AUG 5 1980

SGML:RLJ
70-687

NOTE TO: J. Partlow

FROM: R. Jackson

THRU: R. Brightsen

SUBJECT: EVALUATION OF UNION CARBIDE - TUXEDO'S REQUEST TO PERMIT URANIUM WASTE FORM PROCESS

On July 7, 1980, the licensee submitted a revised Fundamental Nuclear Material Control Plan for the purpose of combining the MC&A procedures required by 10 CFR 70.51 and 70.58. Included in the revision are several changes the licensee considers as in accordance with 10 CFR 70.32c.

The primary purpose of the submittal was to seek approval for their new highly enriched uranium waste form processing operation as a part of the existing waste handling operations. The uranium waste form process is a conversion of the uranium and other fission products in the routine waste solution from a sulfate to an oxide form which is compatible with the Savannah River uranium reprocessing operation which is where UC's recovered oxide will be shipped. The licensee considers their procedure to be a significant step in relieving the U.S.'s commercial radioactive waste disposal problem.

On December 28, 1979 and April 2, 1980, the licensee submitted to the NRC's Uranium Fuel Licensing Branch a "License Amendment Request to Permit Uranium Recovery from Waste." MCL received a copy of the December 28, 1979 request on February 20, 1980 and on February 22, 1980 issued to the Uranium Fuel Licensing Branch an "Action B." Another "Action B" was issued by MCL regarding the April 2, 1980 request on April 22, 1980.

During a visit to Union Carbide - Tuxedo on March 7, 1980, R. Brightsen and myself discussed the licensee's proposed recovery process, describing in detail the necessary revisions needed in their FNMC Plan before the operation could begin; on April 3, 1980, a rough draft copy of the FNMC Plan was received. Review of the Plan and subsequent telephone conversations with J. McGovern, UC-Tuxedo SNM Accountability Representative indicated a meeting was necessary to discuss our review findings and to enable the licensee to make changes to the Plan for our final approval. The meeting was held in our office on May 15, 1980 with H. Bartz and myself and UC's J. McGovern, Fred Morse and Marcus Voth in attendance. Our suggested revisions to the FNMC Plan were discussed and agreed to by the UC personnel. It was again stressed that the final revised Plan must be reviewed and approved prior to the start-up of the Uranium Waste Recovery Process. Mr. McGovern stated he would have the revised Plan back to us the week of May 19, 1980.

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On July 2, 1980, MCL received a copy of a letter dated June 27, 1980, from L. Rouse, Chief, NRC Advanced Fuel and Spent Fuel Licensing Branch to UC-Tuxedo informing them that, based on their June 2, 1980 submittal (which revised their 12/18/79 and 4/2/80 amendment requests), the procedures to initiate the Uranium Waste Recovery Process had been approved and they could proceed on a fullscale basis. MCL did not receive a copy of UC's June 2, 1980 submittal. Another "Action B" would have been assigned since, as of June 27, 1980, we had not received UC's final revised FNMC Plan.

Upon receipt of L. Rouse's June 27, 1980, letter UC immediately started up their Uranium Waste Recovery Process. On July 3, 1980, I called J. McGovern and told him not to proceed further, other than completing the then current 200g recovery run, until he had submitted and we had reviewed and approved his yet unreceived revised FNMC Plan. In the late P.M. on July 8, 1980, the Plan was received.

In order to enable UC to meet a critical shipping schedule to Savannah River in mid-September 1980 and giving UC the benefit of a doubt over a possible misunderstanding concerning the submittal review and aproval of their FNMC Plan prior to start-up of the recovery process, we agreed to review that portion of the Plan relating to the Uranium Waste Recovery Process and relay our review findings to them via telephone on July 10, 1980.

Figure A gives a brief overview of UC-Tuxedo's SNM operations while Figure B depicts the facility SNM flow and key measurement points. Figure B, as well as UC's revised FNMC Chapters 3, 4, and 7 were the primary sources for our review. Figure C and D provide descriptions of UC's Uranium Waste Form Process.

Union Carbide - Tuxedo has an SNM thruput of approximately 2 kilograms of highly enriched uranium per month. The measurement procedures portrayed on Figure B are considered adequate for the typical operations conducted and for the Uranium Waste Form Process. Procedures A thru E are described in detail in Appendix F of the FNMC Plan. Procedure A is the modified Davis - Gray method for determining total uranium, Procedure B is a NDA for non-irradiated U-235, Procedure C and D is NDA for irradiated U-235 and Procedure E is NDA for irradiated U-235 in waste drums. Note that once the SNM has been placed in the Reactor, the material becomes highly irradiated and stays as such throughout the remainder of the process.

Incoming SNM, normally in the form of highly enriched UO_2 or U_3O_8, is dissolved upon receipt in nitric acid to create uranyl nitrate and is sampled within 24 hours for U and U-235 analysis using Procedures A and B. Prior to disolving, the SNM is gross weighed and Gamma scanned for U-235 content. UC's routine receipts of SNM for processing is (2) batches at 300 grams each approximately every (2) weeks. Once the material has become irradiated, it becomes extremely unattractive SNM for possible theft and diversion.

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UC will make measurements of acetate prior to conversion (Procedures A and B) to the uranium oxide state and will assume that, since their process is a closed system, the uranium content of the oxide powder is exactly the same as the acetate - they will ship at this element and isotope value. The gross weight of the shipped SNM is based on scale value and the known tare container weight deducted to determine the Net Uranium oxide.

Upon receipt of UC's material, Savannah River will give an interim credit to UC at as - shipped values. The shipper's values are subject to modification after Savannah River reprocessing determinations are final at which time, if necessary, UC will adjust their records. The reprocessing of the SNM may take up to a year after receipt of the material. UC plans on shipping (3) 40.0Kg batches the first year with each batch having a ratio of 10-1 (Al-U) - approximately 4 Kgs Net SNM per batch.

As a result of discussions held with you, R. Brightsen, H. Bartz and myself on July 10, 1980, approval was given by yourself and R. Brightsen to notify UC-Tuxedo's J. McGovern per telephone that SGML had approved of their proceeding with the Uranium Waste Form Process. The attached letter formalizes that decision.

The projected schedule to complete the remaining review of UC's revised FNMC Plan (not related to the recovery process) is August 8, 1980.

Raymond Jackson
Commercial Fuel Facilities Section

Attachments:
As stated

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FIGURE A

UNION CARBIDE - TUXEDO

SNM OPERATIONS DESCRIPTION

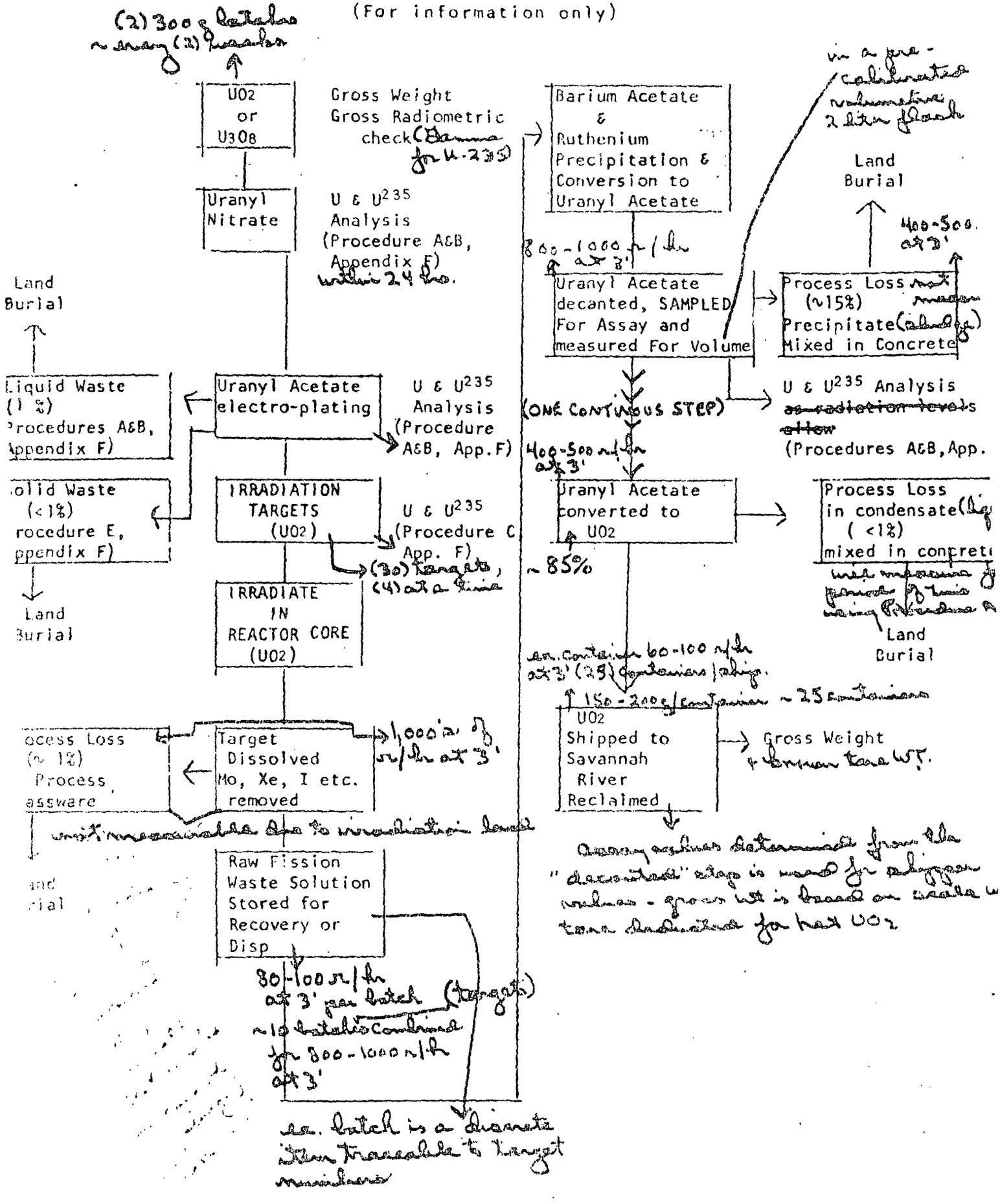
Highly-enriched uranium oxide (UO_2), received from an SNM supplier, is dissolved in nitric acid to form a uranyl nitrate solution. The uranium is then plated out of solution in a layer on the inside of steel tubes. The amount of SNM plated in each tube is determined by radiometric assay. The tubes are welded closed. After quality control measurements, the sealed tubes are irradiated in a nuclear reactor. Irradiated tubes are transferred to shielded hot cells where they are opened, the uranium is dissolved, and the desired products, (radio-pharmaceuticals), are chemically spearated. After a minimum 1 month decay the uranium solution, still highly radioactive, is either packaged for shipment to a licensed burial ground or the fission products, mainly MO-99, are stripped-out from the uranyl sulfate solution to a dry UO_2 powder for shipment to Savannah River for further purification and eventual return to production.

The flow diagram (Figure "C") shows the sequence of operations, the waste streams, and the locations in the process where SNM logs are maintained [Material Balance Areas (MBA)].

UNION CARBIDE - TUXEDO

URANIUM FLOW CHART

(For information only)



APPENDIX A

DESCRIPTION OF URANIUM WASTE FORM PROCESS

Fission product Mo-99 is produced by irradiating targets containing fully enriched uranium. Waste solutions, generated by this procedure, consist of uranium and mixed fission products dissolved in a dilute (~2 N) sulfuric acid containing ~2% nitric acid. The gaseous fission products (Xe, Kr, I_2) are removed during the fission product Mo-99 process. Each process creates approximately 150 ml of such waste, which is then stored in a borosilicate glass bottle and labeled with a sequential process number, for further disposition. Since only a small portion (~1%) of the U-235 present in each target tube fissions during irradiation, each of these bottles may be assumed to contain essentially the same amount of U-235 as was originally present in the targets. The basic process steps are listed below and shown in Figure A-4. ~~Figures A-1 through A-3 show the processing equipment dimensions and arrangment inside the hot cell, along with the minimum unit cell boundary defined by the proposed license conditions; however, the license conditions are governing regarding these matters.~~ Process steps include:

1. Combine the contents of borosilicate waste bottles containing up to 100 g U-235 in Container A. Repeat for Container B.

2. Precipitate ruthenium from the raw fission waste solution as a sulfide at 90°C by the use of thioacetamide added to Container A and B. This adds a volume of approximately 25 ml to the 750 ml of waste solution in each flask. (This is a precautionary step; its need will be evaluated during the startup program. If it is found unnecessary it will be eliminated.)

3. Precipitate the sulfates from the raw fission waste solution at 90°C by the use of barium acetate solution added to Containers A and B. This adds a volume of approximately 650 ml to the 775 ml of waste solution in each flask. (Steps 2 and 3 may be reversed or done simultaneously.)

4. Decant and filter the solution to remove the $BaSO_4$ precipitate (Container A to C or B to C).

5. Measure the filtrate volume and take a sample for assay (Container C).

6. Transfer the solution to an aluminum can placed in an electrically heated furnace and heat to dryness. The vacuum differential distillation is run at approximately 50°C. At the end of the vacuum drying step, the temperature of the dry Uranyl Acetate will be approximately 280°C. (Heating done in Container D with distillate passing to containers E and F.)

7. Continue to raise the temperature of the dry power to calcine the uranium to a final temperature of 320°C. During this stage, the imput and output of the aluminum can (Container D) are vented through glass wool particulate traps to the hot cell atmosphere.

8. Weigh the aluminum container to determine the net weight of mixed uranium oxides content (Container D).

9. Seal the aluminum container and store for subsequent shipment to the reprocessing facility (Container D).

10. Dispose of the precipitate sludge (Container A and B) and the distillate (Container F) by solidifying in concrete for burial.

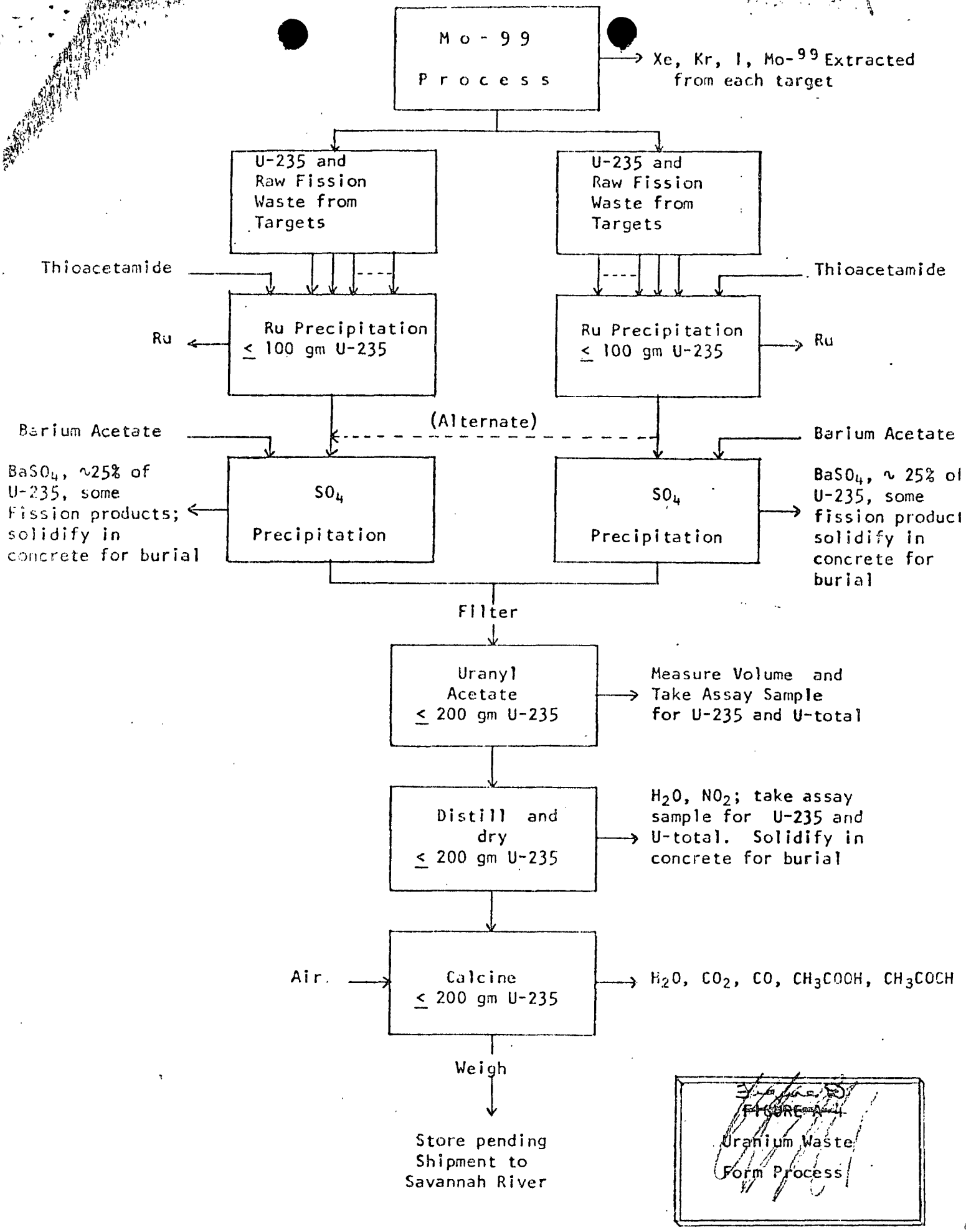


FIGURE A-1
Uranium Waste Form Process