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FLOWSHEET FOR THE REPROCESSING OF TYPE 2, RA-1 FUEL ELEMENTS CONTAINING GRAPHITE.

By W.D. Jamrack (IAEA Expert and adviser on reprocessing).

Buenos Aires, December 1967.

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by W.D. Jamrack (I.A.E.A. Adviser on Reprocessing)

1.- INTRODUCTION

The first charge of fuel elements in the RA-1 reactor consisted of Uranium/aluminium plates irradiated to low power level. A second charge, (and later charges), has now been made, consisting of uranium oxide/graphite rods, canned in aluminium tubes, so as to allow a much higher irradiation.

This type of fuel element cannot be dissolved completely, with the aluminium, in nitric acid, since the aluminium would reduce the acid concentration to such an extent that the subsequent leaching of the uranium would be inefficient. Consequently, it is necessary to design a mechanical decanning process to remove the aluminium before dissolving the uranium. Normal methods of mechanical decanning, which involve stripping the aluminium from the complete fuel rod, are not applicable, since the fuel is brittle and would break into small pieces. Similarly, a chemical decanning process, involving dissolution of the aluminium in alkali would, inevitably, give a fairly highly active alkaline solution, heavily contaminated with powdered fuel, and yet the main proportion of the fuel would still be in large pieces which are not suitable for the subsequent leaching action of the nitric acid.

A preliminary mechanical treatment is proposed which involves cutting the whole fuel element into small pieces, separating the aluminium, and grinding the fuel to a powder. Uranium losses are expected to be a little higher than those from

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reprocessing of the more conventional uranium/aluminium fuel elements from experimental reactors.

Some non-active experiments have been carried out to obtain dat on the preliminary mechanical treatment stages, but, because of the limited time and equipment available, it has not been possible to complete a thorough investigation.

2.- DESCRIPTION OF THE TYPE 2, RA-1 FUEL ELEMENTS

The RA-1 reactor has completed the irradiation of a full charge of the new type of fuel elements, a second charge is ready for irradiation and (at least) a third charge is to be manufactured.

Each charge consists of about 220 fuel elements containing 19.7% enriched uranium and about 30 elements containing 10% enriched uranium, both elements being of the same uranium oxide/graphite type.

The reactor has operated on the first charge for 1 1/2 years at a power level of up to 100 KW and for the later charges, with additional cooling facilities, a power level of 150 KW may be achieved. The irradiation time and dose rate do not affect the principles of the reprocessing flowsheet and a standard irradiation of 550 days, at 100 KW will be assumed for each charge. This is 55 MW days irradiation over about 16 Kg uranium, or say 3,400 MWD/T. (This figure may, in fact, be excessive, if the reactor has not operated for the whole of every day). In view of the variation in irradiation conditions from one reactor charge to another, and the variable decay period, it is of little value to calculate the precise fission product and plutonium content of this irradiated uranium.

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Comparison with other fuels with similar irradiation conditions and a decay period of 150 days suggests that the total beta plus gamma fission product content should then be about 400 curies per Kg and the plutonium content about 2 g per Kg. The gamma activity (over about 0.1 MeV) is thus about 160 curies per Kg (after 150 days decay).

Each of the 250 fuel pins is 65 cm in length, with a fuel length of 55 cm, one end having a taper. The fuel is canned in cylindrical aluminium tubes of external diameter 9.8 mm, and wall thickness 1 mm, the diameter of the fuel itself being 7.7 mm.

The fuel, as originally canned, consists of single rods of uranium dioxide and graphite which have been calcined at about 300°C. The fuel is made by extruding a mixture of uranium oxide, graphite and pitch. Each element contains the following quantities of materials:-

UO_2	67 to 71 g	(say 69 g)
Carbon	32 to 33 g	(say 32 g)
	(of which 9 g was from calcination of the pitch)	
Aluminium can	50 to 52 g	(say 51 g)
Total weight	149 to 156 g	(say 152 g)

Each fuel rod, with say 60.8 g uranium, will be assumed to contain 125 mg plutonium. A 973 gram uranium (less loss with aluminium = 953 g) batch for reprocessing, will arise from 16 elements.

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3.- DECANNING EXPERIMENTS

It was assumed that decanning could only be carried out conveniently by cutting the whole fuel element into pieces.

3.1 Simple Guillotine Cutting

The first experiments, with non-irradiated scrap elements, were with a simple hand-operated guillotine having blades about 20 cm in length. Elements were cut into various lengths from 0.3 cm to 2 cm. The whole of the fuel remained enclosed by the aluminium can in the case of the larger size pieces (2 cm and 0.5 cm), and frequent blows with a hammer failed to dislodge it. At 0.4 cm, the aluminium pieces were compressed to an oval shape and the fuel fell away completely from about half of them, in the form of pieces varying from about 0.5 cm down to powder. The others had various proportions of fuel trapped in them.

At 0.3 cm, the aluminium rings were compressed so that the face leaving the knife blade became almost a line and the other face was usually oval. Under these circumstances, a high proportion of the fuel fell out of the rings and was disintegrated to a powder. However, a few percent of the fuel remained firmly trapped in this shape of ring.

3.2 Rolling

Elements were passed through a simple hand rolling mill at various pressures. At a fairly high pressure, the fuel element was pressed so that its smallest dimension was about 30% of the original diameter. The element split at the edges, wide open cracks appearing every few centimetres.

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It did not appear practicable to make use of this phenomenon for any simple decanning process. For example, the resulting powdered fuel did not fall out of such a rolled can easily.

Rolling at a lower pressure, so that the can did not split, powdered the fuel fairly well, but again it would not shake out of the element in any simple operation. The powder did fall out of the pieces of can fairly well if such rolled elements were subsequently cut to 0.5 cm pieces in the guillotine. This process is regarded as rather complex for remote operation in a heavily shielded cell, since it involves rolling, cutting and shaking.

3.3 Cutting with Special Guillotine Blades

Fuel elements were cut with blades which were made specially with semi-circular notches, in an attempt to avoid compressing the small rings and trapping particles of fuel. At 0.4 cm, the rings were circular and most remained filled with fuel. Very little compression of the rings was observed, and it was possible to vibrate the fuel cleanly from many of the pieces by simple shaking in a tube. Some was not removed, even after one hour on a light shaking machine, but it was estimated that more vigorous treatment in a ball-mill, or particularly a vibrating ball-mill, would probably dislodge all the fuel.

0.3 or 0.2 cm rings become compressed, but with this special blade they were not completely closed. Most of the fuel was removed on cutting the element, but a little was retained, in a position less tightly trapped than that from the straight blades.

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It is probable that a suitable process could be devised using blades with semi-circular notches, cutting into 0.2, 0.3 or 0.4 cm lengths, followed by vibro-ball milling. The precise size of cut can be decided best when a vibro-mill is available to complete the process. The irradiated fuel may behave slightly differently from the non-irradiated material but, if necessary, the writer suggests that a small loss of fuel with the can should be accepted. Less than 1% loss is anticipated.

4.- DISSOLVING EXPERIMENTS

4.1 12N Nitric Acid

Experiments were with a large excess of acid so as to maintain the concentration fairly constant, particularly in case some of the carbon reacted. 150 cc of 12N acid was used with 10 g of the unground fuel from a fuel element which had been cut into 0.3 cm lengths.

The nominal weight of uranium per sample was 6.02 g. The solution was maintained boiling under reflux for 1, 4, 16, and 24 hours, and gave uranium concentrations of 43.8 g/l, 44.4 g/l, 45.0 g/l, and 48.9 g/l respectively. The nominal concentration for complete dissolution was 40.1 g/l, so the results suggest fairly complete dissolution coupled with gradual evaporation. The final volume after 24 hours was measured as about 125 cc, which partially agrees with this conclusion.

The final weight of undissolved material was 2.0 g, which suggests only little dissolution of the graphite. The final uranium content of this residue was determined by analysis as

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7.8 mg, which gives an extraction efficiency of 99.9%. In view of this favourable result, a similar experiment with a sample of fuel which had been ground to pass 48 mesh was abandoned.

4.2 Fuming Nitric Acid

Similar experiments with fuming nitric acid gave uranium concentrations of 40.3 g/l, 44.0 g/l, 44.8 g/l, and 47.7 g/l, after 1, 4, 16, and 24 hours respectively. The final volume was 145 cc and the weight of residue 1.2 g.

The final uranium content was determined by analysis as 14.4 mg, which gives an extraction efficiency of 99.7 %. These results are similar to those with 12N acid but an obvious difference was the large quantity of nitric fumes at the beginning and the reduction in quantity of graphite remaining after a few hours. In view of the results, an experiment with ground fuel was again considered not to be necessary.

4.3 6N Nitric Acid

Similar experiments with 6N nitric acid used only 5 g of fuel per 150 cc, since this was sufficient acid for complete oxidation of the carbon. (The results of the first two experiments suggesting only partial oxidation of the carbon were not available at the time these experiments were started). One experiment was with unground fuel, and the other with fuel ground to pass a 48 mesh sieve. Uranium concentrations after 1, 4, 10, 12, and 16 hours were respectively 22.4 g/l, 22.2 g/l, 22.4 g/l, 22.5 g/l, and 22.5 g/l for the unground fuel and 19.7, 20.0, 20.3, 20.6, and 20.7 g/l for the ground material.

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with no volume change, 100% extraction would have given a concentration of 20.1 g/l, so the efficiency is assumed to be fairly high.

In view of the above results, it seems reasonable to dissolve the uranium in 12N nitric acid, allowing the acidity to fall a little. This avoids the possibility of a high corrosion rate with fuming acid, and the general difficulty of handling this acid.

The flow properties of the residues are such that the ground fuel is preferred, so as to minimize the possibility of blockages in the pipes of the filter system, even though grinding way not be essential to obtain a high dissolution efficiency.

5.- PROPOSED REPROCESSING FLOWSHEET

Four fuel elements, of weight 608 grams, and packing volume about 0.5 l, would be cut into lengths of about 0.3 or 0.4 cm, by means of a guillotine. The precise length of a piece would be decided by experiment. The pieces would then be vigorously vibrated in a vibro-ball mill and the aluminium rings separated from the fuel by sieving.

Four batches of fuel from the mill, weighing about 1.6 Kg, would be combined and re-milled in the absence of aluminium rings, to provide a ground powder for dissolving, containing 953 g uranium. It is a little difficult to decide this batch size until dissolving experiments have been carried out with non-irradiated scrap fuel elements, since the volume of suspended graphite must be small enough to be passed to the filters and washed there. Also, the combined volume of the dissover liquor

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and wash liquor must not exceed the volume of the conditioning vessel.

It is proposed, in the first instance, to dissolve the fuel in a very light aluminium container, using only about 8 l of 12N nitric acid. This would allow the acidity to be reduced to about 10.5 N as a result of the reaction with uranium and aluminium, and give a uranium concentration of about 119 g/l, with 0.5 Kg of graphite occupying a settled volume of about 0.5 l. Filtration could be followed by prolonged washing on the filter with water (actually very dilute nitric acid), or a reslurry wash could be given first by returning the graphite to the dissolver. Since the preferred final acidity of the uranium solution is 3N, the combined volume of washes should be restricted to about 19 l, to give a total volume of 27 l.

The uranium feed concentration would then be 35 g/l. This is the procedure shown on the flowsheet.

If the acid extraction was found to leave more than about 3% of the uranium with the graphite, then a second extraction with nitric acid would be recommended. In order to keep the final volume down to 48 l, under these conditions, it is suggested that the two acid treatments should then be at 12N and 7.5N respectively. The cycle could then be two treatments each with 8 l of acid, followd by washes on the filter of 10 l, a reslurry wash of 10 l and washing on the filter of 12 l.

With the flowsheet uranium concentration of 35 g/l, it should be possible to extract continuously with 20% TBP under conditions such that the loaded solvent has almost the same uranium concentration (59.4 g/l) as that obtained after the first

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extraction cycle of the uranium blanket for the RA3 reactor (1). In the blanket reprocessing, this loaded solvent is obtained as a result of multiple batch extraction, followed by a continuous aqueous scrub and a solvent scrub of the aqueous scrub. This procedure was considered advisable in that case in order to keep a high proportion of the 1,000 curies of activity per batch in the highly active cell, and in order to limit the amount of the highly extractable cerium isotope which enters the solvent when extracting from an aluminium nitrate solution. The present case is different, since the fission product activity is lower and, by excluding high concentrations of aluminium, the fission product and uranium distribution coefficients are much less.

A completely continuous solvent extraction process is, therefore, recommended for RA-1 graphite type of fuel elements. The 160 curies of gamma activity after 150 days decay, is still rather high to move from the highly active cell, and it should be noted that the graphite filtration problem makes this type of movement necessary also, since the filters are located outside the cell. However, the reprocessing of the graphite type fuel elements is not regarded as very urgent, and a considerable additional decay time can be allowed before reprocessing. Much of the activity after only 150 days decay is from the gamma active Zirconium 95 isotope (and its Niobium daughter) with a half-life of only 65 days. Decay for a full year, therefore, i.e. an extra 214 days, would reduce the Zirconium/Niobium activity by a factor of about 10, and the total gamma activity, per 16 element batch would probably then only be about 40 curies. This long decay time has, for these reasons, been incorporated in the flowsheet.

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After the first extraction of uranium into solvent, followed by scrubbing with nitric acid, it is possible to proceed according to the flowsheet for RA 3 blanket reprocessing, except that the volumes per batch are different.

It is not practicable to accumulate the first cycle uranium product into larger batches for reprocessing in longer runs, as for the RA 3 blanket, since sixteen fuel elements is the most convenient batch size for criticality control, without the use of special geometrically safe tanks for storage, etc.

It should be noted that, although the plutonium content of the RA-1 graphite type elements is less,per batch, than the RA 3 blanket, the ratio of plutonium to uranium is likely to be higher. Consequently, it is recommended that the full plutonium splitting and purification cycles are retained, as for blanket material, since this plutonium can make a significant contribution to the total quantity required in the early stages of plutonium fuel element development.

6.- EQUIPMENT AND PROCEDURE FOR THE PREPARATION OF FUEL FOR DISSOLVING

The sixteen fuel elements per reprocessing batch could, after a full year's decay, be transferred together to a special dry shielded cutting and milling cell, in a standard type of shielded container. Such containers, for a few tens of fission products, will be required by the Metallurgy or Reactor Departments for their work on the examination of irradiated fuel elements. It is even possible that the milling and cutting cell could be used jointly by those wishing to examine the fuel elements from RA-1 (or even RA-3

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if the shielding is suficcient). The milling and cutting cell might, therefore, be build for general purposes, to a general purpose design, with standard types of shielded container, none of which need be discussed in the present report.

Within the cell, a simple pneumatically operated guillotine should be provided, with a 20 cm blade, the blade and anvil having notches to fit the circular shape of the graphite type element. Elements would be cut singly into 0.3 or 0.4 cm lengths, or another suitable length decided by experiment. The pieces would fall directly from the guillotine into a vibro-ball mill of about 2 l capacity, having simple clamp type openings at top and bottom. The mill should be capable of easy operation, including opening and closing of the clamp-type doors, by means of manipulator tongs, through the shield wall. Suitable laboratory types of vibro-mill are made commercially, which might only require modification of the entry ports and clamps for use behind shielding. The mill should be about half full of steel balls. The best size and weight of ball charge, the vibration amplitude and frequency, and the time cycle, would be determined by experiment with scrap, non-irradiated, fuel elements. It is probable that a wide variety of conditions would be satisfactory for releasing the fuel powder from the aluminium rings. A mill charge would comprise the cut pieces from four of the sixteen elements. After each milling operation, the mill would be emptied from the lower port, with the aid of very brief periods of vibration. The mill should have fitted to this port, a very wide mesh grid, say 2 cm, sufficient to just retain the balls but allow the powder and rings to fall through. The powder and rings

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should fall into a funnel containing a grid of about 0.4 cm mesh, sufficient to retain the rings and allow the powder to pass through into a small container. When four batches of powder had been obtained in this container, i.e. from the whole sixteen elements, this larger batch should be returned to the vibro-mill, free from rings in order to grind to pass 48 mesh. The conditions of milling the powder may require to be changed from those needed for the initial separation of the rings, but it should not be necessary to change the balls. The final powder size need not be controlled with great precision, and a sieving operation is not necessary, once the milling conditions have been established in experiments with non-irradiated fuel.

The powdered charge from 16 fuel elements will weigh about 1.6 Kg and should have a packing volume of about 0.8 l. In order to provide a clean transfer to the reprocessing plant, it is proposed to collect this powder in a bag (or bags) made of aluminium foil. These weigh only a few grams and they are readily available for the packaging of dried foods. A simple welding or crimping device for sealing the bags would be required inside the cell. Alternatively, a light aluminium can, with an interference fitting lid, could be used, but it is suggested that the total weight of aluminium should not exceed about 75 grams. This would give an aluminium concentration of only about 0.1 M, on dissolution, and it should not have more than a small effect upon the distribution coefficients of the fission products at the solvent extraction stage.

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7.- DISSOLVING PROCEDURE

A coffin has been designed, and a procedure devised, for the transfer of RA-3 irradiated core boxes and blanket boxes into the reprocessing plant dissolver. Whilst this is much larger than necessary, in the absence of other equipment it should be satisfactory for transferring an aluminium foil bag or cannister, containing the ground graphite fuel, into the same dissolver.

A little mercury may be used to achieve complete dissolution of the aluminium in the 12N nitric acid, and to provide rapid penetration of the aluminium so as to allow the uranium to dissolve quickly.

The equipment already exists for passing the dissolver solution, with the finely divided graphite, to the stainless-steel filter, coated with a few grams of a diatomaceous earth type of filter-aid. This filter is located in a shielded cavity in the cell wall, but not in the cell itself.

This filter enclosure could have additional lead bricks provided if the present ones are insufficiently thick for shielding when filtering the solution obtained from the graphite type elements. If 40 curies of gamma activity are present in a 9 l batch, the gamma activity fed to the filter (of about 2 l capacity) at any one time, should only be about 10 curies.

The volume of the filter is about 2 litres, and it is anticipated that, after filtration is complete, this should only have to contain about 0.5 litres of graphite and a little diatomaceous earth.

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Facilities are available for the reslurry and return of the filter-cake to the dissolver. The graphite powder should be fine enough to suspend satisfactorily in the solution and not cause blockages.

8.- SOLVENT EXTRACTION

This requires little comment. The first extractor would be used for extraction and scrub as in the case of type 1, RA-1 fuel plates, and unlike the RA-3 blanket, where it is only used for aqueous scrub and solvent scrub. The second extractor, as always, would be for first cycle strip. From this point, the RA-3 blanket flowsheet would be followed, which uses the second cycle for extraction, scrubbing and plutonium stripping only. Uranium stripping is then carried out in a new extractor, under relatively non-active conditions, located in a ventilated area at the end of the existing plant. This is to be installed, in any case, for the RA-3 blanket.

9.- PLUTONIUM PURIFICATION AND EXTRACTION

This, also, follows the RA-3 blanket, using the same ion-exchange and evaporation equipment.

10.- SOLID RESIDUAL DISPOSAL

It will be necessary to have facilities for the disposal of aluminium scrap and wet graphite, both of which will be contaminated with irradiated enriched uranium. These facilities will be needed, in any case, for the residues from the fuel element examination, and will not be discussed in the present report. Ground disposal, under Health Physics control, would probably be appropriate.

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11.- CONCLUSIONS AND RECOMMENDATIONS

1. Reprocessing of the type 2, RA-1 fuel elements is possible in the existing pilot plant, as modified for the RA-3 blanket reprocessing, without batch extraction, but also using the feed filter.

2. A decay period of one year is recommended before reprocessing.

3. A mechanical pre-treatment process is recommended, in which the elements are cut into small pieces, vibro-milled to loosen the aluminium rings from the fuel, the fuel sieved away from the rings, and then vibro-milled to 48 mesh.

4. A batch size of 16 fuel elements is suggested, which is safe for criticality purposes (containing less than 200 g Uranium 235) and suitable for filtration.

5. Losses of a few percent should be allowed, most of this being sent to solid waste storage with the aluminium rings, or with the filtered graphite.

6. Facilities would be needed for the disposal of solid waste.

12.- ACKNOWLEDGEMENTS

The assistance of Dr. J. Flegenheimer and Dr. F. Kaufmann of the Reprocessing team, in discussion of the potentialities of the existing reprocessing pilot plant, is gratefully acknowledged.

The writer is also most grateful for the collaboration of Dr. Aráoz, Dr. Marticorena and Mr. Divito, of the Metallurgy

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Department, in experimental work on the cutting of fuel elements, the separation of aluminium, the leaching with nitric acid, and the analytical determination of uranium contents.

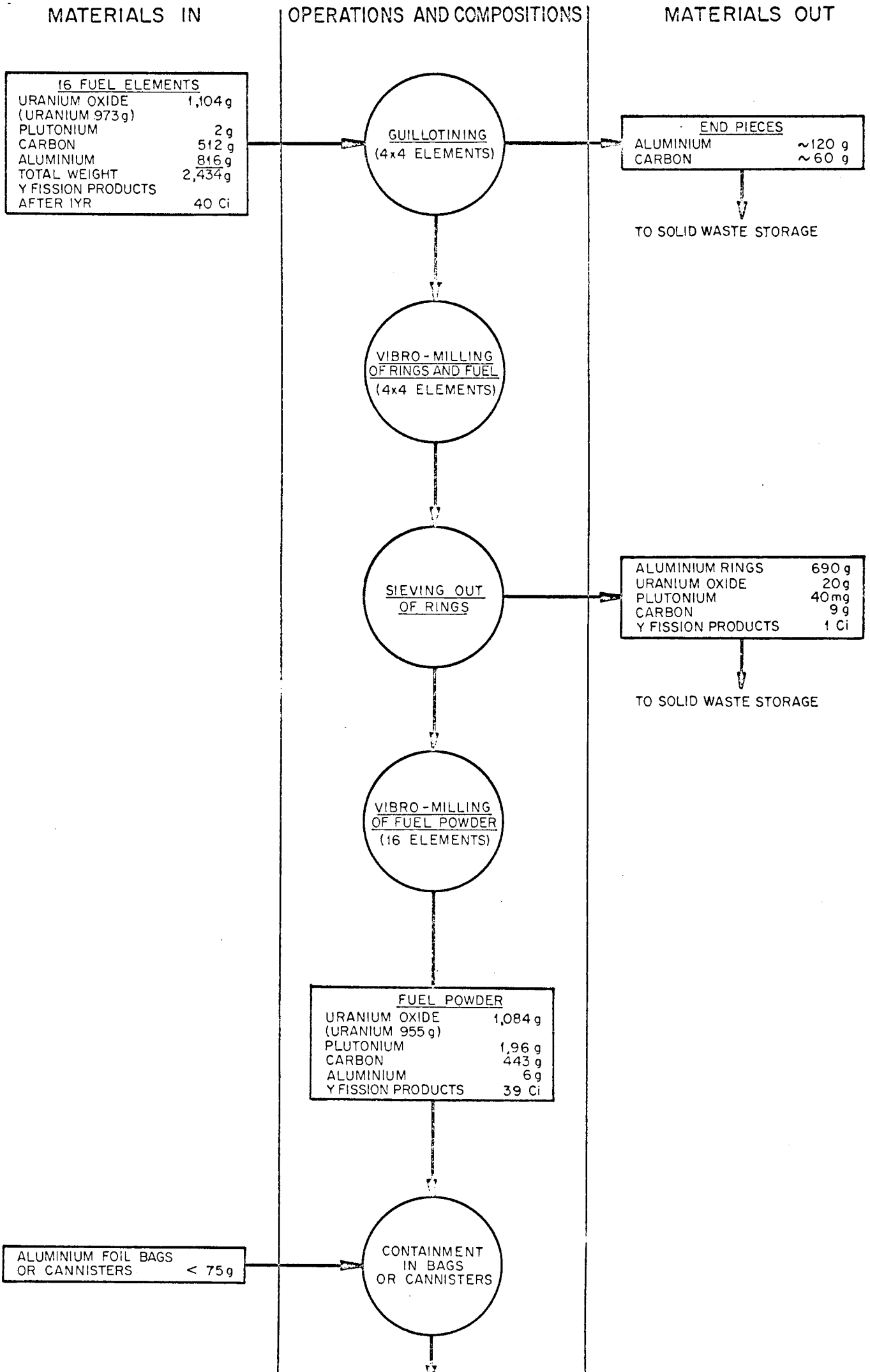
13.- REFERENCE

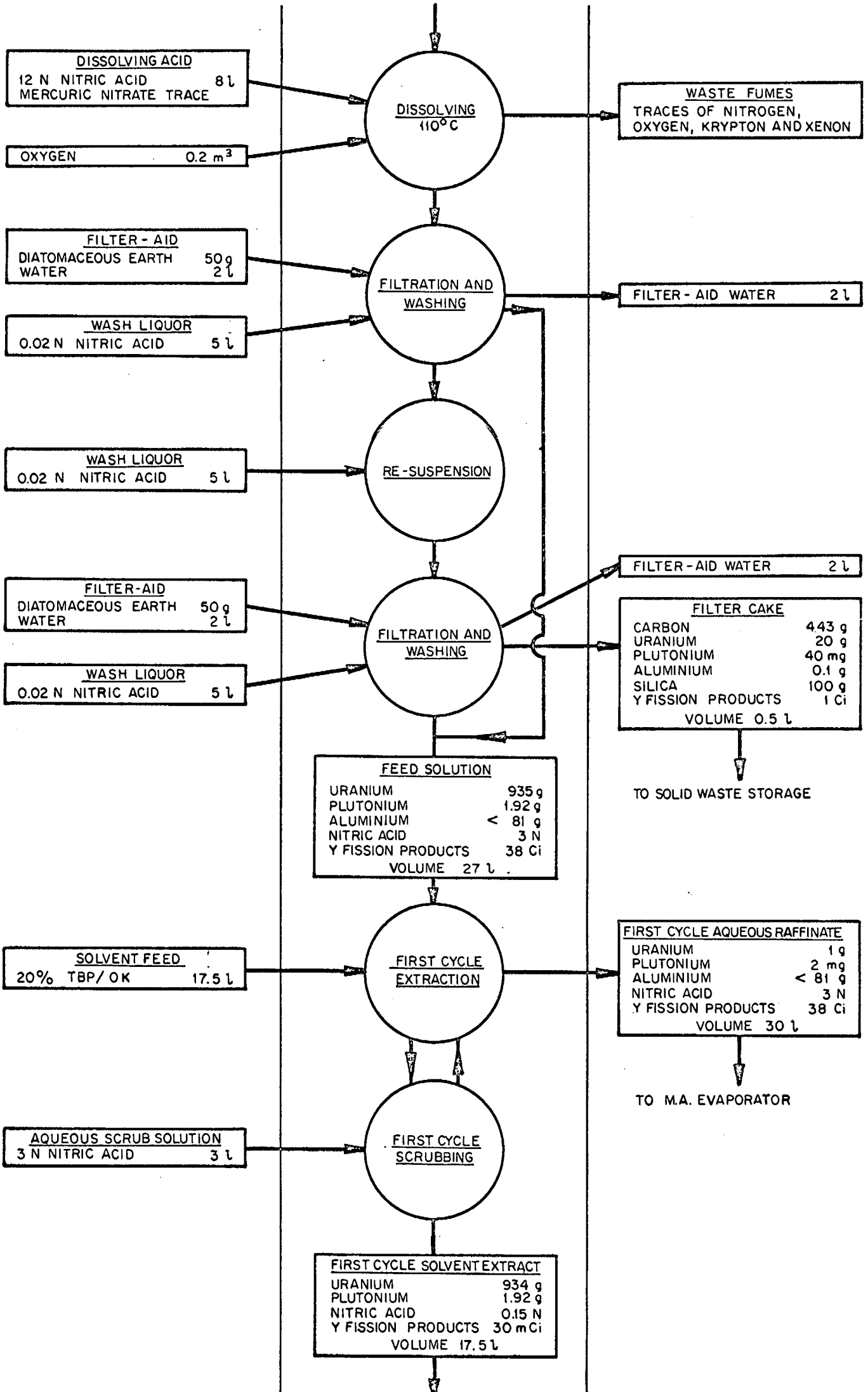
1. Jamrack, W.D. - A Flowsheet for the Reprocessing of the RA-3 Blanket. November, 1967.

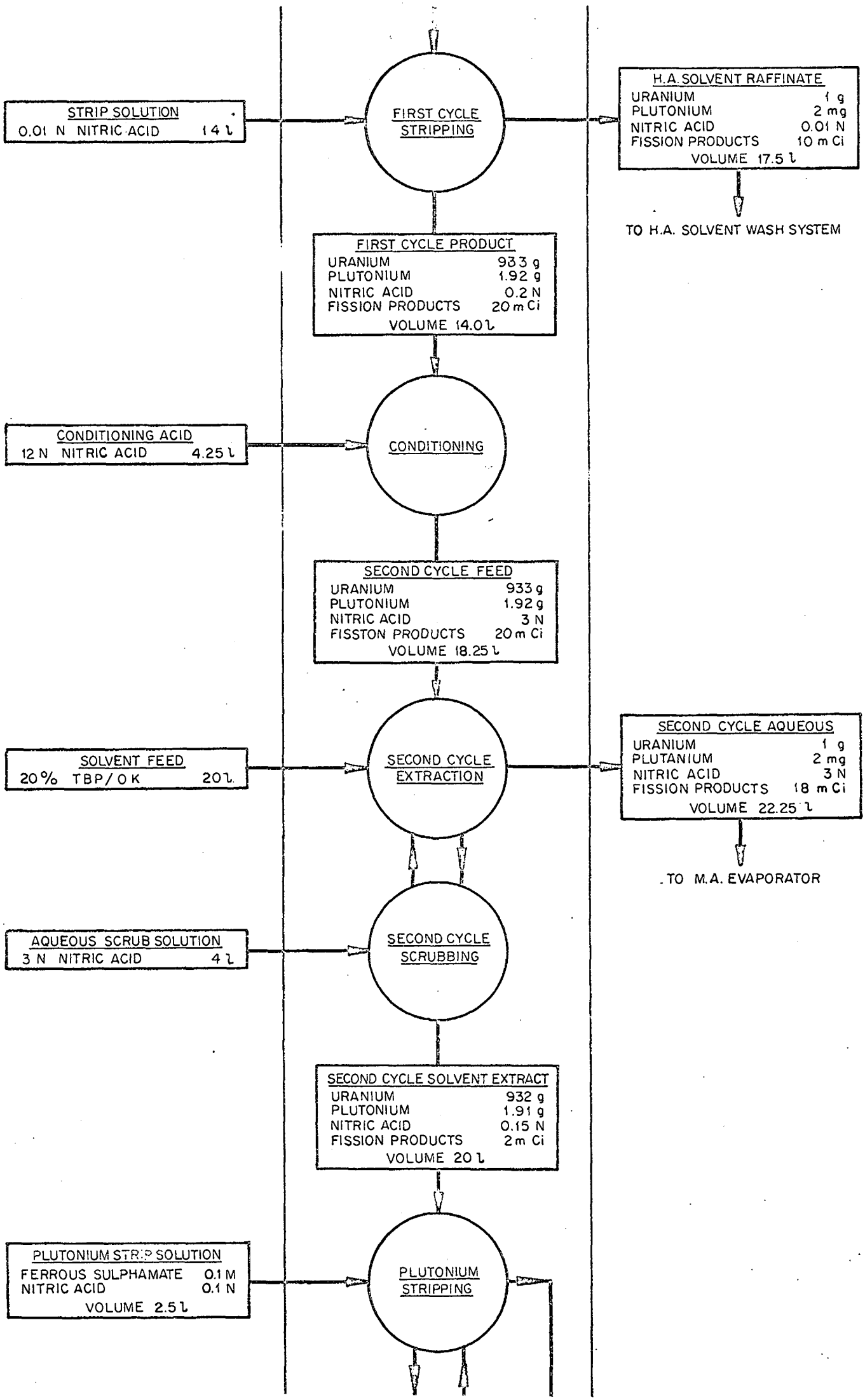
14.- DISTRIBUTION

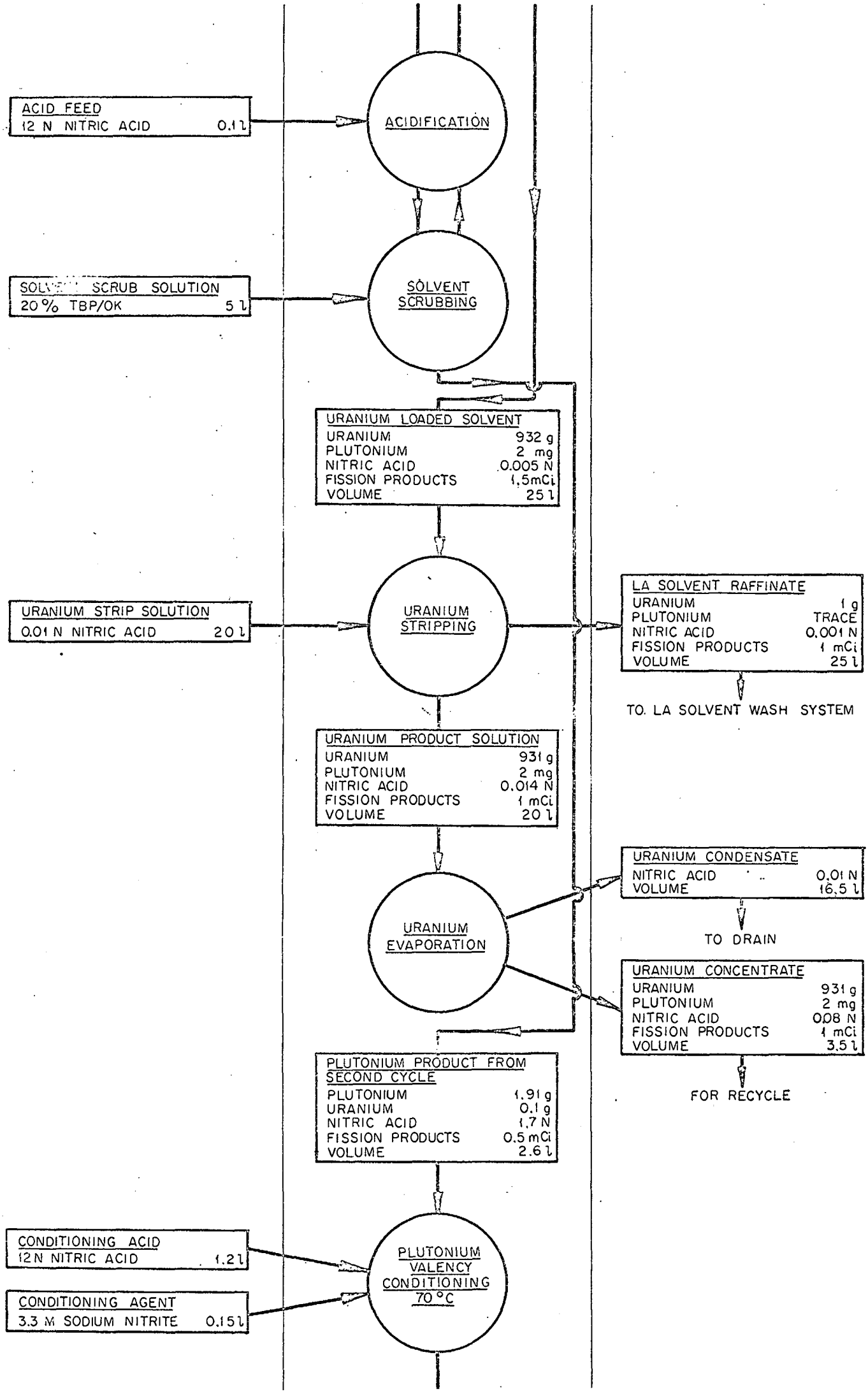
1. The President of the C.N.E.A.
2. Dr. J.A. Cosentino
3. Dr. J. Flegenheimer
4. Dr. F. Kaufmann (2 copies)
5. Dr. A.J. Carrea
6. Dr. C. Aráoz
7. Dr. J.O. Marticorena
8. Dr. D. Beninson
9. Dr. M.A. Geiger
10. Mr. F.A. Medina, Programme Director, I.A.E.A., Vienna
11. Mr. W.D. Jamrack

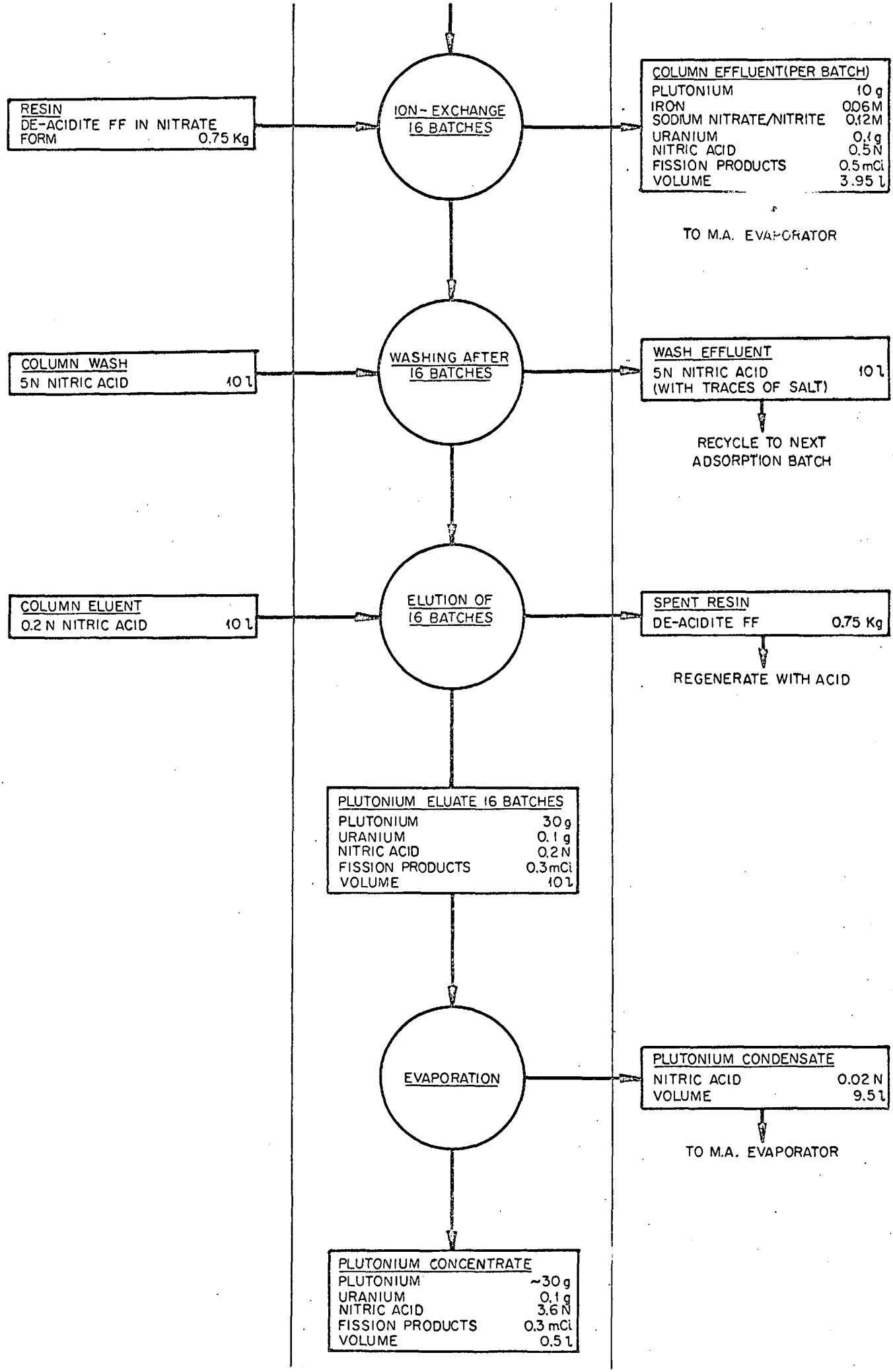
FLOW-SHEET FOR THE REPROCESSING OF THE RA I. GRAPHITE TYPE FUEL ELEMENTS




DISSOLVING ACID
12 N NITRIC ACID 8 l
MERCURIC NITRATE TRACE
OXYGEN 0.2 m³
DISSOLVING
110°C
WASTE FUMES
TRACES OF NITROGEN, OXYGEN, KRYPTON AND XENON
FILTER - AID
DIATOMACEOUS EARTH 50 g
WATER 2 l
WASH LIQUOR
0.02 N NITRIC ACID 5 l
FILTRATION AND WASHING
FILTER - AID WATER 2 l
WASH LIQUOR
0.02 N NITRIC ACID 5 l
RE - SUSPENSION
FILTER-AID
DIATOMACEOUS EARTH 50 g
WATER 2 l
WASH LIQUOR
0.02 N NITRIC ACID 5 l
FILTRATION AND WASHING
FILTER-AID WATER 2 l
FILTER CAKE
CARBON 443 g
URANIUM 20 g
PLUTONIUM 40 mg
ALUMINIUM 0.1 g
SILICA 100 g
Y FISSION PRODUCTS 1 Ci
VOLUME 0.5 l
TO SOLID WASTE STORAGE
FEED SOLUTION
URANIUM 935 g
PLUTONIUM 1.92 g
ALUMINIUM < 81 g
NITRIC ACID 3 N
Y FISSION PRODUCTS 38 Ci
VOLUME 27 l
SOLVENT FEED
20% TBP/ OK 17.5 l
FIRST CYCLE EXTRACTION
FIRST CYCLE AQUEOUS RAFFINATE
URANIUM 1 g
PLUTONIUM 2 mg
ALUMINIUM < 81 g
NITRIC ACID 3 N
Y FISSION PRODUCTS 38 Ci
VOLUME 30 l
TO M.A. EVAPORATOR
AQUEOUS SCRUB SOLUTION
3 N NITRIC ACID 3 l
FIRST CYCLE SCRUBBING
FIRST CYCLE SOLVENT EXTRACT
URANIUM 934 g
PLUTONIUM 1.92 g
NITRIC ACID 0.15 N
Y FISSION PRODUCTS 30 mCi
VOLUME 17.5 l


STRIP SOLUTION
0.01 N NITRIC ACID 14 l
FIRST CYCLE STRIPPING
H.A. SOLVENT RAFFINATE
URANIUM 1 g
PLUTONIUM 2 mg
NITRIC ACID 0.01 N
FISSION PRODUCTS 10 m Ci
VOLUME 17.5 l
TO H.A. SOLVENT WASH SYSTEM
FIRST CYCLE PRODUCT
URANIUM 933 g
PLUTONIUM 1.92 g
NITRIC ACID 0.2 N
FISSION PRODUCTS 20 m Ci
VOLUME 14.0 l
CONDITIONING ACID
12 N NITRIC ACID 4.25 l
CONDITIONING
SECOND CYCLE FEED
URANIUM 933 g
PLUTONIUM 1.92 g
NITRIC ACID 3 N
FISSTON PRODUCTS 20 m Ci
VOLUME 18.25 l
SOLVENT FEED
20% TBP/OK 20 l
SECOND CYCLE EXTRACTION
SECOND CYCLE AQUEOUS
URANIUM 1 g
PLUTANIUM 2 mg
NITRIC ACID 3 N
FISSION PRODUCTS 18 m Ci
VOLUME 22.25 l
TO M.A. EVAPORATOR
AQUEOUS SCRUB SOLUTION
3 N NITRIC ACID 4 l
SECOND CYCLE SCRUBBING
SECOND CYCLE SOLVENT EXTRACT
URANIUM 932 g
PLUTONIUM 1.91 g
NITRIC ACID 0.15 N
FISSION PRODUCTS 2 m Ci
VOLUME 20 l
PLUTONIUM STRIP SOLUTION
FERROUS SULPHAMATE 0.1 M
NITRIC ACID 0.1 N
VOLUME 2.5 l
PLUTONIUM STRIPPING


ACID FEED
12 N NITRIC ACID 0.1 l
ACIDIFICATION
SOLVENT SCRUB SOLUTION
20 % TBP/OK 5 l
SOLVENT SCRUBBING
URANIUM LOADED SOLVENT
URANIUM 932 g
PLUTONIUM 2 mg
NITRIC ACID 0.005 N
FISSION PRODUCTS 1,5mCi
VOLUME 25 l
URANIUM STRIP SOLUTION
0.01 N NITRIC ACID 20 l
URANIUM STRIPPING
LA SOLVENT RAFFINATE
URANIUM 1 g
PLUTONIUM TRACE
NITRIC ACID 0.001 N
FISSION PRODUCTS 1 mCi
VOLUME 25 l
TO. LA SOLVENT WASH SYSTEM
URANIUM PRODUCT SOLUTION
URANIUM 931 g
PLUTONIUM 2 mg
NITRIC ACID 0.014 N
FISSION PRODUCTS 1 mCi
VOLUME 20 l
URANIUM CONDENSATE
NITRIC ACID 0,01 N
VOLUME 16,5 l
TO DRAIN
URANIUM EVAPORATION
URANIUM CONCENTRATE
URANIUM 931 g
PLUTONIUM 2 mg
NITRIC ACID 0,08 N
FISSION PRODUCTS 1 mCi
VOLUME 3.5 l
FOR RECYCLE
PLUTONIUM PRODUCT FROM SECOND CYCLE
PLUTONIUM 1.91 g
URANIUM 0.1 g
NITRIC ACID 1,7 N
FISSION PRODUCTS 0.5 mCi
VOLUME 2.6 l
CONDITIONING ACID
12 N NITRIC ACID 1.2 l
CONDITIONING AGENT
3.3 M SODIUM NITRITE 0.15 l
PLUTONIUM VALENCY CONDITIONING 70 °C



FLOW-SHEET FOR THE REPROCESSING OF THE RA I GRAPHITE TYPE FUEL ELEMENTS