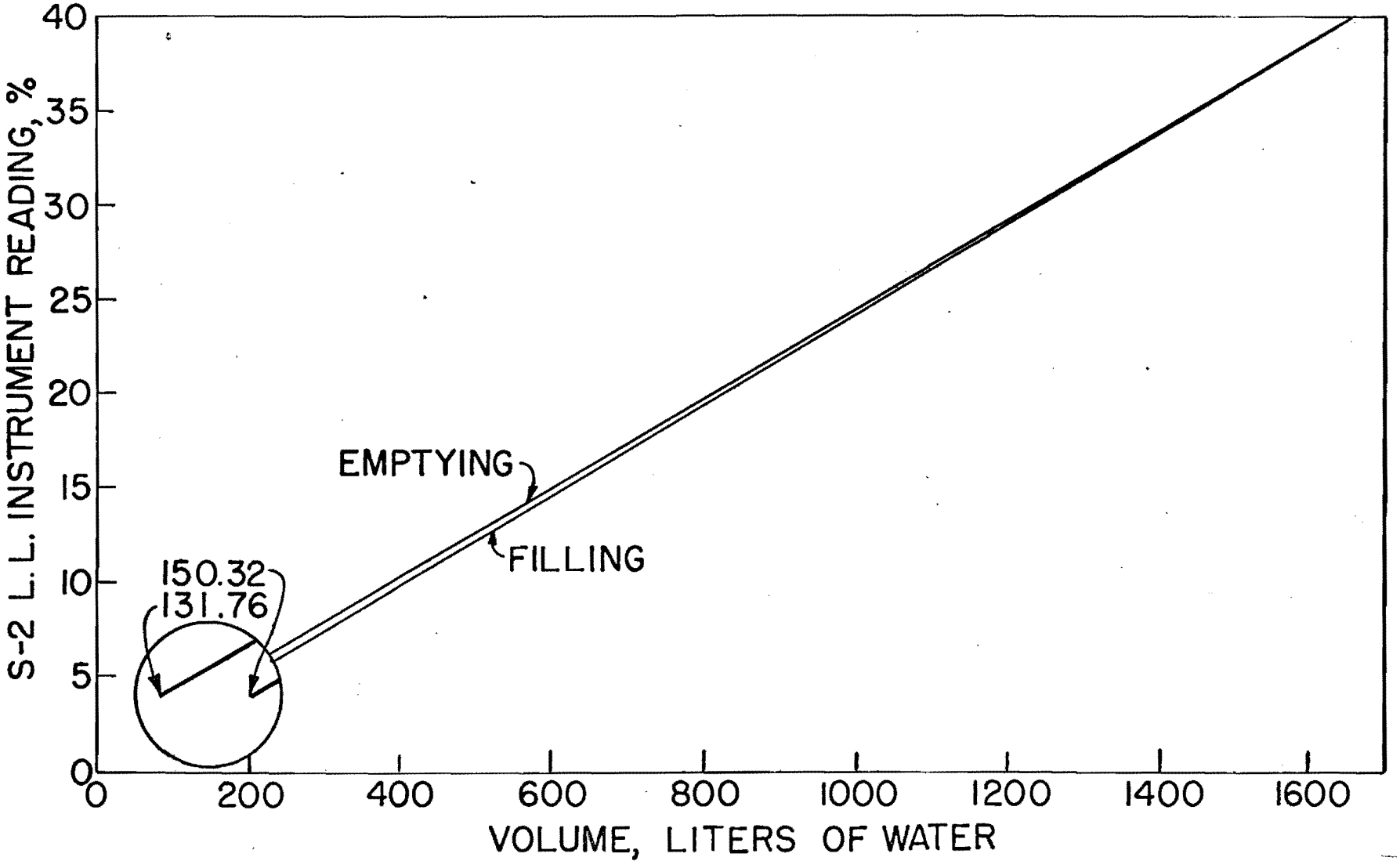


S-2 CALIBRATION

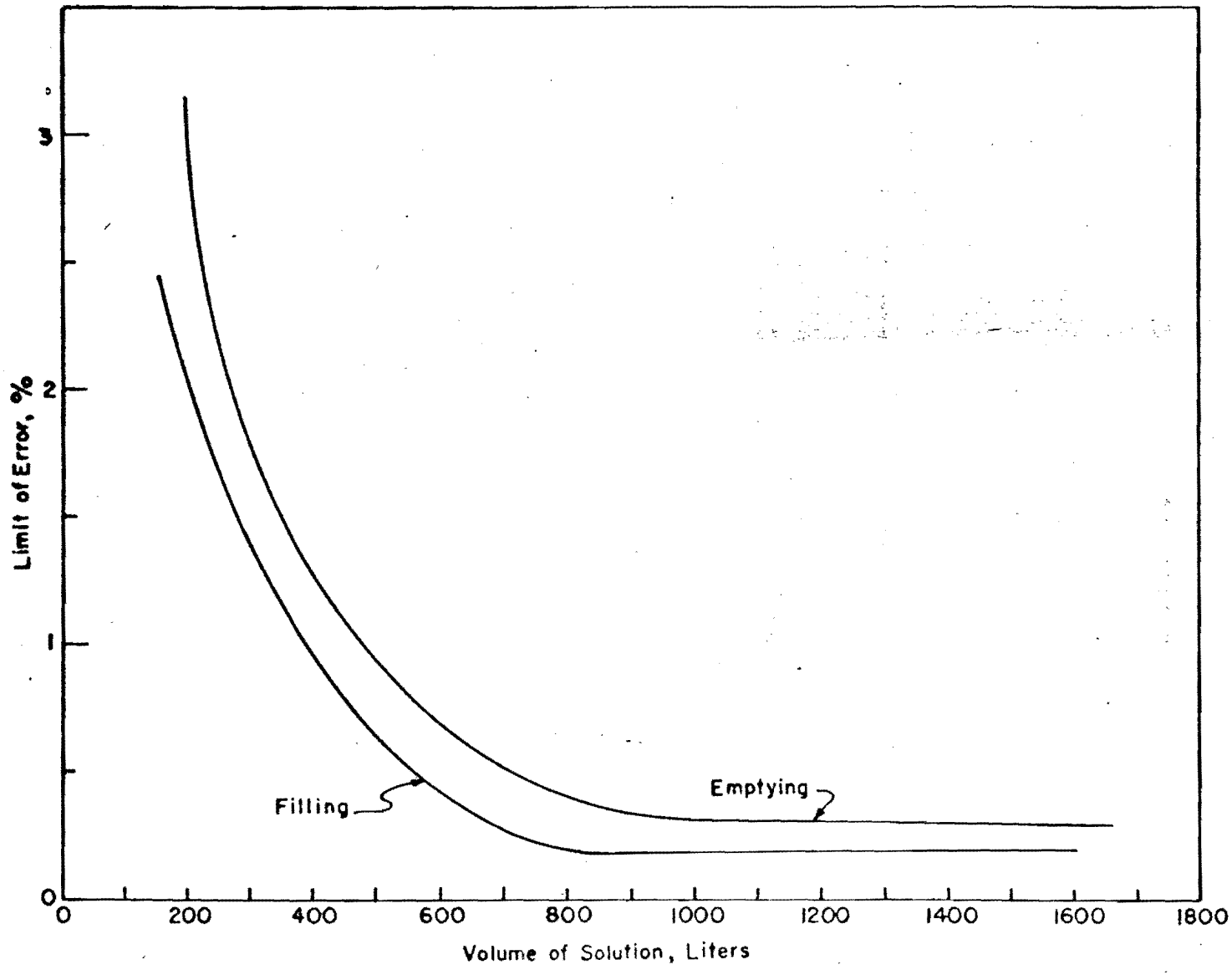
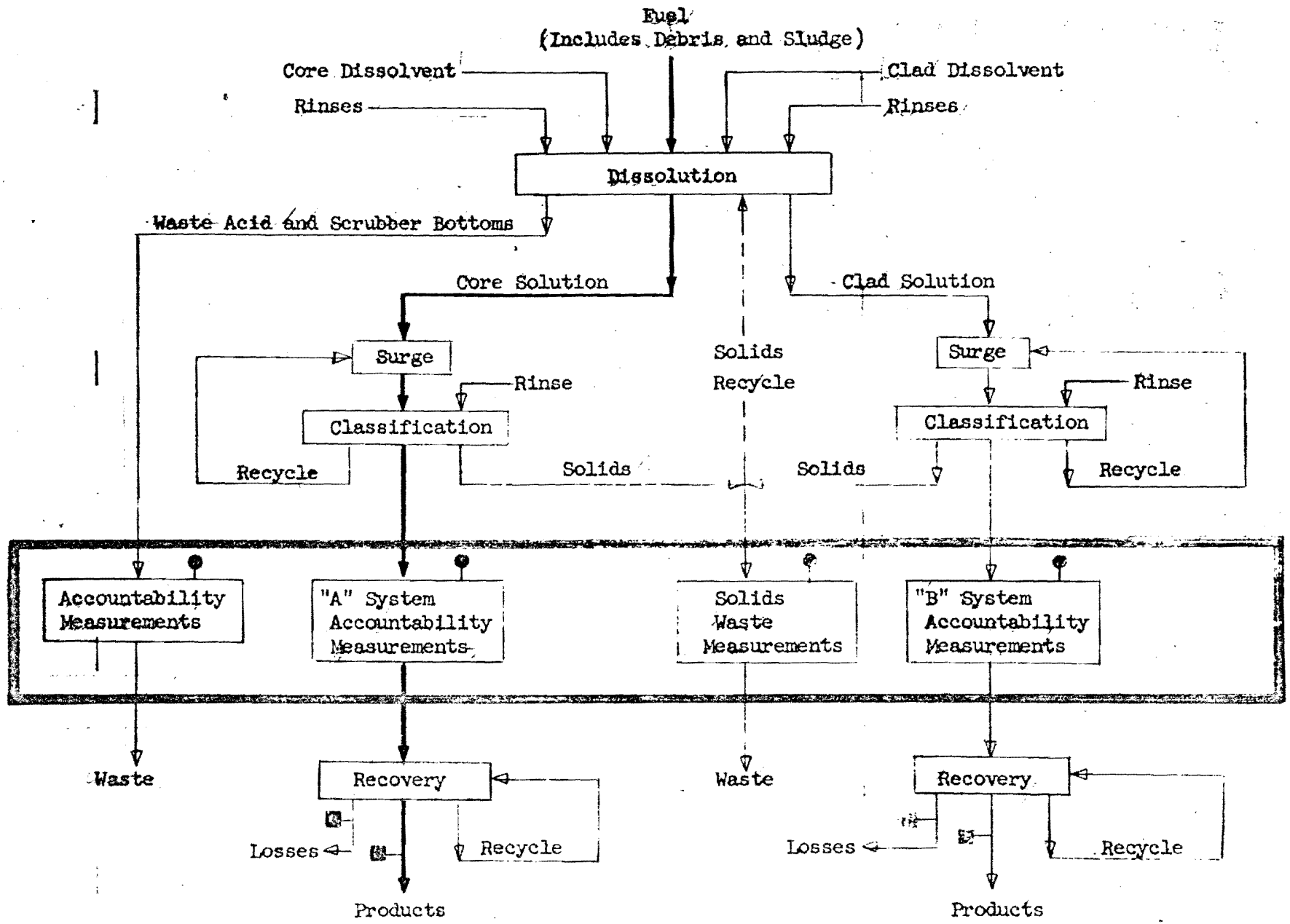


Fig. 5. Limit of Error Curves for S-2 Vessel Calibration

Fig. 6. Power Reactor Fuel Reprocessing Accountability Flow Diagram



Sampling Legend

- Samples for Measuring Plant Input
- Samples for Measuring Product plus Losses

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Table 1. Irradiated Fuel Recovery

[illegible]	Shippers' Estimate			Feed Measurements			Recovery, Product and Losses		
[illegible]	U, kg	U-233, g	Pu, g	U, kg	U-233, g	Pu, g	U, kg	U-233, g	Pu, g
[illegible] fuel	-	-	15,005.5 ± ?	-	-	14,716 ± 440	-	-	14,713 ± 103
[illegible]	-	92,143 ± ?	-	-	92,509 ± 5,551	-	-	87,299 ± 1,053	-
[illegible]-2	5,620.4 ± ?	-	2,163.4 ± ?	5,336 ± 200	-	2,028 ± 55	5,329 ± 38	-	2,019 ± 13
[illegible]1	-	-	1,153.7 ± ?	-	-	1,120 ± 105	-	-	1,111 ± 6
[illegible]3	1,[illegible] ± ?	-	1,[illegible] ± ?	1,495 ± 54	-	1,518 ± 25	1,469 ± 12	-	1,495 ± 10
[illegible]-1	9,818 ± ?	-	5,163 ± ?	9,731 ± 248	-	4,610 ± 108	9,654 ± 69	-	4,609 ± 32

Distribution

1-3. DTIE, AEC
4. M. J. Skinner
5-6. Central Research Library
7. Document Reference Section