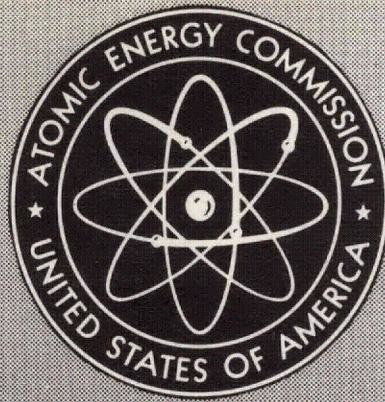


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## FUNDAMENTAL AND APPLIED RESEARCH AND DEVELOPMENT IN METALLURGY. CONCEPTUAL DESIGN OF A U-233 FUEL REFABRICATION PLANT

Summary Technical Report to the United States Atomic Energy  
Commission for the Period July 1, 1960 through June 30, 1961

By  
W. B. Nowak  
B. W. Wessling

July 30, 1962

Nuclear Metals, Inc.  
Concord, Massachusetts

metadc502682

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Nuclear Metals, Inc., Concord, Massachusetts

Prime Contractor

  
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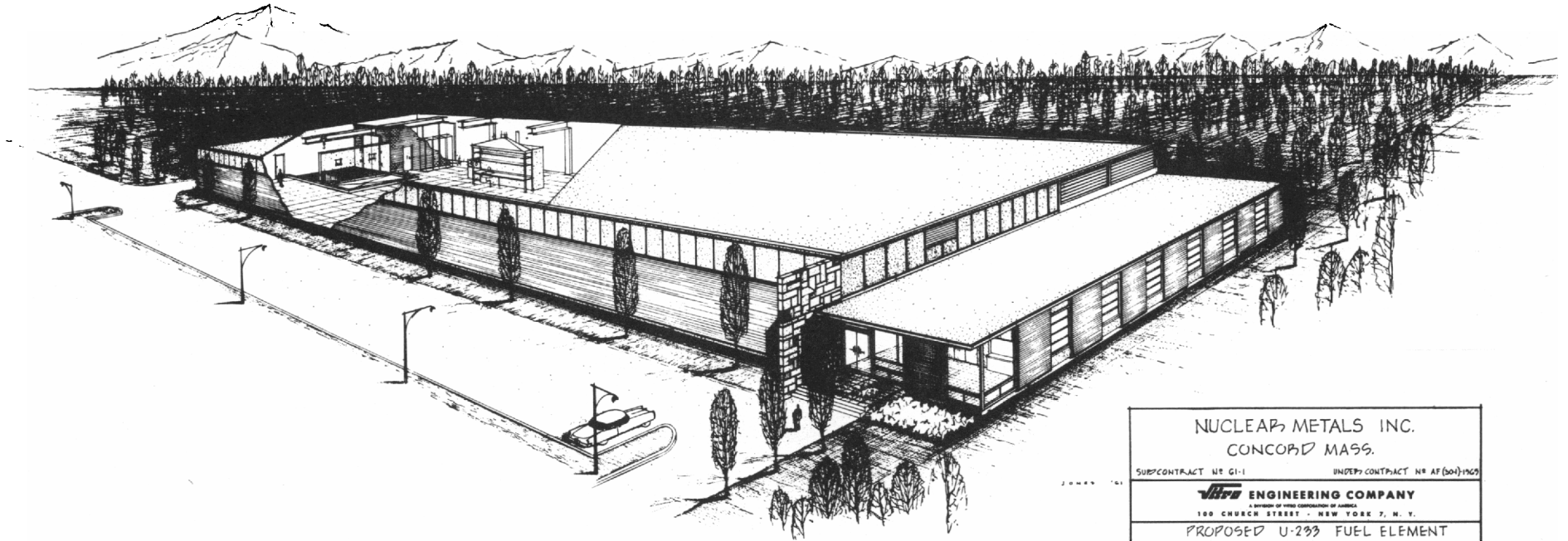
July 30, 1962

Nuclear Metals, Inc.  
Concord, Massachusetts

Contract No. AT(30-1)-1565

A. R. Kaufmann  
Technical Director





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### ABSTRACT

A preliminary study has been made of a direct access, streamlined process for the rapid fabrication of (4-7)%  $\text{UO}_2$ - $\text{ThO}_2$  fuel using  $\text{U}^{233}$ . The fabrication of nuclear fuel elements containing  $\text{U}^{233}$  is complicated by the high alpha activity and very energetic gamma rays arising from the unavoidable presence of small quantities of  $\text{U}^{232}$ . Analysis of the fuel expected from the Consolidated Edison Indian Point Reactor indicates an average contamination of about 400 ppm  $\text{U}^{232}$  in U for material cycled three times and about 200 ppm for material cycled once. Feasibility for most of the major steps of a streamlined process has been demonstrated experimentally. The process is: (1) separation of  $\text{U}^{232}$  daughters by solvent extraction, (2) coprecipitation of ADU and  $\text{Th}(\text{OH})_4$ , (3) filtration, (4) calcination, reduction, and simultaneous ball-milling to produce powder, (5) cold compaction of pellets, (6) sintering of pellets, (7) grinding, (8) subassembly, and (9) assembly of fuel element. The process is adaptable to fabrication by vibratory compaction or swaging. About 8 days are required to fabricate three complete fuel elements for the Consolidated Edison reactor. A detailed engineering evaluation of the process has been made, a plant layout developed, and equipment specified. Calculated estimates of the radiation levels expected from contamination of glove boxes and equipment and from the fuel at various stages in this process indicate that only minimal shielding and inexpensive manual equipment is required, permitting direct, rather than remote, access to the fuel.

## SUMMARY

As is well known, the natural decay chain for  $U^{232}$  results in high alpha-activity and in some very energetic gamma-emitting daughters both of which present a serious health hazard in the refabrication of fuel from the thorium fuel cycle. There are two general approaches for the solution of this problem. The first approach is to conduct the refabrication in a fully shielded, remotely operated and maintained facility. The second approach, the subject of this year-long feasibility study at Nuclear Metals, consists of decontaminating the fuel material by nearly complete removal of the  $U^{232}$  daughter products, followed by rapid fabrication of a ceramic fuel element using streamlined process techniques in gloveboxes.

Estimates have been made of the variation in  $U^{232}$  content of the various fuels which will be produced over the next several years from reactors using thorium as a fertile material. These estimates include the first three cycles for both the Consolidated Edison and Elk River Reactors; for the next decade, the Consolidated Edison reactor may provide the bulk of available  $U^{233}$  for recycling. The  $U^{232}$  content of the Consolidated Edison fuel will vary from less than 100 ppm to almost 300 ppm with an average of 200 ppm for the first cycle and on an increasing scale for subsequent cycles, reaching an average of about 400 ppm for material cycled three times. It should also be noted that second cycle  $U^{233}$ -bearing fuel from Consolidated Edison will probably not be available for refabrication until about 1968 and third cycle material about 1972. This assumption is based on a core life of 2 years, the second core of  $U^{235}$ , and start-up in 1962. Therefore, between now and the year 1972, the bulk of the fuel resulting from the reactors presently being built to operate on the thorium fuel cycle will contain only low levels of  $U^{232}$ . Because of the variations in uranium concentration and in reactor position for different elements, the elements will contain different amounts of  $U^{232}$ , and it is not necessarily desirable to mix the irradiated fuels to provide for uniform fuel activity. Obviously, if the more highly contaminated fuels are not recycled, the level of activity which must be accommodated is decreased.

Based on the present work, it appears that, at most, only minimal shielding and, in general, rather inexpensive manual equipment with direct rather than remote access to the fuel is all that would be required to refabricate this  $U^{232}$  fuel material, providing of course that all of the daughter products are removed and a sufficiently streamlined fabrication technique is used.

Although a fully shielded remote facility may be required for the more heavily contaminated fuel which will be available after 1974-1975, a manual access plant should be capable of providing for the refabrication of the bulk of the fuel produced in the first few cycles. The heavily contaminated fraction of the fuel produced in the first few cycles could be stored until large quantities of heavily contaminated fuel from continued recycle become available for remote fabrication. In addition to the economic advantages of a manual access plant, many years' worth of valuable experience would be gained in handling large quantities of  $U^{232}$ -bearing fuel by streamlined process techniques. This experience would be invaluable in designing an economical remote facility and might even result in cost reductions for conventional fabrication. Additional thorium cycle reactors may start up, providing additional feed over longer periods.

The direct-access approach to the  $U^{232}$  problem utilizes a streamlined procedure for the preparation of a (4-7) %  $UO_2$ - $ThO_2$  powder material that could be used as feed for any of four fuel element fabrication techniques: (1) hot extrusion, (2) pellets in tubes, (3) vibratory compacting, (4) swaging. Of these, only the pellets-in-tube route has had sufficient irradiation experience to demonstrate its reliability as a power reactor fuel. The other three techniques, although certainly desirable because of the streamlining possibility they offer, have not yet been demonstrated to be reliable power reactor fuels. Consideration was given to several methods for powder and pellet preparation. These included: (1) the ANL  $U_3O_8$ - $ThO_2$  pellet process, (2) the ORNL sol-gel technique, (3) the Sylcor

oxalate coprecipitation  $\text{UO}_2\text{-ThO}_2$  process, (4) a  $\text{UO}_2\text{-ThO}_2$  mixture process, and (5) a coprecipitation  $\text{UO}_2\text{-ThO}_2$  process. The last has been selected because: (1) early mixing of uranium and thorium eases problems of radiation and criticality, and (2) material transfers and powder handling are minimized, thereby expediting the process and reducing decontamination problems. This streamlined process as developed in this work appears to be ideally suited for the pellet-in-tube and hot coextruded fuel elements. An additional process step to transform the powdered material to particles of high density material would have to be devised to make the fuel suitable for elements fabricated by swaging or vibratory compacting.

The approach chosen is felt to be desirable since it does not require any changes in present reactor fuel specifications, yet it does allow for cheaper and more rapid fabrication by the new techniques once their reliability has been proven.

The Consolidated Edison reactor fuel element was selected as the prototype for the output of the refabrication plant, since this reactor will require the largest quantities of fuel and has the closest tolerated element of the thorium reactors.

The use of fresh  $\text{U}^{235}$  for part of the core loading has been given consideration. In order to replace the fuel that has undergone burn-up, some  $\text{U}^{235}$  will have to be added to the  $\text{U}^{233}$ -bearing fuel from the reactor. In addition, since the rod loading patterns for Consolidated Edison fuel elements are very complex, some elements require the processing of separate small batches of  $\text{UO}_2\text{-ThO}_2$  at special concentrations. It appears feasible to use  $\text{U}^{235}$  for these small batches. A fuel allocation has been devised in which only one kind of fuel rod (i.e. fuel at one concentration) for each type of fuel element need be fabricated from  $\text{U}^{233}$ -bearing fuel. In this allocation 31.6 percent of the rods in the new core loading are prefabricated with  $\text{U}^{235}$ . The scheduling of many batches of  $\text{U}^{233}$ -bearing fuel at six different concentrations, and the concomitant problems, are therefore eliminated.



In the design conceived in this program, fuel is received from the thorium and fission product removal facility and, as a first step, is decontaminated in a fully shielded cell by solvent extraction from a chloride solution using 20 percent Alamine 336 in xylene as the extractant, and 0.1 M  $\text{NH}_4\text{Cl}$  as the stripping agent.  $\text{Th}^{228}$  and other daughter products go through in the raffinate. The clean uranium solution, after leaving the shielded cell, is added to a thorium nitrate solution from which a mixture of ADU and  $\text{Th}(\text{OH})_4$  is coprecipitated by  $\text{NH}_4\text{OH}$ , washed and filtered.

Fabrication by vibratory compaction, swaging, hot coextrusion, and pelletizing has been considered. Since the first three are not yet well established, and since pelletizing is the most complex and time consuming, it was chosen as the basis for estimating feasibility of the direct access refabrication plant. After the  $\text{UO}_2$ - $\text{ThO}_2$  mixture is precipitated and filtered, it is simultaneously calcined, ball-milled, and reduced in a special rotary kiln. The material is then granulated three times, mixed with a lubricant, and pressed into pellets. After sintering, the pellets are ground, inspected, loaded into tubes and the end caps welded in place. The tubes are then inspected and leak-tested. If acceptable, they are briefly stored until assembled into a complete fuel element. The entire process of refabrication (starting after decontamination and ending with final inspection) is estimated to require about 8 days for three complete fuel assemblies.

The technical feasibility of the above process was demonstrated by laboratory experiments. The flow rates, number of necessary stages, and decontamination efficiency were determined for the solvent extraction circuit by shakeout tests. Coprecipitation, filtration, and direct hydrogen reduction of ADU and  $\text{Th}(\text{OH})_4$  to a solid solution of  $\text{UO}_2$ - $\text{ThO}_2$  were also shown to be technically feasible. The hydrogen reduction was carried out in a rotary kiln while simultaneously ball milling the material. Preliminary experiments indicated that the resulting powder can be successfully pelletized using a few percent of water instead of a lubricant and binder. This implies that metering of the powder into the cavity be accomplished on a weight basis. Several sintering experiments yielded pellet densities from 93 to 96 percent.

For fuel containing as much as 400 ppm of  $U^{232}$ , the gamma activity attributable to  $U^{232}$  daughters does not equal that due to natural thorium within the fabrication period estimated.

A detailed engineering evaluation of the process was also carried out from the viewpoints of chemical engineering, procurement of equipment, and general plant design. A plant layout was developed; cost and specifications of nearly all equipment were ascertained. It is estimated that plant and equipment can be constructed for about \$1,250,000 exclusive of site acquisition and certain test equipment.

I. INTRODUCTION

Within the next few years, two reactors in the United States will be releasing for recycle fuel which will contain substantial percentages of  $U^{235}$  and of  $U^{233}$  converted from thorium. This fuel will be contaminated with significant amounts of  $U^{232}$ . The pressurized water reactor of Consolidated Edison at Indian Point, New York, will be releasing  $U^{232}$ -bearing fuel at the approximate rate of 840 kgs every 600 days. The boiling water reactor of the Rural Cooperative Power Association at Elk River, Minnesota, will be releasing about 70 kgs per year of fuel contaminated to a lesser degree. In addition, in several years, a third reactor is expected to go on-stream at Peach Bottom, Pennsylvania, which will release moderate quantities of slightly contaminated fuel. Since other reactor groups are giving serious consideration to the  $U^{233}$ -thorium fuel cycle, even greater quantities of contaminated fuel may be produced in the future.

At the present time, no capability exists for refabricating this fuel into useful power reactor fuel elements. If we assume that the fuel is worth \$13 per gram, we find that these three reactors will be releasing about twelve million dollars worth of fuel every two years. Because of the  $U^{232}$  contamination, this fuel must be uselessly stockpiled until refabrication capability becomes available.

Another impelling reason for developing a recycle technology is the need for establishing a market for the synthetic fuels,  $U^{233}$  and plutonium. It is reasonable to expect that when a recycle technology is demonstrated, the values of both  $U^{233}$  and plutonium relative to the values of  $U^{235}$  will be determined by ordinary economic factors, as a function of the relative demands for these fuels by the various utilities. This of course would obviate the necessity for establishing "fair prices".\* In addition, in

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\* Establishing fair prices for  $U^{233}$  and plutonium is complicated by the fact that fuel values relative to  $U^{235}$  vary from one reactor to another, making it necessary to estimate the future supply and demand situation for fuels in the entire nuclear power industry.

order to establish fuel values and long-term fuel cycle economics, it is necessary to estimate and predict the fuel fabrication cost for synthetic fuels. In the absence of a recycle technology, such estimates can be little more than guesses.

Some work has been performed in the development of fabrication techniques for the recycle of low g/T plutonium. It is presumed that additional work to solve the problems of handling higher g/T material will follow. Similarly, additional work is needed for the recycle of  $U^{233}$  fuel. In the refabrication of fuel from the thorium fuel cycle, a serious health hazard is encountered because of the high alpha activity and energetic gamma-emitting daughters which result from the natural decay chain for  $U^{232}$ . One method of handling this problem is to conduct the refabrication in a fully shielded, remotely operated and maintained facility. This approach has been studied at the Oak Ridge National Laboratories and is also the basis of a plant designed by the Allis Chalmers Manufacturing Company for the Comitato Nazionale Per L'Energia Nucleare (CNEN), the PCUT Facility. A second approach is the subject of this report, which is the result of a year-long feasibility study at Nuclear Metals, Inc. This approach consists of decontaminating the fuel material by removal of the  $U^{232}$  daughter products, followed by rapid refabrication of the fuel element using streamlined process techniques in direct-access glove boxes.

As part of the definition of the problem, estimates were made of the  $U^{232}$  content of the various fuels which will be produced over the next several years from reactors presently using thorium as a fertile material. The  $U^{232}$  content of the fuel was found to vary from less than 100 ppm (based on the total uranium) to about 400 ppm, with an average of about 200 ppm for the first cycle material. For continued recycle of the material, an average  $U^{232}$  content of approximately 400 ppm will be attained for material cycled three times. Due to the variations in fuel element tolerances and in fuel element concentrations for different elements, it is not necessarily desirable to mix the fuels to provide for uniform fuel activity. Obviously, if the more highly contaminated fuels are not recycled, the level of activity which must be accommodated is decreased in two ways: by increasing the dilution factor and by using less contaminated fuel to start.

It should be noted that the first  $U^{233}$  fuel which will result from the Con Ed Indian Point Reactor will probably not be available for refabrication until about 1965. Fuel material which has been cycled through the reactor three times therefore would probably not be available until 1972-1973. These estimates are based on a core life of two years with start-up in 1962 and a second core loading of  $U^{235}$ . It is therefore fair to say that between now and the year 1970 the bulk of the fuel resulting from reactors presently being built to operate on the thorium fuel cycle will contain only low levels of  $U^{232}$ . For these reasons, this work has investigated a refabrication plan which would, at most, require only minimal shielding and, in general, rather inexpensive manual-access equipment with direct rather than remote access to the fuel, providing of course that all of the daughter products are previously removed and a sufficiently streamlined fabrication technique is used.

The major objective of this program is, therefore, a conceptual design of a manual-access fuel refabrication plant capable of handling  $U^{233}$ . In order to give some direction to the program, the Consolidated Edison Indian Point Reactor was chosen as a reference, and the majority of the plant's production was geared to this reactor in terms of both reactor requirements and reactor output. This choice was not difficult since the Indian Point reactor will provide the largest output of  $U^{233}$  in the foreseeable future, the fuel will be the most heavily contaminated with  $U^{232}$ , and the fuel element specifications are sufficiently rigid to provide a conservative basis for investigation of the feasibility of rapid direct-access refabrication.

Among the ceramic techniques for the fabrication of nuclear fuel elements the most attractive for application to the  $U^{233}$  problem are: (1) hot extrusion, (2) pellets in tubes, (3) vibratory compaction, (4) swaging. Of these possibilities, only the technique of pellets-in-tubes has been demonstrated, by considerable irradiation experience, to be reliable as a power reactor fuel. The other three techniques are very desirable because of the streamlining possibilities they offer. However, it was felt that it would be premature, and not sufficiently conservative, to incorporate

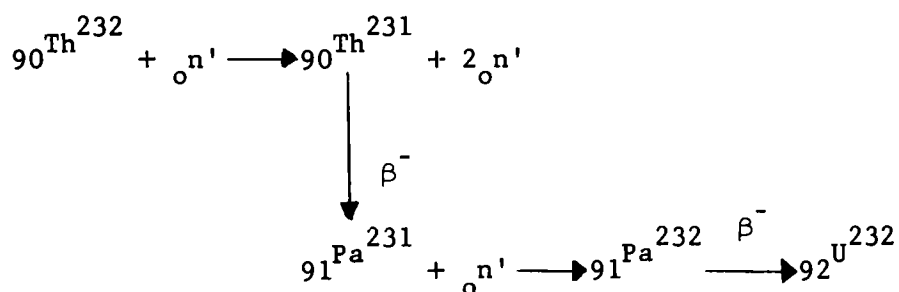
them in the present study since they have not yet been demonstrated to be reliable power reactor fuel elements. Rather, our approach has been to devise a streamlined technique for the preparation of a mixed  $\text{UO}_2\text{-ThO}_2$  powder material which could be used as a feed for pelletizing. This material could also be used as a feed for coextruded fuel elements. An additional process step to transform this powder to a high particle density material (comparable to the Spencer Chemical Company's proprietary arc-fused ceramic fuel) would be necessary to make this ceramic fuel amenable to swaging or vibratory compacting. The route of pellets-in-tubes is the longest (time-wise) of the possibilities; if this route is feasible, the more streamlined methods of loading fuel into tubes would certainly be readily accomplished with  $\text{U}^{233}$ -bearing fuel. Furthermore, the pellet approach does not require any changes in present reactor fuel specifications; for example, both Consolidated Edison's Indian Point Reactor and the Rural Cooperative Power Association's Elk River Reactor specify  $\text{UO}_2\text{-ThO}_2$  pellets in stainless steel tubes. If and when the newer techniques of vibratory compaction, hot extrusion, and swaging become acceptable to reactor operators and are proven to be reliable power reactor fuel materials, their incorporation into this conceptual  $\text{U}^{233}$  plant will be an easy matter.

## II. PLANT INPUT

For the purposes of this investigation it has been assumed that the fuel taken from the reactor is processed in a "hot" separation facility, where the uranium is separated from the thorium and the fission products. The resultant product is available as a uranyl solution containing principally  $\text{U}^{235}$  and  $\text{U}^{233}$  contaminated with  $\text{U}^{232}$ . Since it is the daughters of  $\text{U}^{232}$  that cause the refabrication difficulties, it is necessary to estimate the quantity of  $\text{U}^{232}$  that will be present in the fuel to be refabricated. A fairly accurate appraisal must be made, because a gross overestimation would magnify the entire problem so much that a realistic engineering evaluation would be impossible and the cumbersome and costly procedures of remote fabrication would automatically appear indicated.

A. Factors Affecting the  $U^{232}$  Content of Fuels Containing  $U^{233}$ .

The primary source of  $U^{232}$  in a thorium reactor is beta decay of  $Pa^{231}$  which is formed as a result of the neutron capture in  $Th^{231}$ .  $Pa^{231}$  is formed in a thorium reactor by the (n, 2n) reaction on  $Th^{232}$  to produce  $Th^{231}$  which subsequently decays by beta emission to  $Pa^{231}$ .



All of the half-lives involved in these reactions are very short compared to the normal fuel exposures. It is, therefore, reasonable to assume that  $U^{232}$  production is almost immediate. As  $U^{232}$  is thus being built up, it is being destroyed at the same time from neutron absorptions resulting in  $U^{233}$ . It is also being destroyed to a much smaller extent from alpha decay to produce  $Th^{232}$ , about one percent per year. Obviously, the  $U^{232}$  content depends upon irradiation time, flux, spectrum, etc.

The (n, 2n) reaction of thorium is energy dependent, having a threshold energy of 6.37 Mev. In reactors where the ratio of thermal to fission neutrons in the fertile material is fairly high, the effect of the (n, 2n) reaction is relatively minor. Most practical reactors, however, operate with very hard neutron spectra, in which case the (n, 2n) reaction can become quite significant. The most effective way to account for this is to develop an equivalent distribution of the neutron flux. Calculations, detailed in Section B below, yield effective  $Th^{232}$  (n, 2n)  $Th^{231}$  cross-sections (to be used with the thermal flux) of 0.0458 barns in the Elk River Reactor and 0.0524 barns in the Indian Point Reactor. Using these cross-sections, one can define the rates of transmutation of  $Pa^{231}$  to  $U^{232}$  and of the destruction of  $U^{232}$ , so that the  $U^{232}$  concentration can be calculated as a function of the reactor flux and of the time of

irradiation of the fuel. In making this latter calculation (Section C, following) one is limited by the lack of good cross-sections for Pa<sup>231</sup> and U<sup>232</sup>. The dependence of the build-up of U<sup>232</sup> on the Pa<sup>231</sup> cross-section is such that an increase in the effective cross-section of from 200 to 300 barns would increase the U<sup>232</sup> concentration by about 44 percent on the first cycle. This increase is further amplified if the fuel is recycled. Recent MTR measurements including the effect of the absorption resonance involved give an effective cross-section of 350 barns for Pa<sup>231</sup> in the Elk River and Indian Point Power Reactors. Large variations in U<sup>232</sup> capture cross-section are not significant in the first cycle but become more important in recycles.

#### B. Effective Th<sup>232</sup> (n, 2n) Th<sup>231</sup> Cross-Section

The threshold energy for the (n, 2n) reaction in Th<sup>232</sup> is  $6.34 \pm 0.09$  Mev.<sup>(1)</sup> Hence, when thorium is irradiated, the (n, 2n) reaction takes place only with those neutrons having energies greater than 6.34 Mev. In reactor cores moderated with water or containing large quantities of high inelastic scattering cross-section material, the neutron spectrum above 6.34 Mev will have the shape of the fission spectrum.

A measurement of the effective (n, 2n) cross-section averaged over the U<sup>235</sup> fission spectrum has been made by J. A. Phillips.<sup>(2)</sup> A value of  $12.4 \pm 0.6$  millibarn was obtained for the mean value of the (n, 2n) cross-section of Th<sup>232</sup> in a fission spectrum.

That is  $\bar{\sigma}(n, 2n) = 12.4 \pm 0.6$  millibarns

$$\text{where } \bar{\sigma}(n, 2n) = \int_{E=6.34}^{\infty} N(E)\sigma(n, 2n, E) dE \quad (1)$$

$$\int_0^{\infty} N(E) dE = 1 \quad (2)$$

where  $N(E)$  = the fission spectrum normalized to one fission neutron.

$\sigma(n, 2n, E)$  = the Th<sup>232</sup> (n, 2n) cross-section as a function of incident neutron energy E.



Since only 2.05 percent of the fission neutrons have energies greater than 6.34 Mev, the magnitude of  $\sigma(n, 2n, E)$  is approximately 0.6 barns above 6.34 Mev.

To relate the  $(n, 2n)$  reaction rate to the thermal flux in the core, the following is derived:

The number of neutrons produced per  $\text{cm}^3$  per second with energies between  $E$  and  $E + dE$  is

$$V \bar{\Sigma}_f \bar{\Phi}_{th} N(E) dE \quad (3)$$

and the number of neutrons removed from this energy interval per  $\text{cm}^3$  per sec is

$$\Sigma_R(E) \phi(E) dE \quad (4)$$

where

$V$  = neutrons produced per fission

$\Sigma_f$  = effective fission cross section ( $\text{cm}^{-1}$ )

$\bar{\Phi}_{th}$  = thermal neutron flux ( $\text{n/cm}^2\text{-sec}$ )

$\phi(E)$  = neutron flux as a function of energy ( $E$ )

$\Sigma_R(E)$  = neutron removal cross-section as a function of neutron energy

at steady state;

$$V \Sigma_f \bar{\Phi}_{th} N(E) dE = \Sigma_R(E) \phi(E) dE$$

and

$$\phi(E) = \frac{V \Sigma_f \bar{\Phi}_{th} N(E)}{\Sigma_R(E)} \quad (5)$$

The reaction rate (R) in the reactor core for the production of  $\text{Th}^{231}$  and hence  $\text{Pa}^{231}$  is

$$R = N_{02} \int_{6.35}^{\infty} \sigma(n, 2n, E) \phi(E) dE \quad (6)$$

where

02 refers to  $90\text{Th}^{232}$

If we define an effective thermal cross-section  $\hat{\sigma}$  for the (n, 2n) reaction such that

$$\hat{\sigma} \phi_{\text{th}} N_{02} = R$$

then

$$\hat{\sigma} N_{02} \phi_{\text{th}} = N_{02} \int_{6.35}^{\infty} \sigma(n, 2n, E) \phi(E) dE \quad (7)$$

Replacing  $\phi(E)$  using equation (5)

$$\hat{\sigma} \phi_{\text{th}} = \int_{6.35}^{\infty} \frac{\sigma(n, 2n, E) \mathcal{V} \Sigma_f \phi_{\text{th}} N(E) dE}{\Sigma_R(E)} \quad (8)$$

Replacing  $\Sigma_R(E)$  by an average value  $\bar{\Sigma}_R$

$$\hat{\sigma} = \frac{\mathcal{V} \Sigma_f}{\bar{\Sigma}_R} \int_{6.35}^{\infty} \sigma(n, 2n, E) N(E) dE$$

Replacing the integral by the experimental value<sup>(2)</sup>

$$\hat{\sigma} = \frac{\mathcal{V} \Sigma_f (0.0124 \text{ b})}{\bar{\Sigma}_R}$$

The macroscopic removal cross-section was calculated from the following average microscopic removal cross-sections for greater than 6.35 Mev neutrons:

<u>Material</u>	<u><math>\sigma_R</math> (barns)</u>
H	1.17
Th	3
Fe	2
Zr	2

The relationship between fission cross-section, thermal flux, and power density is

$$\Sigma_f \phi_{th} = P(c)$$

where  $P$  = power density - watts/cm<sup>3</sup>

$$c = 3.168 \times 10^{10} \text{ fissions/watt-sec}$$

Hence, if the power density and thermal flux are given for a reactor, the value of  $\Sigma_f$  may be determined.

For the Elk River Reactor the following data are available:

$$V_c \text{ (core volume)} = 2.465 \times 10^6 \text{ cm}^3$$

$$\text{Total Power} = 58.2 \text{ Mw}$$

$$\nu = 2.43 \text{ n/fission}$$

$$\phi_{th} = 1.1 \times 10^{13} \text{ n/cm}^2 \text{ sec}$$

$$\Sigma_R = 0.04475 \text{ cm}^{-1}$$

$$\begin{aligned} \Sigma_f &= \frac{P(c)}{\phi_{th}} = \frac{(58.2 \times 10^6 \text{ watts}) (3.168 \times 10^{10} \text{ fission/watt sec})}{2.465 \times 10^6 \text{ cm}^3 (1.1 \times 10^{13} \text{ n/cm}^2 \text{ sec})} \\ &= 0.0680 \text{ cm}^{-1} \end{aligned}$$

Hence:

$$\hat{\sigma}_{ERR} = \frac{\sqrt{\Sigma_f}}{\Sigma_R} (0.0124)b = \frac{2.43(0.0680)(0.0124 \text{ b.})}{0.04475} = 0.0458 \text{ b.}$$

So the effective (n, 2n) cross-section,  $\hat{\sigma}$ , to be used with the thermal flux is 0.0458 b.

For the Indian Point Reactor

$$\begin{aligned} V_c &= 6.4365 \times 10^6 \text{ cm}^3 \\ \text{Total power} &= 585 \text{ Mw} \\ \phi_{th} &= 2.16 \times 10^{13} \text{ n/cm}^2 \text{ sec} \\ \Sigma_R &= 0.0766 \text{ cm}^{-1} \end{aligned}$$

Solving for the effective (n, 2n) cross-section

$$\hat{\sigma}_{Con Ed} = 0.0524 \text{ b.}$$

### C. Build-Up of Pa<sup>231</sup> and U<sup>232</sup> in Irradiated Fuel

#### 1. Build-Up Equations for Pa<sup>231</sup>

The primary source of Pa<sup>231</sup> in thorium reactor fuel is the Th<sup>232</sup> (n, 2n) reaction to produce Th<sup>231</sup> which subsequently decays by beta emission to Pa<sup>231</sup>. Since the half-life of Th<sup>231</sup> is short relative to normal fuel exposure, it is assumed that the Pa<sup>231</sup> is produced immediately after the Th<sup>232</sup> (n, 2n) reaction.

As shown in the previous section the reaction rate for Pa<sup>231</sup> production is  $\hat{\sigma} \phi_{th} N_{02}$ . The differential equation governing its build-up is:

$$\frac{dN_{11}}{dt} = \hat{\sigma} \phi N_{02} - \lambda_{11} N_{11} - \sigma_{11} \phi N_{11} \quad (9)$$

Since the half-life of Pa<sup>231</sup> is 34,000 yr.,  $\lambda_{11} \sim 0$

$$\frac{dN_{11}}{dt} = \hat{\sigma} \phi N_{02} - \sigma_{11} \phi N_{11} \quad (10)$$

Solving

$$\ln \left[ \frac{\hat{\sigma}}{\sigma} \phi N_{02} - \sigma_{11} \phi N_{11} \right] = -\sigma_{11} \phi t + c \quad (11)$$

when

$$t = 0, N_{11} = 0$$

Therefore,

$$c = \ln \left[ \frac{\hat{\sigma}}{\sigma} N_{02} \phi \right]$$

Substituting and rearranging,

$$\frac{N_{11}}{N_{02}} = \frac{\hat{\sigma}}{\sigma_{11}} \left[ 1 - e^{-\sigma_{11} \phi t} \right] \quad (12)$$

## 2. Build-Up Equations for U<sup>232</sup>

The primary source of U<sup>232</sup> in a thorium reactor is beta decay of Pa<sup>232</sup> which is formed as a result of neutron capture in Pa<sup>231</sup>. Since the half-life of Pa<sup>232</sup> is short (1.3 days), we assume that the U<sup>232</sup> is formed immediately after a neutron capture in Pa<sup>231</sup>. U<sup>232</sup> is destroyed both by neutron capture to produce U<sup>233</sup> and by radioactive alpha decay to Th<sup>228</sup>.

The differential equation governing its build-up is:

$$\frac{dN_{22}}{dt} = \sigma_{11} N_{11} \phi - \lambda_{22} N_{22} - \sigma_{22} \phi N_{22} \quad (13)$$

Inserting equation (12),

$$\frac{dN_{22}}{dt} + (\lambda_{22} + \sigma_{22} \phi) N_{22} = \phi \frac{\hat{\sigma}}{\sigma} N_{02} \left[ 1 - e^{-\sigma_{11} \phi t} \right] \quad (14)$$

The integrating factor is  $e^{(\lambda_{22} + \sigma_{22} \phi)t}$

Solving:

$$N_{22}e^{(\lambda_{22} + \sigma_{22}\phi)t} = \hat{\Lambda} N_{02}\phi \int_0^t \left[ e^{(\lambda_{22} + \sigma_{22}\phi)t} - e^{(\lambda_{22} + \sigma_{22}\phi - \sigma_{11}\phi)t} \right] dt + c_1 \quad (15)$$

$$N_{22}e^{(\lambda_{22} + \sigma_{22}\phi)t} = \hat{\Lambda} N_{02}\phi \left[ \frac{e^{(\lambda_{22} + \sigma_{22}\phi)t}}{(\lambda_{22} + \sigma_{22}\phi)} - \frac{e^{(\lambda_{22} + \sigma_{22}\phi - \sigma_{11}\phi)t}}{(\lambda_{22} + \sigma_{22}\phi - \sigma_{11}\phi)} \right] + c_1 \quad (16)$$

at  $t = 0$ ,  $N_{22} = N_{22}^0$

Therefore,

$$N_{22}^0 = \hat{\Lambda} N_{02}\phi \left[ \frac{1}{(\lambda_{22} + \sigma_{22}\phi)} - \frac{1}{(\lambda_{22} + \sigma_{22}\phi - \sigma_{11}\phi)} \right] + c_1$$

$$c_1 = N_{22}^0 - \hat{\Lambda} N_{02}\phi \left[ \frac{1}{(\lambda_{22} + \sigma_{22}\phi)} - \frac{1}{(\lambda_{22} + \sigma_{22}\phi - \sigma_{11}\phi)} \right] \quad (17)$$

Inserting  $c_1$  into equation (16) and rearranging,

$$N_{22} = \hat{\Lambda} N_{02}\phi \left[ \frac{1 - e^{-(\lambda_{22} + \sigma_{22}\phi)t}}{\lambda_{22} + \sigma_{22}\phi} - \frac{e^{-\sigma_{11}\phi t} - e^{-(\lambda_{22} + \sigma_{22}\phi)t}}{(\lambda_{22} + \sigma_{22}\phi - \sigma_{11}\phi)} \right] + N_{22}^0 e^{-(\lambda_{22} + \sigma_{22}\phi)t} \quad (18)$$

The accurate determination of the  $U^{232}$  in irradiated fuels requires that the constants in the preceding equations be reasonably well defined. Although the value of the half-life of  $U^{232}$  is well established, neutron absorption cross-sections for  $Pa^{231}$  and  $U^{232}$  are not well known. The commonly quoted values of  $\sigma_{11} = 260 \pm 50b$  and  $\sigma_{22} = 300 \pm 100b$  were determined radiochemically and are effective cross-sections in the Materials Testing Reactor. The neutron spectra for the Elk River and Indian Point reactors are so different from the MTR spectrum that it is not at all clear that MTR cross-sections are applicable to this work. More recent values are discussed in Section D below.

### 3. $U^{232}$ in the RCPA Elk River Reactor

The effect of the values of  $\sigma_{11}$  and  $\sigma_{22}$  on the concentration of  $U^{232}$  in Elk River fuel was investigated. An average fuel exposure of 10,000 MWD/T\* and a thermal neutron flux of  $1.1 \times 10^{13}$  n/cm<sup>2</sup>-sec served as a basis for the calculations. The concentration of  $U^{232}$  as a function of  $\sigma_{11}$ ,  $\sigma_{22}$  and the number of cycles is shown in Figs. 1 and 2. Note that the concentration of  $U^{232}$  is sensitive to the value of  $\sigma_{22}$  only after several cycles, but that the cross-section of  $Pa^{231}$  is immediately important.

The effect of fuel exposure on the concentration of  $U^{232}$  and  $Pa^{231}$  in Elk River fuel is shown in Fig. 3. Again, a constant thermal flux of  $1.1 \times 10^{13}$  n/cm<sup>2</sup>-sec is assumed, and the cross-sections for  $Pa^{231}$  and  $U^{232}$  are taken to be 260b and 300b respectively. General data for the Elk River Reactor calculations were obtained from "Final Reactor Physics Report for the RCPA Elk River Reactor"<sup>(3)</sup> and are summarized in Table I.

The effect of thermal neutron flux on the  $U^{232}$  concentration was investigated. Using the same constants for cross-section values, and a lifetime of 10,000 MWD/T, the thermal flux was varied from  $1.1 \times 10^{13}$  n/cm<sup>2</sup>-sec to  $1.5 \times 10^{13}$  n/cm<sup>2</sup>-sec. The  $U^{232}$  concentration increased from 2.18 ppm to 2.88 ppm for the first cycle and from 5.4 ppm to 6.7 ppm for the third cycle.

The effect of non-uniform power distribution was also studied. The core was divided into three radial zones of equal volume with six axial regions in each zone. For an average thermal flux of  $1.1 \times 10^{13}$  n/cm<sup>2</sup>-sec, the average  $U^{232}$  concentration in each zone was calculated:

<u>Core Zone</u>	<u>Avg. <math>U^{232}</math> Concentration (ppm)</u>
Central Zone	3.84 (in thorium)
Middle Zone	2.36 (in thorium)
Outer Zone	1.22 (in thorium)

\* Per metric ton of metal.

This gives an average  $U^{232}$  concentration for the core of 2.5 ppm compared to the values of 2.18 obtained by using an average value for the flux. Expressing these concentrations in terms of parts per million in uranium fuel accentuates the variation in composition since more fuel is burned out in the center of the core:

Central Zone	94 ppm (in uranium)
Middle Zone	55 ppm (in uranium)
Outer Zone	28 ppm (in uranium)
Core Average	57 ppm (in uranium)

If the effect of the variation of thermal flux over the core life-time is taken into account, maximum concentrations of  $U^{232}$  of 125, 73, and 37 ppm (in uranium) in the central, middle, and outer zones respectively are obtained.

In summary, using MTR cross-sections, concentrations of  $U^{232}$  as high as 125 ppm (in uranium) may be expected in some Elk River fuel elements, with the average composition near 75 ppm.

#### 4. $U^{232}$ in The Consolidated Edison Indian Point Reactor

The data used to calculate the concentration of  $U^{232}$  in the fuel of the Indian Point Reactor are given in Table II. The cross-sections of  $Pa^{231}$  and  $U^{232}$  were again assumed to be 260b and 300b respectively.

The build-up of  $U^{232}$  in the fuel was calculated using three models:

- I. A core with uniform and average thermal neutron flux and fuel composition.
- II. A core with three-zone loading, as described, with the power density assumed equal in each zone. Thermal flux and fuel composition are uniform in each zone.
- III. A core with three-zone loading, as described, with the power density assumed equal in each zone. The effect of non-uniform axial power and flux distribution is taken into account.



It is apparent that Model III should give the best result but it is of interest to investigate the variation that may arise through the choice of models:

Model I: Using an average thermal flux of  $2.16 \times 10^{13}$  n/cm<sup>2</sup>-sec, the concentration of U<sup>232</sup> after one 20,800 MWD/T fuel cycle is 6.9 ppm in thorium or 142 ppm in the uranium. If the uranium were recycled through the core, the concentration of U<sup>232</sup> would be 11.8 ppm in thorium (or 242 ppm in U) after the second cycle and 15.3 ppm in thorium (or 312 ppm in U) after the third cycle.

Model II: Assuming that the power density in each zone is the same, the thermal flux would vary inversely with the enrichment in the zone. The average thermal neutron fluxes would then be  $2.95 \times 10^{13}$ ,  $2.21 \times 10^{13}$ , and  $1.77 \times 10^{13}$  n/cm<sup>2</sup>-sec in the central, middle and outer zones respectively. The production rate of Pa<sup>231</sup> is assumed constant at  $1.132 \times 10^{-12}$ /sec per atom of Th<sup>232</sup>. If the fuel were recycled through the same zone, the concentration of U<sup>232</sup> in uranium would build up as follows:

	<u>U<sup>232</sup> ppm in Uranium</u>		
	<u>Central Zone</u>	<u>Middle Zone</u>	<u>Outer Zone</u>
First Cycle	240	146	91
Second Cycle	390	247	158
Third Cycle	481	319	209

Model III: Assuming an average thermal neutron flux of  $2.16 \times 10^{13}$  n/cm<sup>2</sup>-sec and taking into account the spatial variation of power density, fuel composition and burnout after an average exposure of 20,800 MWD/T, the concentration of U<sup>232</sup> is 9.5 ppm in thorium or 260 ppm in uranium in the central zone, 157 ppm in uranium in the middle zone and 98 ppm in uranium in the outer zone. The average first cycle concentration for the entire core would be 150 ppm.

D. Recent Status of Nuclear Cross-Section Data and the Buildup of  $U^{232}$  in Fuel.

The strong dependence of the build-up of  $U^{232}$ , in the irradiated fuel of reactors using thorium as fertile material, on the neutron absorption cross-section of  $Pa^{231}$  is demonstrated by Fig. 2. An increase of the effective  $Pa^{231}$  cross-section from 200 b to 300 b increases the  $U^{232}$  concentration by 44% on the first cycle. This increase is amplified if the fuel is recycled.

The effect of the neutron absorption cross-section of  $U^{232}$  on its concentration is shown in Fig. 1. Large variations of the  $U^{232}$  cross-section do not reflect significantly in the first fuel cycle but if fuel is recycled the cross-section becomes somewhat important.

Since the data reported in the literature give no indication of the resonance capture integral, a survey was made to obtain the latest data available.

The commonly quoted values of the  $Pa^{231}$  and  $U^{232}$  cross-sections are not accurately determined. The value quoted in BNL 325<sup>(4)</sup> for the activation cross-section of  $Pa^{231}$  is  $200 \pm 15$  for "pile neutrons". No value of the  $U^{232}$  cross-section is given in BNL 325. The Reactor Handbook<sup>(5)</sup> lists a value of  $260 \pm 50$  b for  $Pa^{231}$  and  $300 \pm 100$  b for  $U^{232}$ . No indication of the resonance absorption cross-sections for these isotopes is given in the literature.

Fortunately, measurements are now in progress to determine the cross-section of  $Pa^{231}$  at the Materials Testing Reactor by Phillips Petroleum Company. The preliminary data obtained indicate that the 2200 M/sec total cross-section value of  $Pa^{231}$  is  $211 \pm 5$  b ( $\sigma_n = 200$  b) and the resonance integral is about  $1200 \pm 200$  b. About 75% of the resonance integral is due to the 0.396 Mev resonance. These measurements were made with a "fast chopper" and are sufficiently accurate for the present purposes. These data will be reported in MTR-ETR Technical Quarterly Progress Report IDO-16633.

Using these values, an effective neutron cross-section of  $Pa^{231}$  in the Elk River and Indian Point reactors is 350 b.

To obtain the latest data available on the  $U^{232}$  cross-section, the following agencies were contacted:

AEC Cross-Section Advisory Committee at Brookhaven National Laboratory,  
 The Nuclear Data Project of the National Research Council,  
 The Reactor Physics Branch of the AEC, and  
 Oak Ridge National Laboratory

Apparently, no measurement exists except the one yielding  $300 \pm 100$  b for pile neutrons. The reason that no precise measurements have been made is that the quantity of pure  $U^{232}$  necessary to make a sample does not exist.

Since no data exist on the resonance cross-section of  $U^{232}$ , one can only speculate as to its magnitude. In general, the effect of the neutron absorption resonance is to add an additional 30% to 50% to the effective neutron cross-section in the Elk River or the Indian Point reactors. Fortunately, increasing the  $U^{232}$  cross-section tends to reduce the  $U^{232}$  concentration, which means the previous calculations are conservative in this respect. It is, however, unfortunate that the thermal cross-section of  $U^{232}$  is not known better than  $\pm 30\%$  and the resonance integral has not been measured.

Since the effective neutron absorption cross-section of  $Pa^{231}$  is now known to be 350 b, it is of interest to apply this new value to the calculations of  $U^{232}$  content in the Elk River and Indian Point reactor fuels.

In the Elk River Reactor after a 10,000 MWD/T average fuel exposure, assuming  $\sigma_{22}$  at 300 b, and taking into account the variation of the radial and axial flux distribution, the concentration of  $U^{232}$  would be:

Central Zone	168 ppm (in uranium)
Middle Zone	98 ppm (in uranium)
Outer Zone	50 ppm (in uranium)
Core Average	105 ppm (in uranium)

In the Indian Point Reactor after an average fuel exposure of 20,800 MWD/T the concentration of  $U^{232}$  would be:

Central Zone	350 ppm (in uranium)
Middle Zone	211 ppm (in uranium)
Outer Zone	132 ppm (in uranium)
Core Average	219 ppm (in uranium)

### III. PLANT OUTPUT

As stated in the Introduction, the Consolidated Edison Indian Point Reactor was chosen as the reference reactor for this study. Consequently, both the size of the conceptualized plant and its output of refabricated fuel elements are determined by the needs of this reactor.

The fuel elements of the Indian Point reactor are assemblies of stainless steel tubes containing  $UO_2$ - $ThO_2$  pellets wherein the  $UO_2$  content lies between 4 and 7 percent. Some of the fuel element specifications are:

cladding	304 SS, boron modified
fuel rod OD	$0.304 \pm 0.0005$ inch
clad thickness	0.0205 inch
fuel pellet diameter	$0.260 \pm 0.0005$ inch
fuel pellet l/d	3:1
pellet density	92.3%
number rods/element	195 (14 x 14 - 1)
active fuel length/rod	98.5 inches
number elements/core	120

#### Helium Gap (annular width)

0.001 inch minimum
0.0015 inch average
0.002 inch maximum

#### U loading:

Zone I	32 elements at 6.75 kg U/element = 216 kg
Zone II	44 elements at 8.79 kg U/element = 386 kg
Zone III	44 elements at 11.30 kg U/element = 498 kg

TABLE IElk River Reactor Parameters (3)

Initial Enrichment	$\frac{U^{235} \text{ (as metal)}}{\text{total Th + U (as metal)}}$	-	4.3%
Total Amount of Fuel (Th + U) (as metal)		-	3970 kg
Spent Fuel Composition (10,000 MWD/T)		-	based on $U_o^{235}$
	$U^{235}$	-	0.73 $U_o^{235}$
	$U^{236}$	-	0.051 $U_o^{235}$
	$U^{233}$	-	0.106 $U_o^{235}$
Core Volume		-	$2.465 \times 10^6 \text{ cm}^3$
Thermal Neutron Flux (beginning of cycle)		-	$1.1 \times 10^{13} \text{ n/cm}^2 \text{ sec}$
Thermal Neutron Flux (at 10,000 MWD/T)		-	$1.5 \times 10^{13} \text{ n/cm}^2 \text{ sec}$
Thermal Power		-	58.2 Mw

TABLE IIConsolidated Edison Indian Point Reactor Parameters

Thermal Power	- 585 Mw
Average Fuel Exposure (per metric ton of thorium)	- 20,800 MWD/T
Initial Fuel Loading	
Total Thorium (as metal)	- 16,900 kg
Total U <sup>235</sup> (as metal)	- 955 kg
Thermal Neutron Flux (average in fuel)	- 2.16 x 10 <sup>13</sup> n/cm <sup>2</sup> sec
Three Zone Core Composition - Initial	
Central Zone - 32 subassemblies	- 4.2% UO <sub>2</sub>
Middle Zone - 44 subassemblies	- 5.6% UO <sub>2</sub>
Outer Zone - 44 subassemblies	- 7.0% UO <sub>2</sub>

Initial Composition of One-Half Subassembly

(5.75" x 5.75" x 49.5")

Material	Central Zone	Middle Zone	Outer Zone
U	3.2 kg	4.2 kg	5.2 kg
Th	71	70	69
O <sub>2</sub>	10.2	10.2	10.2
Steel	25.6	25.6	25.6

Fuel Composition of One-Half Subassembly  
After 20,800 MWD/T Average Exposure

Material	Central Zone	Middle Zone	Outer Zone
U <sup>233</sup>	0.706 kg	0.706 kg	0.519 kg
U <sup>234</sup>	0.044	0.043	0.020
U <sup>235</sup>	1.400	2.106	3.338
U <sup>236</sup>	0.285	0.347	0.309
U - total	2.6	3.4	4.5
Th	70	69	68

Fig. 1 -  $U^{232}$  concentration in Elk River Fuel as a function of  $U^{232}$  neutron absorption cross-section.

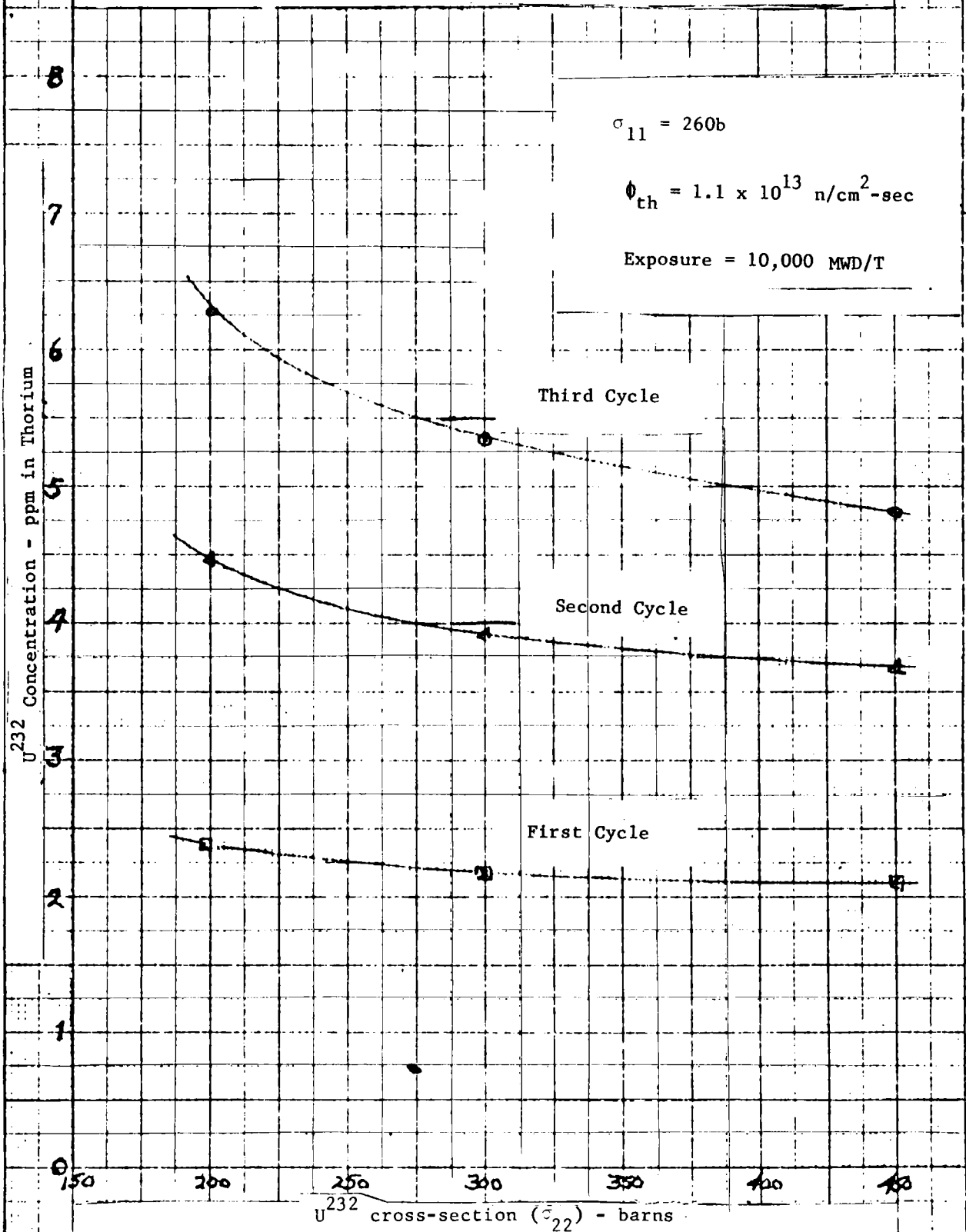
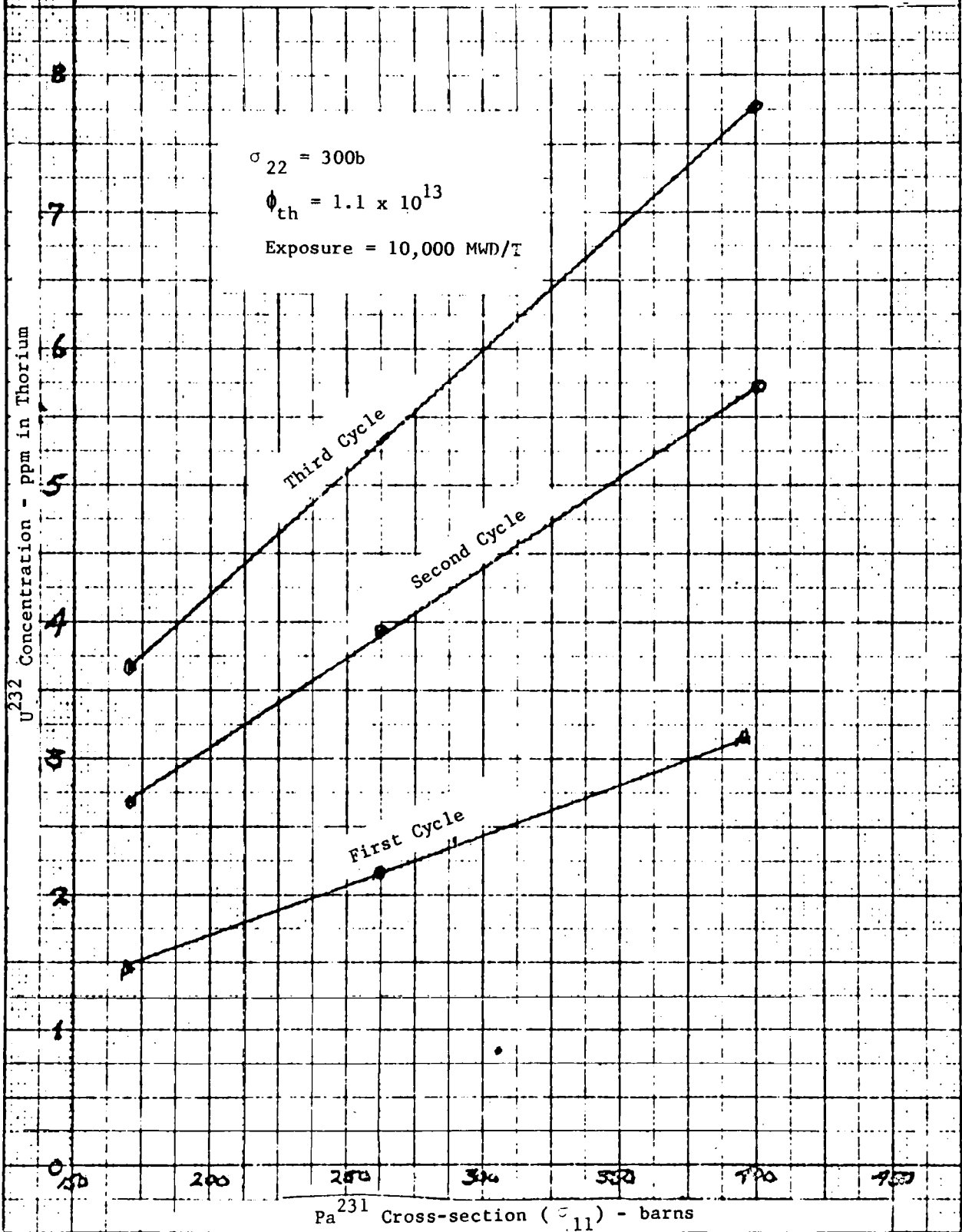
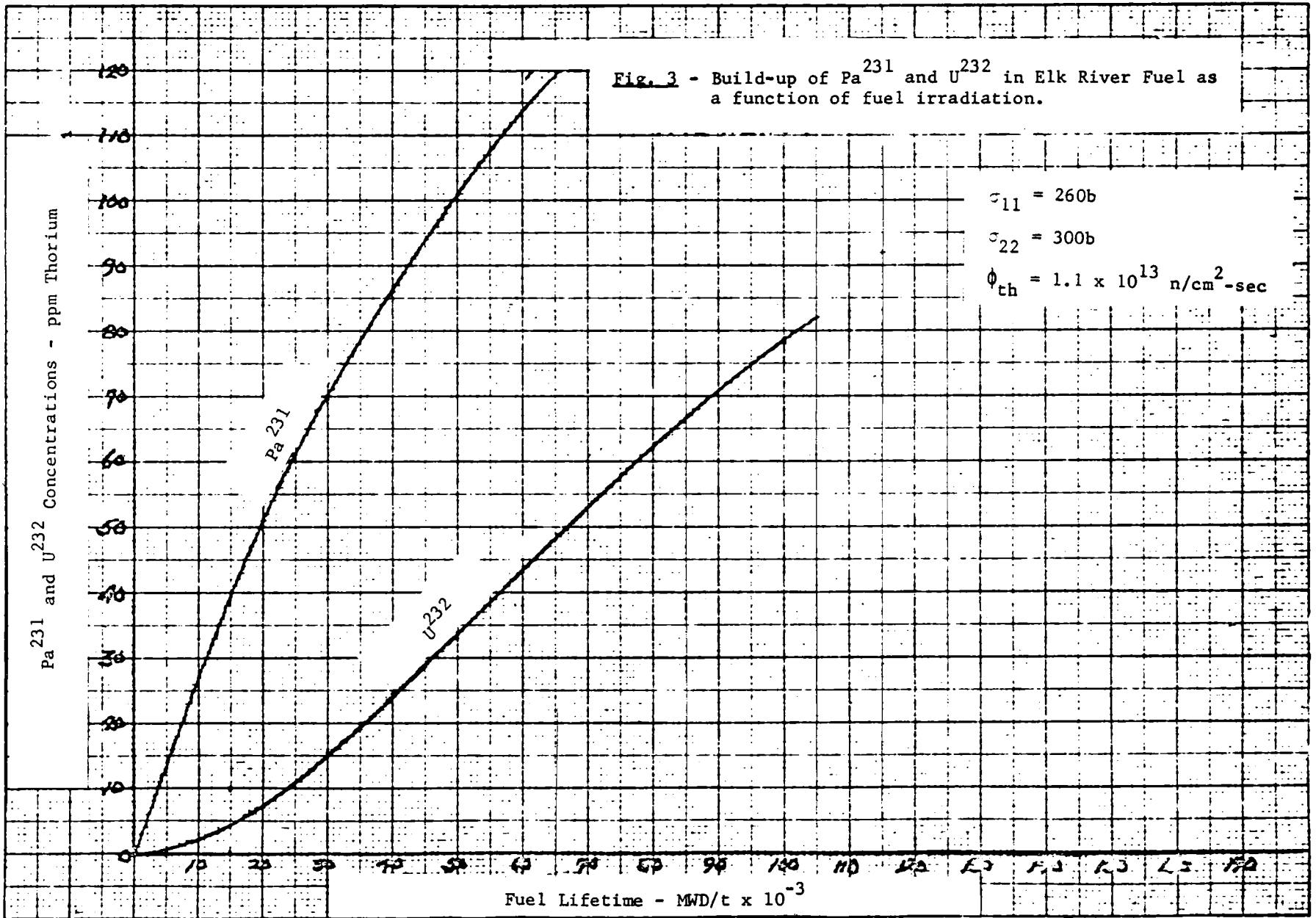


Fig. 2 -  $U^{232}$  concentration in Elk River Fuel as a function of  $Pa^{231}$  neutron absorption cross-section.







Each fuel element contains 195 loaded tubes in a (14 x 14 - 1) array as shown in Fig. 4. Figure 4 also depicts the fuel loading in each of the three types of fuel elements. Figure 5 is a sketch of a complete fuel element. Each 195-rod element contains fuel rods of at least four of the six possible loadings (varying in  $\text{UO}_2$  percentage). The assembly of a complete element therefore would require processing four different batches of  $\text{UO}_2$ - $\text{ThO}_2$  and temporarily storing some batches in the form of completed rods until they are needed in the assembly.

This complicated fuel loading introduces problems in production scheduling, especially because of the radiation build-up and the requirement for only small batches of certain fuel concentrations. To alleviate this situation, consideration has been given to the use of fresh  $\text{U}^{235}$  for part of the core loading. It is necessary in any event to add some  $\text{U}^{235}$  to the recycled  $\text{U}^{233}$ -bearing fuel to replace the fuel that has undergone burn-up. Therefore, we have adopted a scheme which uses  $\text{U}^{235}$  for the small batches. Table III presents a fuel allocation in which only one kind of fuel rod (i.e., fuel at one concentration) for each type of fuel element need be fabricated from  $\text{U}^{233}$ -bearing fuel. This allocation requires that 31.6 percent of the rods in the new core loading be prefabricated with  $\text{U}^{235}$  and that 23.1 percent of the fuel is fresh  $\text{U}^{235}$ . The scheduling of many batches of  $\text{U}^{233}$ -bearing fuel at six different concentrations, and the concomitant problems, are thereby eliminated. Present plans for the Indian Point Reactor call for replacing the entire core simultaneously at the end of the 600-day core life. Inasmuch as the plant should be capable of refabricating fuel for other  $\text{U}^{233}$ -cycle reactors, we have chosen a throughput rate such that an entire core loading for Indian Point will be produced in about 420 days. This throughput corresponds to a rate of two fuel elements per week. Since the Indian Point Reactor will have remote loading equipment and storage facilities for more than one core load, it is planned to ship the fuel elements to the reactor site promptly upon their completion. Limited storage facilities are, however, part of the refabrication plant.

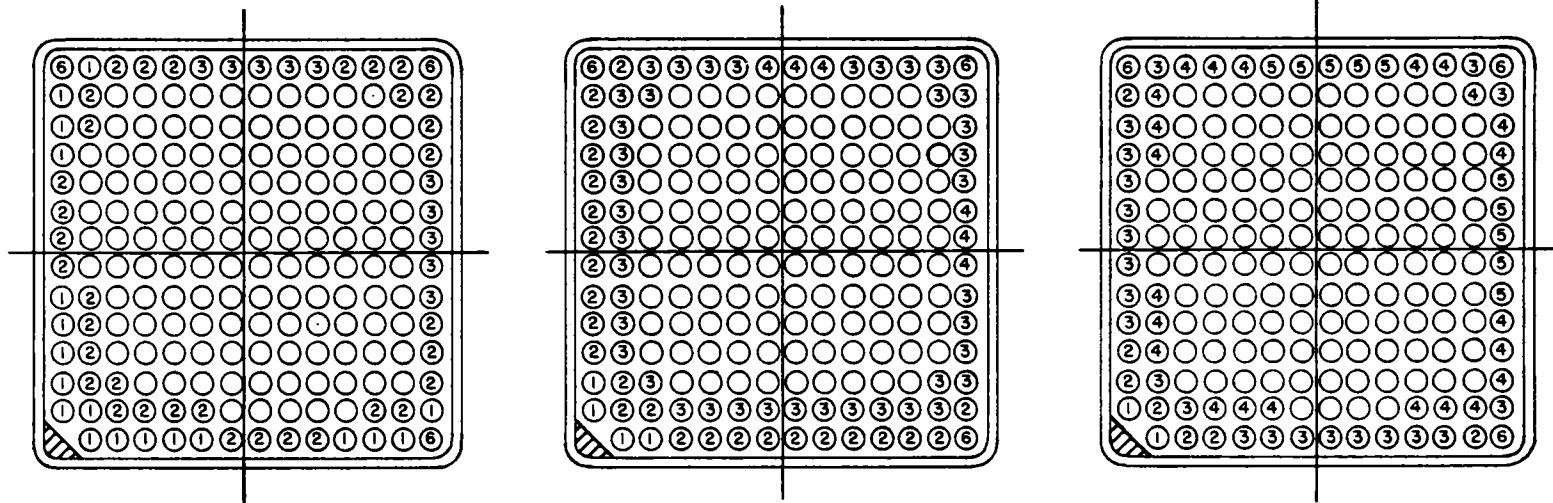
TABLE III

Allocation of Fuel for Fuel Element Types I, II and III \*

Type of Rod	Fuel Content of Each Rod (gms)	No. of Rods in Each Type Fuel Element			Total U <sup>235</sup> (clean) (kg)	Total U <sup>235</sup> + U <sup>233</sup> (kg)
		I	II	III		
No. 1	20.2	608	176	88	17.6	
No. 2	28.8	1088	1100	308	71.8	
No. 3	38.5	<u>4448**</u>	1760	968	105.0	171
No. 4	52.1		<u>5412**</u>	1012	52.8	282
No. 5	65.1			<u>6072**</u>		395
No. 6	13.5	<u>96</u>	<u>132</u>	<u>132</u>	<u>4.9</u>	<u>    </u>
		6240	8580	8580	252	848

\* See Fig. 4. Total number of rods prefabricated = 7418 or 31.6% of total.

\*\* Use contaminated fuel - all others use clean U<sup>235</sup>.

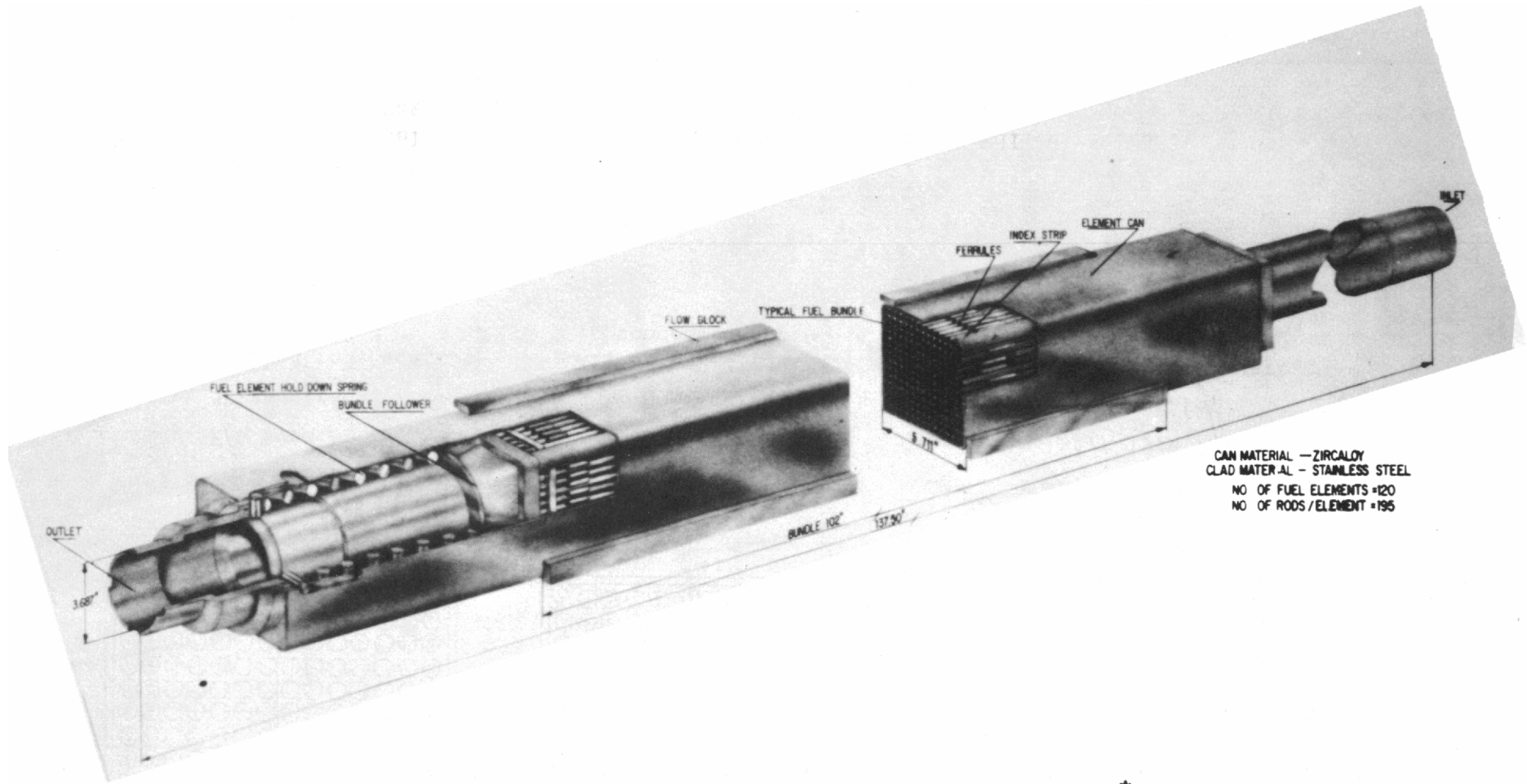


F/A Type I  
32 required  
All unmarked rods are  
concentration 3

F/A Type II  
44 required  
All unmarked rods are  
concentration 4

F/A Type III  
44 required  
All unmarked rods are  
concentration 5

**Fig. 4** - Fuel element loading pattern for Consolidated Edison reactor.  
Drawing No. RA-1916.



CAN MATERIAL - ZIRCALOY  
 CLAD MATERIAL - STAINLESS STEEL  
 NO OF FUEL ELEMENTS = 120  
 NO OF RODS / ELEMENT = 195

**Fig. 5 - Fuel Element, Consolidated Edison Thorium Reactor.\***

\* Reprinted by courtesy of Babcock and Wilcox Company, Lynchburg, Virginia, from "Hazards Summary Report, Consolidated Edison Thorium Reactor", USAEC, Docket No. 50-3, Exhibit K-5 (Rev. 1), January 1960

#### IV. FUEL PROCESSING

##### A. Refabrication Process (General)

The  $U^{233}$  fuel to be refabricated is shipped from the thorium and fission product removal facility as a uranyl solution in shielded casks. As a first step in the direct-access, rapid refabrication process, it is necessary to separate the  $U^{232}$  daughters from the uranium. This operation establishes the time-zero for buildup of the high-energy, gamma-emitting daughters of  $U^{232}$ .

Urania-thoria powder is then produced from the decontaminated uranium and fresh thorium. As stated in Section I (Introduction), the process of pellets-in-tubes has been selected for study in this work since it is the longest and most tedious process and, if feasibility can be demonstrated with it, the other processes will not be more difficult to adapt. The preparation of  $UO_2$ - $ThO_2$  powder suitable for making pellets may be carried out in a number of ways including the following:

1. Dry blending of  $UO_2$  with  $ThO_2$  in a ball mill followed by conventional powder metallurgy techniques for pelletizing.
2. An Argonne National Laboratory technique of dry blending  $U_3O_8$  and  $ThO_2$  in a ball mill, pelletizing, and reducing the  $U_3O_8$  to  $UO_2$  in a high temperature sintering furnace. This technique has a disadvantage in that the transformation of  $U_3O_8$  to  $UO_2$  leads to a large shrinkage of the pellet, making conformance to tight tolerances with low grinding losses very difficult.
3. A technique proposed by Sylcor involving a precipitation-evaporation process for obtaining a mixed crystalline material of urania and thoria. This technique is better known as partial precipitation, and utilizes the addition of oxalate to a solution of uranium and thorium. All of the thorium and most of the uranium are precipitated as

the respective oxalates. However, a portion of the uranium oxalate is soluble and remains in solution, necessitating evaporation of the last traces of solution in order to minimize uranium losses. The advantage of this technique is that the precipitated material is one which is easily filtered and readily calcined to  $\text{UO}_2\text{-ThO}_2$  in open pans without spattering.

4. A method developed at Nuclear Metals, Inc. involving coprecipitation of the uranium and thorium using either aqueous ammonia or ammonia gas. The finely interspersed precipitates are ammonium diuranate (ADU) and thorium hydroxide. The precipitated mixture can be filtered without great difficulty on a modified rotary drum filter. Calcining the resulting product does involve spattering, but the use of a rotary kiln, described in a later section, obviates this problem. In the rotary kiln, the ADU and thorium hydroxide are simultaneously ball milled and calcined, and then ball milled and hydrogen reduced to form a powder of  $\text{ThO}_2\text{-UO}_2$  in solid solution. Powder prepared by this method appears to be ideally suited for pelletizing and sintering.

The  $\text{UO}_2\text{-ThO}_2$  powder is pressed into pellets and the pellets are sintered in a reducing atmosphere. After the pellets are inspected and automatically ground to correct dimensions, they are loaded into stainless steel fuel tubes.

The fuel tubes are then welded shut and leak tested. Bundles of fuel tubes are brazed together to form a fuel element. The fuel element is assembled and shipped or temporarily stored.

Figure 6 presents a process flow sheet for the conceptual fabrication plant. The following sections describe the operations involved in the flow sheet, and their background information.

## B. Daughter Product Separation

### 1. General

ORNL practice with  $U^{233}$  has been to decontaminate the uranyl nitrate solution, just prior to shipment, to a maximum activity of 100 mr/hr as indicated on an ionization chamber detector held externally at the middle of the polyethylene shipping bottle (14 liter capacity). Just before starting the refabrication process, a separation of  $U^{232}$  daughters from the uranyl solution is necessary, since, if the material is allowed to stand, the radioactivity doubles in three to four days and reaches 1/2 to 1 R/hr after a few weeks (10 liters of solution containing 125 to 150 grams uranium per liter). The following methods are available for daughter product separation.

#### a. Ion Exchange

(1) Cation (Resin - Dowex 50). The separation of  $U^{232}$  daughters from  $U^{233}$  fuel material by means of Dowex 50 has been studied at ORNL in fuel containing up to 40 ppm of  $U^{232}$ . Virtually 100 percent of the  $U^{232}$  daughters are removed from the first few column volumes of solution. However, daughter products begin to break through the column rather quickly. At moderate-to-high throughputs, the daughter removal is rather low. Another disadvantage is that approximately 5 percent of the uranium processed is retained by the resin.

(2) Anion. Some work on the separation of uranium and thorium has been done using anion resins.<sup>(6)</sup> Dowex 1 can be used to absorb uranium from strong HCl while thorium and radium do not load. Similarly, Decadite FF, an anion resin, can be used to load uranium and thorium from sulfate systems. Uranium is then selectively eluted from the column with one molar nitric acid. Unfortunately, all of these systems have the same disadvantage, i.e., retention of some uranium by the resin.



b. Solvent Extraction (SX)

In considering solvent extraction techniques for the separation of uranium and thorium, two alternatives present themselves. The first is to remove the desirable species, uranium, from the unwanted species, the daughter products, into an organic phase leaving the unwanted species in the aqueous raffinate. The second approach is to extract the unwanted species into the organic phase and leave the desirable species, uranium, in the aqueous phase. If the first approach is taken, a step must be included to strip the uranium from the organic back into the aqueous phase for the further processing (coprecipitation, etc.). In the case of  $U^{232}$  daughters, separation of the uranium from the aqueous phase into the organic phase is desirable since the daughters are chemically so different from each other that the existence of an extractant which will remove all the daughters and leave the uranium is not apparent.

At Oak Ridge, the thorex facility has been used to separate the  $U^{232}$  daughters from uranium fuel material. It should be pointed out that this approach may have been desirable at Oak Ridge because it utilized existing equipment and technology. However, the thorex process was designed for the separation of major amounts of thorium and fission products from smaller amounts of uranium. In order to extend its usefulness to the  $U^{232}$  daughter product separation, it is necessary to use a large number of stages. This is accomplished in a small apparatus by continually recycling the uranium through the system, taking only a small "center cut" of uranium as product at each pass. An average of 2.5 passes through the circuit is necessary to achieve reasonable decontamination. Hence, it would not be economical to reproduce this system in a plant designed to refabricate  $U^{233}$  fuel material.

Because of the inherent disadvantages of an ion exchange system, it was felt to be most desirable for the present purposes to utilize a solvent extraction system specifically tailored for the separation of  $U^{232}$  daughters from uranium. The selected system uses a commercially available tertiary amine extractant from a three molar hydrochloric acid solution (which may contain up to 0.2 molar nitric acid).

In this conceptual plant design, mixer-settler units rather than SX columns have been used. Mixer-settler units have an advantage in very closely approximating a theoretical extraction stage; thus, prediction of the number of stages necessary for adequate separation is facilitated. Further, it is anticipated that the proposed separation facility will be used intermittently rather than continuously, and on shutting down the SX system, mixer-settlers do not lose equilibrium. Although column geometry is more desirable from a criticality viewpoint, mixer-settlers were used since the separation data collected and presented below were most conveniently obtained in this type of equipment.

## 2. Separation Data for Solvent Extraction

The problem involved is the separation of  $U^{232}$ ,  $U^{233}$ , and  $U^{235}$  from the radioactive daughter products of  $U^{232}$ ; i.e.,  $Th^{228}$ ,  $Ra^{224}$ , and  $Pb^{212}$ . The other daughters,  $Rn^{220}$ ,  $Po^{216}$ , and  $Bi^{212}$  are of short half-life and their behavior is relatively unimportant provided the uranium is separated from the longer-lived parents, 1.9-year  $Th^{228}$ , 3.64-day  $Ra^{224}$ , and to a lesser degree, 10.6-hour  $Pb^{212}$ .

There are several restrictions on any proposed scheme. First, uranium recovery should be virtually quantitative due to the great value of  $U^{233}$  and  $U^{235}$ . Second, the separation from  $Th^{228}$ ,  $Ra^{224}$ , and  $Pb^{212}$  should be good, probably in excess of 90%. The degree of separation necessary, of course, will be a function of the age of the fuel relative to its last purification. Third, the process should allow a large throughput of uranium relative to the quantity of uranium held up in the equipment; for, to avoid problems arising from criticality, the hold up of uranium in the separation system should be less than 300 grams at a throughput of 300 grams per hour (based on processing the desired quantity of uranium in about four hours).

Solvent extraction from a hydrochloric acid medium was chosen. Whereas uranyl nitrate and thorium nitrate exhibit roughly similar extraction characteristics in many systems, there are many solvents which will extract uranyl chloride or uranyl chloride complexes without extracting measurable quantities of thorium or radium.

Three solvents were considered:

Aliquat 336 - a quaternary amine manufactured by General Mills, Alamine 336 - a tertiary amine manufactured by General Mills, and tributyl phosphate (TBP). For the aqueous environment, a high hydrochloric acid concentration would be preferred, but 3 M hydrochloric acid was chosen since it was desired to keep the dilution of the primary feed solution, uranyl nitrate, in dilute nitric acid at a minimum. For the primary feed solution, a uranium concentration of 40 g/l U in 0.2 M nitric acid was chosen.

Of the three solvents considered, Alamine 336 is preferable.

Aliquat 336 is a stronger extractant for uranyl chloride but is difficult to strip. On the other hand, TBP has a rather low extraction coefficient so that quantitative uranium recovery involves the use of too many extraction stages.

For the solvent, 20 volume percent of Alamine 336 in xylene was chosen. Using this solvent and an aqueous phase prepared by making the primary feed 3 M in hydrochloric acid by the addition of 11.7 M hydrochloric acid, an extraction isotherm was prepared. This extraction isotherm indicated that using three extraction stages at a solvent-to-aqueous ratio of about 1.1 to 1, 99.99% uranium extraction could be effected.

The stripping of uranium from the pregnant solvent was investigated. Aqueous ammonium carbonate is very effective; however, two undesirable effects were noted with the use of ammonium carbonate. First, a precipitate slowly formed in the aqueous phase, probably ammonium uranyl carbonate. Secondly, the solvent, prior to recycle, must be re-converted to the amine hydrochloride with hydrochloric acid. Also, if ammonium carbonate is used, the pregnant strip solution must be acidified, then boiled, to allow quantitative uranium precipitation.

A more satisfactory stripping solution is 0.1 M ammonium chloride solution. At a strip-solution-to-pregnant-solvent ratio of 1.3 to 1, 99% of the uranium is stripped in two stages, no precipitation is noted, and the solvent is left as the amine hydrochloride so that it can be recycled without further treatment. Furthermore, the pregnant strip solution

can be precipitated directly with ammonia. The 1% of the uranium remaining in the solvent is not a problem since the solvent is recycled to the extraction circuit, and the uranium is not lost.

Using the data described above, a flow sheet was designed for a 300-gram-per-hour system (see Fig. 7). This flow sheet was tested in the laboratory using a batch counter-current system composed of separatory funnels.

The results of the study are summarized below:

1. Uranium recovery - 99.99%
2. Thorium removal - 97% (thorium added as Th<sup>230</sup>)
3. Radium removal - 100% (radium added as Ra<sup>226</sup>)
4. Lead removal - 90% (lead added as stable lead carrier)

If a higher hydrochloric acid concentration were used, say 6 M hydrochloric acid, uranium extraction would be improved somewhat and thorium and lead removal would be greatly improved. Weighted against this would be increased dilution of the primary feed solution and the need of somewhat larger equipment for equivalent uranium throughput, and the need, probably, of an additional stripping stage.

An aspect of solvent extraction which is of utmost importance is phase disengagement. Phase disengagement in Stage 1 and Stage 2 of the extraction circuit and Stage 1 of the stripping circuit was excellent, no matter how violent the agitation. In the other two stages, Stage 3 of extraction and Stage 2 of stripping, phase disengagement is excellent if the agitation is vigorous enough to give good mixing but not violent. In actual operation, it is unlikely that any difficulty would be encountered provided mixing was not violent.

For a system handling 300 grams per hour of U, all mixers and settlers should be of one-liter capacity. For dimensions, those of a one-liter beaker are perfectly adequate, that is, a height-to-diameter ratio of about 1.5 to 1, where  $d = 10$  cm and  $h = 15$  cm. For the indicated uranium feed rate, the holdup of uranium in the entire system will be of the order of 70 grams.

The uranium-barren raffinate, which will contain about 0.003 g/l U, can be further depleted in uranium by passing the raffinate over a column of IRA-400, 50-100 mesh, in the chloride form, and initially suspended in 3 M hydrochloric acid. To accommodate the anticipated flow rate for a 300-gram-per-hour operation, the resin bed should have a volume of 1.5 liters, the exact dimensions to be determined by the requirements pertaining to criticality.

Summarizing, the following recommendations are made:

1. Feed rate - 300 grams per hour
2. Feed - 30 g/l U, 3 M HCl, 0.15 M HNO<sub>3</sub>; flow rate, 167 ml/min.
3. Solvent - 20 volume % Alamine 336 in xylene, converted to the hydrochloride prior to use in the first cycle. Flow rate, 190 ml/min.
4. Extraction - 3 mixer-settler stages, each mixer and settler 1 liter in capacity. Construction should be of chloride-proof material, but not rubber or neoprene lined. Polyethylene-lined steel should be adequate. This should be tested. Use proportional pumps such as a Proportioneer Chloro-o Feeder. For maximum safety, explosion-proof agitators should be used.
5. Stripping solution - 0.1 M NH<sub>4</sub>Cl, flow rate 250 ml/min.
6. Stripping - Two stages of mixer-settlers, each one liter in capacity. The same corrosion and pumping requirements as in the extraction circuit are necessary.

A sample of the pregnant strip solution was precipitated with ammonia, filtered, washed, and dried at 110°C. The precipitate was then ground and ignited in a crucible at 900°C for fifteen minutes. The final product had a chloride content of 80 ppm on a U basis.

Since the behavior of Bi212 may be of interest, it is worth noting that, based on experience with similar systems, bismuth will be extracted by the solvent, but not stripped by 0.1 M ammonium chloride.

### C. Mixing and Coprecipitation

After the fuel has been decontaminated it is transferred in criticality-safe batches to a tank containing the desired quantity of thorium (approx. 30 kg) in dilute nitric acid at a concentration of 100g Th/L. Care is taken to ensure that the first batch of uranium is well mixed before introduction of the second batch.

The alternative of bubbling  $\text{NH}_3$  into the mixing tank is not desirable since there is a possibility of selective precipitation of either Th or U as the solution passes through the pH range 2 to 3. By pumping the U-Th solution into the ammonium hydroxide, one starts at the high end of the pH scale and thus never goes through the pH range 2 to 3.

The mixing of U and Th early in the refabrication process is desirable, especially because of the larger amount of uranium that may be handled as a criticality-safe batch. Figures 8 and 9 show the minimum critical masses for 5 w/o  $\text{U}^{235}\text{O}_2$  in  $\text{ThO}_2$  and for 7 w/o  $\text{U}^{235}\text{O}_2$  in  $\text{ThO}_2$ . The minimum critical mass for  $\text{U}^{235}\text{O}_2$  in the absence of  $\text{ThO}_2$  is 800 grams, water-reflected and moderated, and 1400 grams  $\text{U}^{235}\text{O}_2$  if unreflected. After complete mixing the U-Th solution is pumped into a tank containing 15 M  $\text{NH}_4\text{OH}$ , which precipitates the U as ammonium diuranate (ADU) and the Th as  $\text{Th}(\text{OH})_4$ .

### D. Filtering and Calcining

The next process step is filtering of the ammonium diuranate-thorium hydroxide precipitate. The volume of the cake obtained in the precipitation is approximately 4 liters per kg of contained  $\text{UO}_2\text{-ThO}_2$ .

In the filtering operation, it is not necessary to remove the last traces of water, since the calcining is being done in the rotary kiln, where spattering of the cake will not cause losses. For the experimental precipitation and filtration runs, solutions of uranium nitrate and thorium nitrate were mixed in the desired proportions to give a 1350-gram batch of 5 percent  $\text{UO}_2\text{-ThO}_2$  mixture in 12 liters of solution at a pH of 1.0.

The precipitation was accomplished by placing 3-1/2 liters of 15 molar ammonium hydroxide into a large plastic bucket. The 12 liters of uranium-thorium solution were then pumped into this plastic bucket. The pumping rate was such that 10 minutes time elapsed for the complete transfer of the 12 liters of solution containing the uranium-thorium. The final pH of the resulting precipitated solution was 9.25. The filtration of this solution was accomplished with a vacuum-filtering apparatus consisting of 9 sq. ft. of EIMCO-PO-801-HF filtering cloth suspended by a stainless-steel mesh over a vacuum chamber. The filtrate, which consisted of 31,560 ml including the necessary wash water, was found to contain 0.0051 gram of uranium per liter.

Large volumes of wash water are necessary to ensure complete removal of ammonium nitrate. This is desirable to prevent carry-over of  $\text{NH}_4\text{NO}_3$  to the rotary kiln. Under certain conditions this could be an explosion hazard.

Figure 10 is a photograph of the laboratory apparatus used as a rotary kiln ball mill. The vessel consisted of a Vycor glass chamber 6 inches in diameter by 15 inches in length. It was mounted and rotated in a conventional Litton glassblower's lathe turning at about 50 rpm. The charge of ammonium diuranate-thorium hydroxide, equivalent to 250 grams of  $\text{ThO}_2$ -5%  $\text{UO}_2$ , was placed in the rotary kiln with a charge of one-inch diameter porcelain balls. Heat was supplied to the apparatus by means of a hydrogen-oxygen torch and a ribbon burner operating on illuminating gas-oxygen mixture. Maximum internal temperature was estimated at 850 to 900°C. The firing time-temperature schedule was as follows:

1. Argon atmosphere, 100°C for one hour.
2. Argon atmosphere, 200 to 300°C for one hour.
3. Argon atmosphere, 600 to 700°C for one hour.
4. 10 percent hydrogen-90 percent argon, 850°C for one hour.

In Step 1, the filter cake goes through an intermediate liquid state which consists of a nitric acid slurry of uranium and thorium compounds. Upon evaporation of all the liquid, a rust-colored, finely divided product remains. During Steps 2, 3 and 4, there is a gradual transition from the rust-colored powder to a fully reduced, greyish powder.

The argon atmosphere was allowed to flow through the system throughout the reduction process. Although this was not necessary from a chemical point of view, it was done to sweep out off-gasses and ensure that no oxygen was present when the hydrogen was introduced into the hot kiln. The argon also probably helps to prevent the uranium from oxidizing to  $UO_3$  in the thermal decomposition and thereby makes the subsequent reduction to  $UO_2$  a bit easier.

The following table lists the particle size distributions obtained on 95 percent  $ThO_2$ -5 percent  $UO_2$  mixtures given either a 700 to 800°C hydrogen reduction (Lot 1), or a 900°C hydrogen reduction (Lot 2).

Lot 1		Lot 2	
700-800°C Firing		900°C Firing	
Micron Fraction	%	Micron Fraction	%
-10	64.7	-10	51.6
-20 +10	0.3	-20 +10	0.7
-30 +20	24.5	-30 +20	33.7
-44 +30	10.4	-44 +30	9.7
elutriator losses 4.1%		elutriator losses 4.3%	

#### E. Pelletizing, Sintering and Grinding

It was found that the material produced in the rotary kiln could be successfully pelletized simply by incorporating 2 to 3 percent water and mixing well. Although this resulting material is not readily metered out on a volume basis, it can easily be metered on a weight basis, and commercial devices are available for doing this.



Ten-gram samples of this water-powder mixture were pressed at 11,000 psi into pellets 0.5 inch in diameter by 0.5 inch long. Average green densities of 6.6 grams/cc were obtained. It was found that, by incorporating a one-mil taper over the bottom 1/8 inch of the die, circumferential cracks, which had been appearing when the pellets were exhausted from the die prior to tapering, no longer appeared.

The pellets were sintered for two hours in argon at a temperature of 1750°C. The furnace consisted of a beryllia crucible inside a graphite sleeve which was heated by induction. Two hours were required to reach 1750°C, and because of the heavy insulation in the furnace, the rate of cooling was very slow, requiring approximately 24 hours to reach room temperature. In other attempts at sintering, it was found that heating the pellets to 1750°C in times shorter than two hours resulted in burst pellets.

The densities of the resulting pellets were as high as 98 percent of theoretical, although the average density for the fired pellets was 94 percent. It is felt that soaking at the 1750°C temperature for a longer time will result in higher densities even though the powder is pressed to a very low green density.

A series of experiments were conducted in which pellets were placed in the sintering furnace so that one pellet was in contact with a thin wafer of high-fired thoria, one in contact with a sheet of tantalum, one in contact with a sheet of molybdenum and another in contact with a piece of pyrolytic graphite. After sintering for two hours at 1750°C, only the pellet sitting on the piece of molybdenum showed no signs of reaction. The pellet sitting on tantalum was tightly stuck to the tantalum. The pellet sitting on thoria could be easily removed; however, it showed signs of reaction. The one definite result from these tests is that UO<sub>2</sub>, thoria and molybdenum are compatible at 1750°C in an argon atmosphere.

No attempts were made to grind the  $\text{ThO}_2\text{-UO}_2$  pellets produced in the laboratory. We know of no reason to suspect any difficulties with the grinding operation. The pellets will be automatically fed into automatic grinders that (1) reduce the diameter to within the specified limits and (2) face the pellet ends flat and perpendicular to the cylinder axis. A tight specification has not been set by Consolidated Edison on the individual pellet length, but only on the total length of fuel per tube. Therefore, the end grinding will be adjusted to remove the minimum amount of material consistent with flat end faces perpendicular to the cylinder axis.

Although there is some divergence of opinion among reactor designers concerning the necessity for specifying a small gap between pellet and tube (and hence necessitating the grinding of pellets and the purchasing of tubing to fine tolerances) we have adopted the conservative approach. This approach assumes that those responsible for the fuel specifications will insist on small gaps between pellets and tubing, and that if the longer times and other problems associated with a grinding operation still lead to a feasible direct-access process any further streamlining will be so much to the good.

#### F. Fuel Element Assembly

After the pellets are ground they must be loaded into stainless steel tubes and the fuel element assembly completed. Much consideration was given to the possibilities for streamlining these operations. The process finally selected is the following:

- (1) One end cap, with a small hole in it, is welded to a suitable length of stainless steel tubing.
- (2) The pellets are inserted automatically into the tube until the required fuel length is reached. Evacuation of the tube through the end cap permits a fast loading rate.

- (3) The second end cap is welded in place and the hole in the first end cap is welded shut. This is done with a tungsten inert gas shielded arc in a helium atmosphere; thus the fuel tube is filled with helium.
- (4) The welds are inspected visually and by helium leak testing.
- (5) 7 x 7 and 7 x 6 arrays of fuel tubes are brazed as subassemblies using the Ni-P brazing technique employed by Westinghouse for the PWR fuel elements.<sup>(7)</sup>
- (6) Three 7 x 7 and one 7 x 6 subassemblies are assembled into a complete fuel element.

As an aid to streamlining, it was suggested at one time that the subassemblies, or perhaps the fuel element, be made by brazing empty fuel tubes together, inserting the fuel pellets, and welding the end caps in place. This method would have the advantage that the time during which appreciable quantities of U<sup>233</sup> were handled is minimized, especially at the end of the process when the fuel is most radioactive. However, a consideration of the rejects likely to be encountered in the welding of end caps indicated that prebrazing is not feasible. D'Amore has stated<sup>(7)</sup> that, of the 46,500 end closure weldments made in manufacturing the Yankee core fuel rods, 0.35% were non-repairable rejects. Total rejects were 3.16%, of which 1% were successfully repaired by re-welding, and an additional 1.8% were repaired by replacing the end plug. Although the non-repairable rejects were very low, they are sufficiently significant so that appreciable losses would result with use of the prebrazing method.

No experimental work on fuel element assembly procedures was deemed necessary in demonstrating the feasibility of the direct-access process. Instead, reference was made to the successful results in fabrication of the Yankee Reactor core.

### G. Inspection

Numerous inspections are required to verify that the fuel elements are fabricated according to specifications. Necessary inspections are listed below in the order of their sequence during the refabrication process.

1. Spectrophotometric analysis for uranium content of incoming fuel solution. This analysis influences the input dilution to the solvent extraction (SX) step.
2. Fluorometric monitoring of SX raffinate for uranium content. If uranium content is high, the SX process must be adjusted.
3. Radioactivity monitoring of SX product. If activity is high, the SX process must be adjusted.
4. Fluorometric monitoring of the ion exchange effluent (waste stream) for uranium content. At breakthrough the column resin must be regenerated or replaced.
5. Spectrophotometric analysis<sup>(8)</sup> of SX product for uranium. If uranium content is off-limits, the SX process must be adjusted. This analysis will also influence the uranium-thorium mixing.
6. Spectrophotometric analysis of uranium-thorium solution for uranium content. If uranium content is off-limits, either uranium or thorium must be added as appropriate.
7. Determination of pH of the coprecipitated slurry. This should be 9 or greater if the uranium precipitation was complete. If pH is less than 9, adjustments must be made.
8. Fluorometric analysis of filtrate for uranium. If uranium content is high, recycle filtrate and repair filter.
9. Spectrophotometric analysis for U/Th ratio in the powder mixed with binder.

10. Determination of green pellet density. This will influence parameters in both binder-mixing and pressing steps, principally the latter.
11. Determination of sintered pellet density. Low density pellets may be resintered. If density is consistently low, previous processing from powder production on must be reviewed.
12. Visual inspection of sintered pellets for cracks, chips or other flaws.
13. Determination of pellet diameter on a go, no-go basis.
14. Inspection of stainless steel tubes for cracks, inclusions, or other imperfections, and for diameter.
15. Determination of rod fuel loading. Weigh tube empty and loaded. With the aid of the analysis of item 9, add or subtract pellets to bring total fuel loading into specification.
16. Helium leak test the welded fuel rods. Leaking closures are repaired, if possible, or rejected and the pellets reloaded in another tube.
17. Inspection of brazed assembly for distortion, lack of wetting, or alloying that is too extensive, and for proper fuel rod position with respect to composition.
18. Inspection of assembled fuel element for dimensional accuracy and for correct fuel positions.

With the exceptions noted on the Process Flow Chart, Fig. 6, no hold-up of the refabrication process appears necessary pending completion of any inspection procedure. Most of the inspections are very rapid and in some cases are electronically performed continuously. Many of the tests constitute a process control function as well as a measure of product conformity to specifications. Wherever an inspection has been judged to affect in a substantial way the scheduling of the streamlined refabrication process, it has been included in the time estimate (see Fig. 6).

## H. Scheduling

The elapsed time and the scheduling necessary for plant operation with concurrent batches is discussed in this section. Figure 6 has presented a time schedule for a single batch of material up to Rod Storage and for the subsequent fabrication of a single fuel element assembly up to Element Storage. It is, of course, not necessary to wait for a given batch to be completely processed into fuel rods before starting another batch.

To simplify the scheduling, to provide for the necessary fresh  $U^{235}$  material, and to aid in rapid refabrication, it is possible to allocate the  $U^{233}$  and the fresh  $U^{235}$  in such a way among the various fuel elements that only one type of rod in each type of fuel element need be fabricated of  $U^{233}$ . This allocation has been shown in Table III. The allocation permits about 32% of the core rods to be fabricated either in the plant on a second line or prefabricated in the plant or at another facility.

By using the data in Fig. 6 and Table III together with concurrent operations with successive batches, an operations time schedule may be derived. Such a schedule is presented in Fig. 11.

This schedule results in a rate of three fuel elements every eight days (3-shift, 24-hour per day operation) unless the prefabricated  $U^{235}$  rods are to be made on the same production line, in which case the elapsed time would be somewhat longer. If one allows 24 hours for cleanup and decontamination (in addition to the cleanup between batches) after every 3 elements, it may be readily computed that a complete Indian Point Reactor core loading would be fabricated in a year. Since this rate is about three times that required by the reactor's projected burnup, ample plant capacity remains for other work. Since the plant is already a minimum size for this refabrication, the excess capacity is in a sense "built-in". Other  $U^{233}$  or  $U^{235}$  core loadings could be handled, in some cases with addition of equipment such as vibratory compactors or swaging machines.

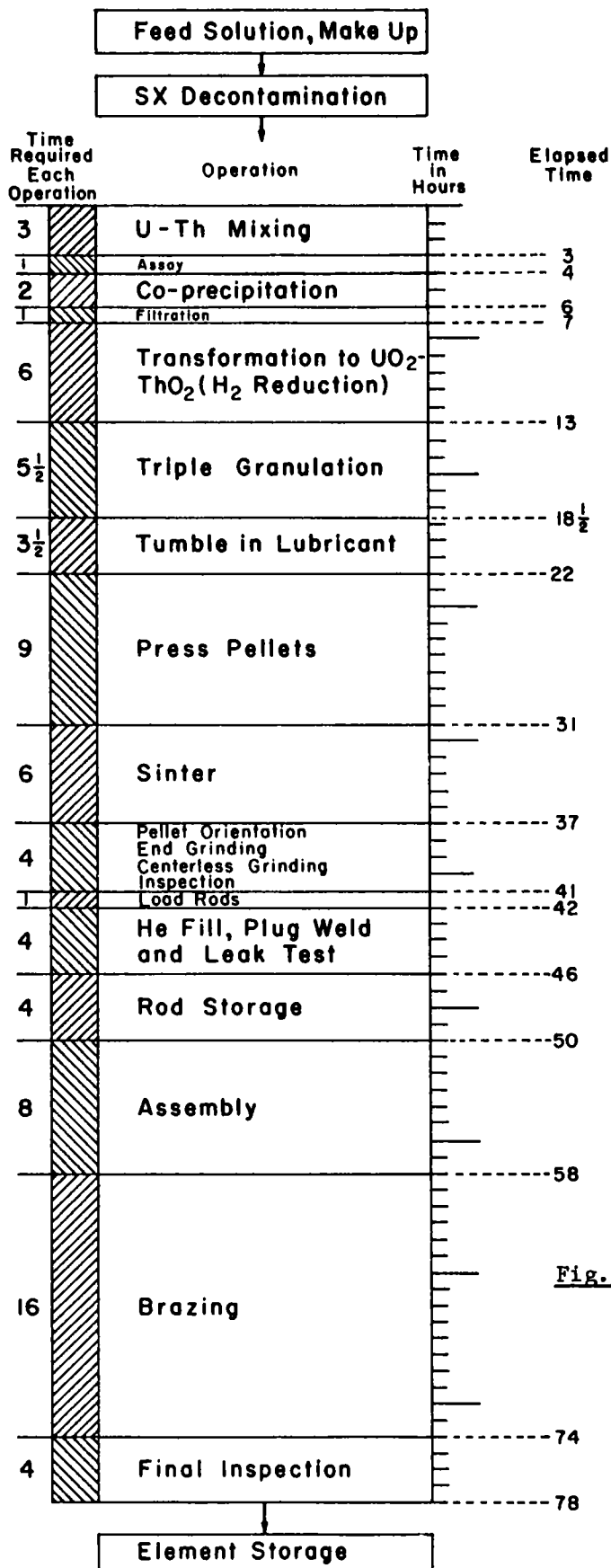


Fig. 6 - Process flow chart.  
Drawing No. RA-2040

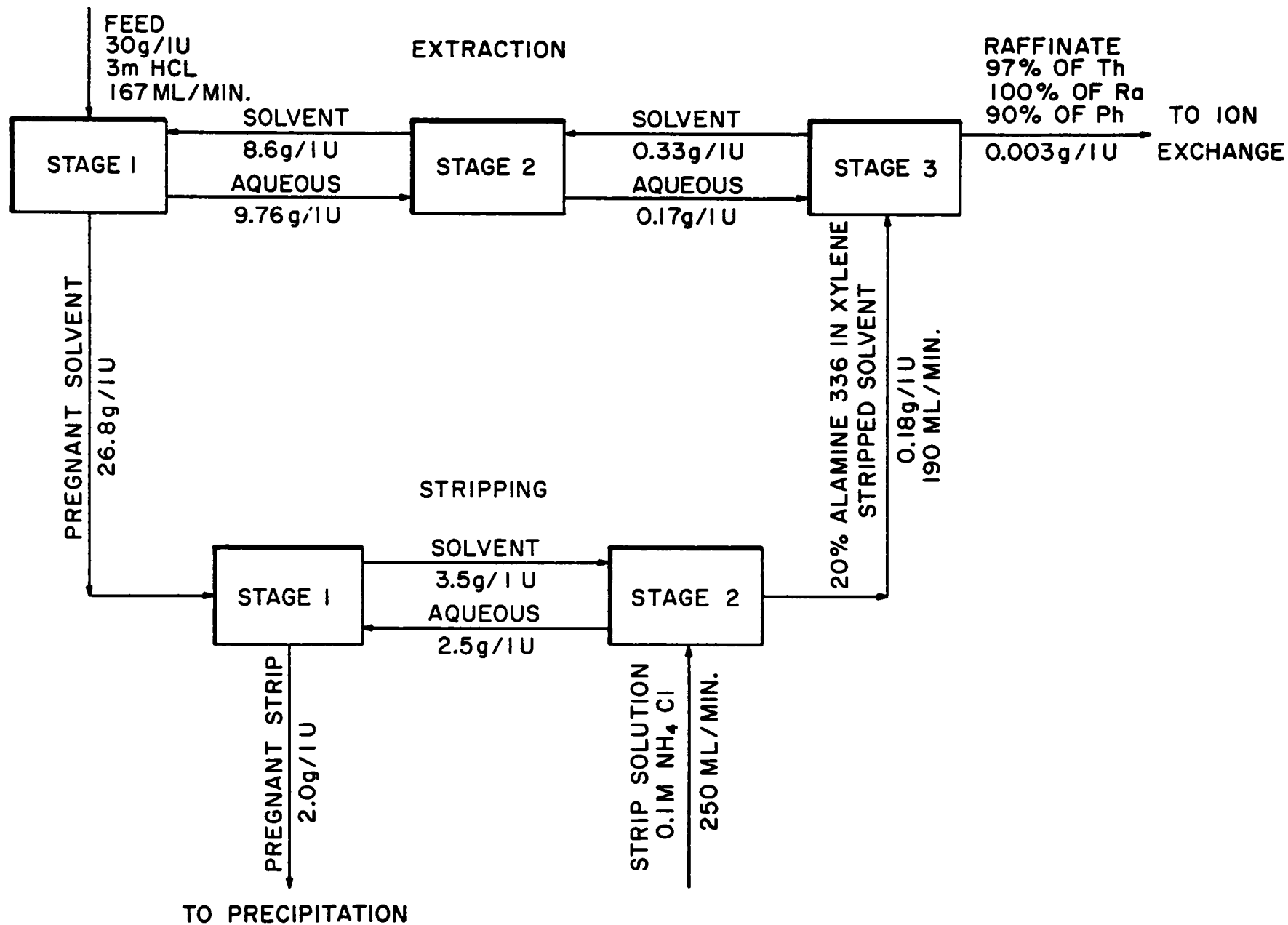
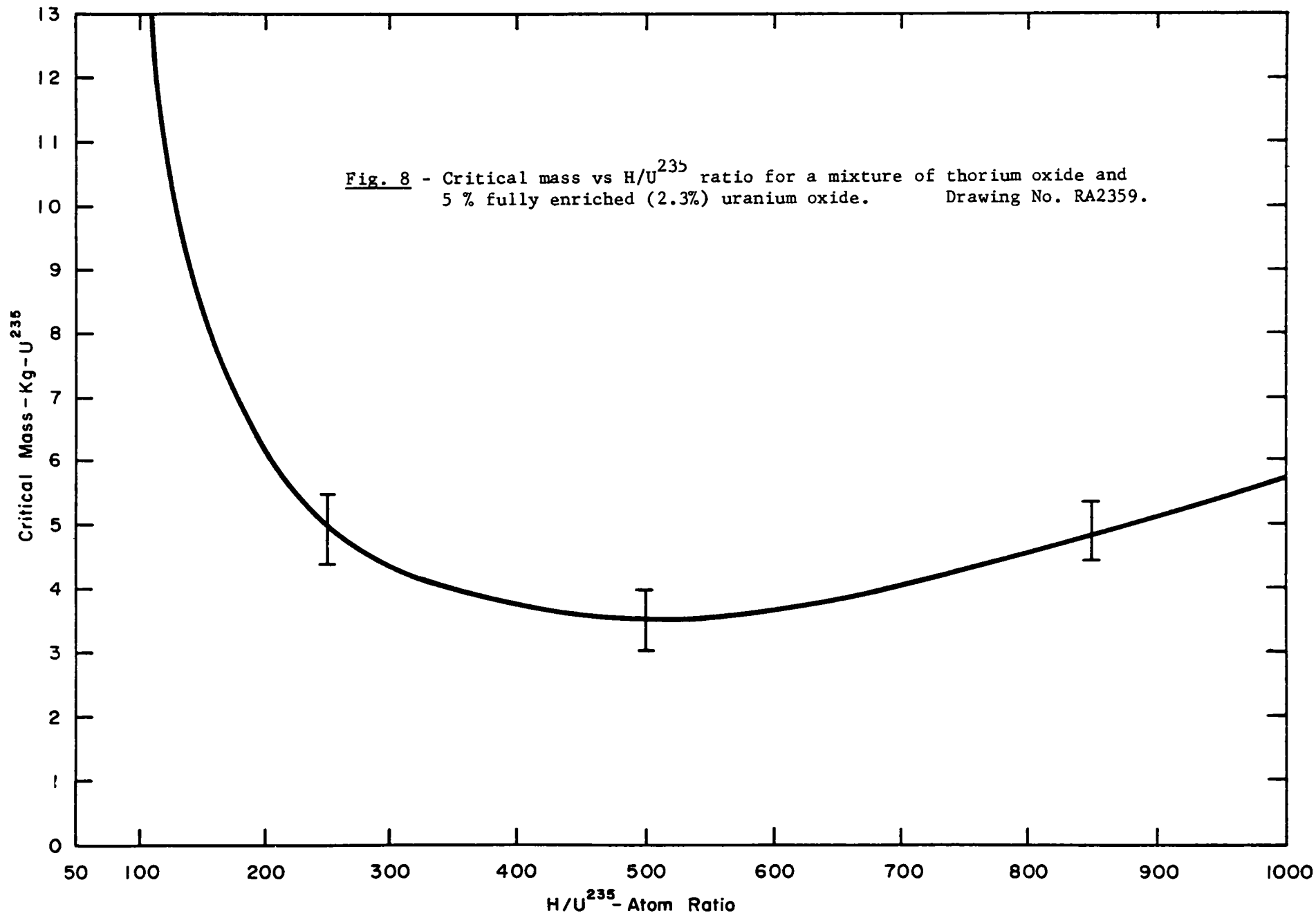


Fig. 7 - Uranium clean-up process flow sheet.  
Drawing No. RA-2358





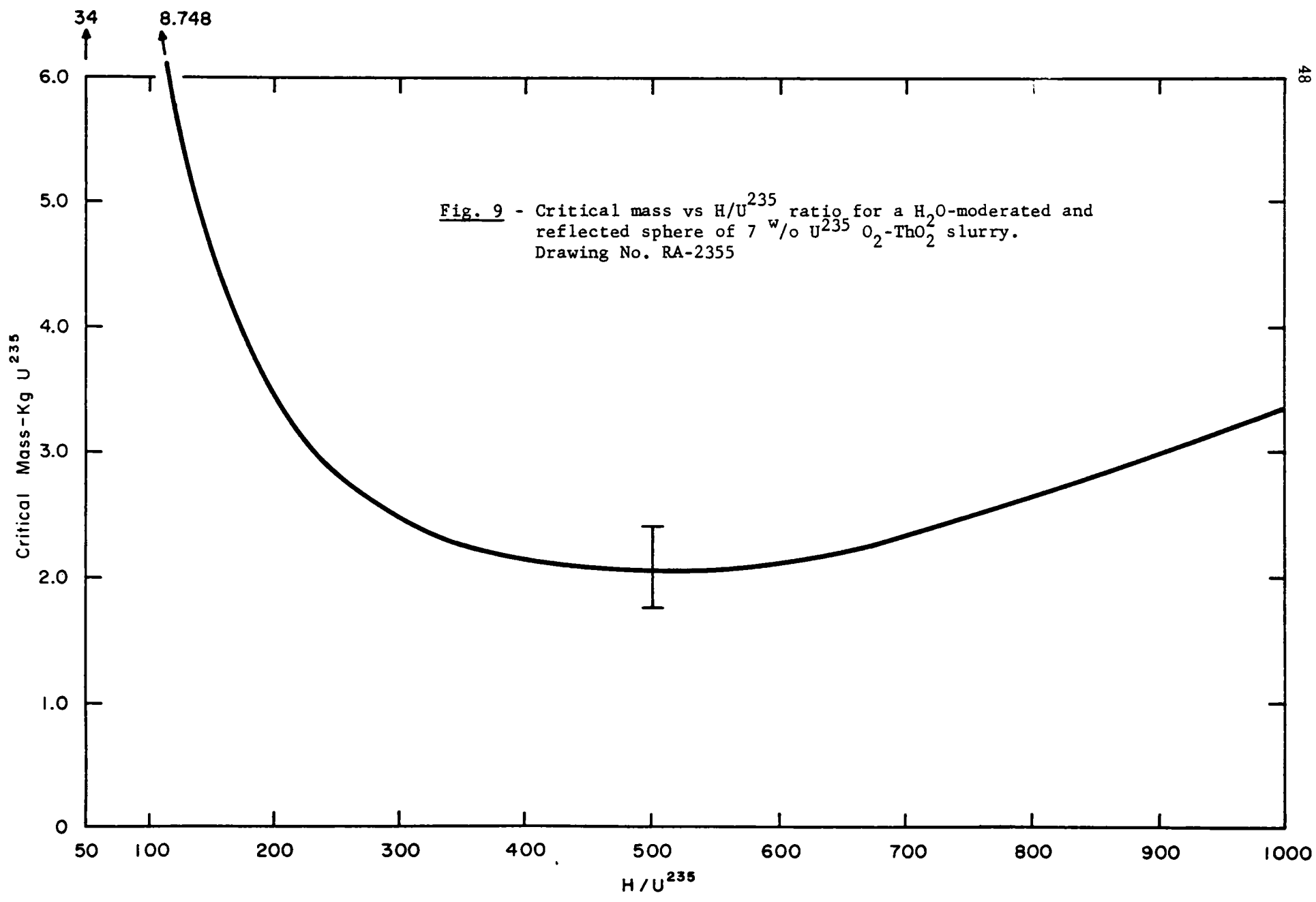
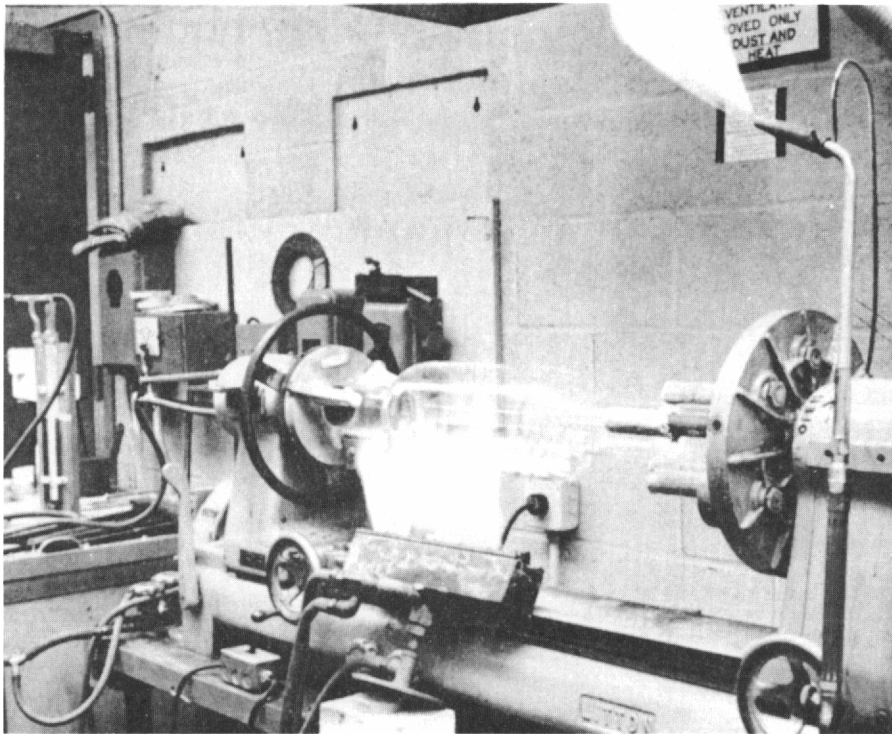


Fig. 9 - Critical mass vs H/U<sup>235</sup> ratio for a H<sub>2</sub>O-moderated and reflected sphere of 7 w/o U<sup>235</sup> O<sub>2</sub>-ThO<sub>2</sub> slurry.  
Drawing No. RA-2355



RF-7916

Fig. 10 - Laboratory apparatus used as a rotary kiln ball mill.

Type	No. of Elements	Rods/Element	Fuel Content Per Rod (gms)	U <sup>233</sup> + <sup>235</sup> Per Element (kg)	No. of Batches Per Element	In-process Time <sup>(a)</sup> (hrs)	Elapsed Time, Including Cleanup <sup>(b)</sup> (hrs)	Elapsed Time to Complete Element (hrs)
I-3	32	139	38.5	5.39	5	$(46) + (4)(4)^{(c)} = 62$	70	102
II-4	44	123	52.1	6.40	6	$(46) + (5)(4) = 66$	102	134
III-5	44	138	65.1	9.00	9	$(46) + (8)(4) = 78$	146	178 (7.4 days)

Notes: (a) Time to complete the number of batches of concentration in the given row assuming concurrent operation for that concentration only. 24 hr/day operations.

(b) 8 hrs. between different concentrations.

(c) Batches staggered by 4 hrs. for cleanup, etc.

Fig. 11 - Operations time schedule for concurrent operations.

V. PLANT DESIGNNOTE: This Section (Section V) Contains Proprietary Information

Confidential information which is the property of Vitro Corporation of America is contained herein. Such information is not to be used for purposes other than those for which it is submitted without prior written consent from Vitro Engineering Company, a division of Vitro Corporation of America.

The direct-access, rapid-refabrication concept as exemplified in the process described in the previous sections has been carefully scrutinized for feasibility from the viewpoint of architect-engineering. This effort has led to a conceptual plant design in which the following factors were evaluated:

1. Process engineering parameters
2. Site requirements
3. Equipment arrangement layout
4. Equipment specifications
5. Radiation and criticality hazards
6. Building arrangement
7. Utility requirements
8. Cost

The recommended design reflects the analysis of alternate unit operations and of alternate equipment for a unit operation. Because of the limited scope of this effort, not all desirable alternatives were investigated and there remains the need for further process and equipment development.

The feasibility of the concept, rapid processing, seems clearly established. Available standard equipment is specified. Special equipment, such as the rotary kiln (Item L-2), is based either on the experimental results gained by Nuclear Metals, Inc., or standard equipment with some modification. The recommended materials of construction reflect use of standard materials.

A detailed breakdown of the cost is presented. The cost for the  $U^{233}$  fuel refabrication plant, including the building and installed major equipment is \$1,253,360. This figure does not include site acquisition, development costs, and certain testing services and equipment.

A. Process Description

1. Chemical Unit Operations (Diagram 1382-M-1, Fig. 12)

a. Receiving Area

The  $U^{233}$  fuel is received as a nitrate solution in tri-compartmented fuel container. The containers are permanently fixed in a shipping frame (see Drawing No. 1382-R-2, Fig. 13). Random loading or storage of the containers will not result in a critical incident. The container is suitably shielded so that a delay in transit or storage of up to 30 days will not lead to surface or external dose rates exceeding the ICC or other regulations. After receipt, the containers are visually inspected and monitored prior to acceptance for storage. An area, directly adjacent to the Receiving Area and convenient to the process line, is set aside for the storage of fuel containers. In the event of an unusually long delay, a temporary shield wall may be erected about the storage area. A supply of lead brick or concrete block is adequate for this purpose.

b. Solvent Extraction

The contents of one compartment containing 350 g U in 120 g/l solution of a tri-compartmented shipping container is transferred to feed adjustment tank F-4 by slow displacement with water. The compartment is then flushed with water and the flushed-out material also added to F-4. The water displacement of the fuel solution and subsequent flush of the compartment is a very good method (Oak Ridge) to decontaminate the shipping container. This is an excellent way to add the required amount of dilution water. The material in this tank is then adjusted to 30 grams of U per liter and made 2.93 molar in HCl. It is then fed to the first stage of solvent extraction at a constant rate of 2.64 gph by means of positive displacement pump J-1.

The solvent extraction system, using mixer-settlers, consists of three stages of extraction and two of stripping. The solvent used is 20% Alamine 336 in Xylene, and the stripping solution is 0.1 M  $\text{NH}_4\text{Cl}$ . Radioactive decay products remain in the aqueous phase while the  $\text{U}^{233}$  is extracted into the solvent. This "rich" solvent is then contacted by the stripping solution fed at a constant rate of 3.96 gpm by pump J-2. The U thereby passes into the stripping solution in purified form, as a  $\text{UO}_2\text{Cl}_2$  solution containing 20 grams of U per liter. This product is received in a 20-liter polyethylene bottle.

One compartment of a shipping container constitutes one batch for the solvent extraction system yielding 17-1/2 liters of purified U solution. Batch throughput time is 1 hour 10 minutes. The waste aqueous raffinate amounting to 11-1/2 liters is passed through an ion-exchange column for recovery of traces of U before being sent to waste.

c.  $\text{Th}(\text{NO}_3)_4$  -  $\text{UO}_2\text{Cl}_2$  Mixing

439 liters of thorium nitrate solution, containing 100 grams of thorium per liter and being 0.1 molar in  $\text{HNO}_3$ , are pumped from storage to mixing tank F-11. This amount of solution will furnish 50 kilograms of  $\text{ThO}_2$ .

The proper amount of purified  $\text{UO}_2\text{Cl}_2$  solution is then added to the same tank, to give the desired  $\text{UO}_2$  content. The  $\text{UO}_2\text{Cl}_2$  solution is added in three portions to avoid the possibility of exceeding its critical mass before dilution with  $\text{Th}(\text{NO}_3)_4$ .

The two solutions in the tank are well mixed by recirculation through pump J-7.

d. Coprecipitation

144 liters of 28% aqueous ammonia are pumped from storage into precipitation tank F-12. The agitator in this tank is then started and cooling water flow initiated in the cooling jacket.

The mixture in tank F-11 is pumped into the aqueous ammonia in F-12 causing precipitation of  $\text{Th}(\text{OH})_4$  and  $(\text{NH}_4)_2\text{UO}_2\text{O}_7$ .

e. Filtering

The coprecipitate,  $\text{Th}(\text{OH})_4$  and  $(\text{NH}_4)_2\text{U}_2\text{O}_7$ , is pumped from the precipitate tank to a modified rotary drum type filter, G-1, similar to a Walney-Lurgi filter. A porous endless belt serves as the filter cloth. The precipitate is placed on the moving belt and the filtrate is drawn out of the precipitate as the belt is passed over a vacuum chamber. A wash water spray washes the precipitate and is drawn through the precipitate into a second vacuum chamber. The filtrate and wash water are pumped to an ion-exchanger (Column No. 2) for U recovery. The precipitate is then conveyed to a rotary kiln (L-2). High pressure wash water is introduced into the hub of the "foot" pulley. As the foot pulley rotates, a surge of wash water enters a quadrant of the perforated cylinder and is used to clean out the pores of the filter cloth. The wash solution containing the residual precipitate is recycled through the filter.

f. Drying and Reduction

The precipitate is dried and reduced to the oxide in a specially designed rotary kiln (L-2). Laboratory experiments at NMI have shown the feasibility of this process. A similar piece of equipment, fabricated of stainless steel, is currently in use; however, operating temperature (700 - 900°C) obviates the use of this specific material.

2. Pellet Preparation

a. Blending and Pelletizing

The powdered oxide, to be pelletized, must be blended with a binder and lubricant. Future experimental work is desirable to verify experience of NMI wherein acceptable pellets were formed from the oxide with only the addition of water. The present design specifies a combination screw conveyor and blender for this operation. The blended powder is deposited into the feed hoppers at the pelletizer.

The equipment used in the production of  $\text{U}^{235}$  pellets is acceptable to this design. The pelletizer specifications are rather detailed, and reference should be made to Section D.



This unit operation is accomplished in a dry box specifically designed for this unit operation. Duplicate equipment each in its own isolation dry box is specified. Product throughput and equipment maintenance, during various material runs, is possible without interruption of product flow.

b. Sintering

The green pellets from pelletizing are guided into special boats required for the sintering operation. The sintering furnace (two are specified) is a semi-automatic muffle-type furnace fabricated in one piece but with three distinct sections, i.e., preheat, soak, and cool. The lower half of the furnace forms part of the dry box enclosure. The upper half of the furnace may be raised within the dry box to allow the servicing of furnace internals under controlled atmosphere conditions. An approximate throughput of 1500 pellets per hour is possible in each furnace. Purge chambers with double gas-tight doors are at each end of the furnace. The pellets are maintained under furnace atmosphere when inside the unit. Operation within the furnace is completely automatic. Boats must be loaded manually into the entrance purge chamber on signal and manually removed from the exit purge chamber on signal.

After a boat has passed through the furnace and is removed from the exit purge chamber, the operator inverts the boat onto a grid plate. The pellets and the emptied boat remain on the grid, and refractory powder (upon which the green pellets are placed) falls through the grid into a previously emptied boat. The boat that contains the refractory is placed upon a roller type conveyor and is returned to the front end of the furnace by gravity. The emptied boat is placed in position below the grid. The pellets slide down a chute into the bin of an orienting elevator. Pellets are oriented end to end and are fed into a V-block feed mechanism at the grinding machines.

c. Grinding

Visual inspection prior to grinding ensures that sintered pellets have the same approximate diameter. Broken or badly cracked pellets are removed from the product line.

Grinding consists of two operations, i.e., centerless grinding of the diameter to within pre-set limits, followed by separate end grinding to ensure squareness of each pellet end within pre-set limits.

Wet-belt grinders are specified for these operations. This type of machine has been used for this purpose and is presently in use at several installations.

The feed operation for grinding the diameter differs from that for grinding the ends. The discussion that follows is specifically the centerless grinding operation although a large portion is descriptive of the end grinding operations.

Inspected, sintered pellets are fed onto a carbide work-rest blade from a V-block type feeder. A regulating wheel drive mechanism maintains an automatically set pressure of the pellet against the revolving abrasive belt. The feed mechanism is controlled by a Federal Gage which also controls the regulating wheel drive mechanism. The gage continuously measures the ground pellets and automatically resets the grind operation when pellets approach a near high tolerance figure. The gage has a range of 0.003 inch with minimum graduation of 0.000050 inch. Three signal lights and an indicating dial gage indicate at all times the exact size of the pellet. The three lights indicate exact, near over, and near under tolerance limits.

A stainless steel collector trough and chute collects the drippings which are then conveyed into proper containers. The chute is so designed as to drain completely when the machine is washed between various material runs.

A plexiglas glove box is designed to fully enclose the machine and fits tightly against the stainless steel trough. Access ports in the cover and utility connections within the glove box are provided.

The automatic feed mechanism consists essentially of a specially designed jig which orients the pellets on diameter for end facing.

The ground pellets slide down a chute into an orienting elevator. The pellets are oriented end to end and channelled by a gravity feed into the loading dry box.

Criticality is not a problem since less than 1/2 a critical mass is handled in each batch. All equipment is interlocked to prevent the accumulation of material at any single point.

### 3. Assembly

#### a. Loading

The loading jig is set up such that the total number of pellets required for each fuel tube can be weighed. The tolerance limit on pellet loading (Con Ed elements) is  $\pm 1$  pellet. A fuel tube is oriented to the jig end and the pellets are then either drawn (by vacuum) or manually guided into the tube. All necessary accessories are inside the loading hood.

#### b. Welding and Inspection

End caps are welded into place at the ends of loaded fuel tubes in a helium atmosphere. The welding operation is fully automatic and is completed within a dry box.

The fuel rods are then transferred to another dry box for inspection. A conventional helium leak detector is used to determine weld quality. Where a bad weld is detected, corrective rewelding may be possible. The faulty end cap may, if necessary, be cut off and an alternate end cap used or the pellets may be returned to loading for reuse.

#### c. Assembly and Brazing

Inspected fuel rods are placed in appropriate jigs such that subassembly of a single fuel rod row may be fabricated. Spacers are placed on the fuel rod row in their proper positions and the entire subassembly is then brazed together. Subassemblies are stored until full

fuel element assembly requirements are reached. Subassemblies are then loaded into a fuel assembly jig. The fuel element assembly is placed in the brazing furnace, required hardware, by client specification, is then fixed into position, and the completed fuel element is put into a storage well or transported directly to the shipping area to be placed into a shipping cask.

The brazing furnace is an open end oven. Full-length brazing of subassemblies and fuel assemblies prevents any local overheating.

#### 4. Storage and Shipping

##### a. Storage

Completed fuel elements are stored dry in storage wells. The storage area is constructed to allow for flooding in the event that unusually long storage periods, upwards of one year, require additional shielding. The storage area is serviced by a 10-ton monorail-type crane.

Fixed radiation area monitors will detect any unusual radioactivity.

##### b. Shipping

Assembled fuel elements are removed from storage with the use of the 10-ton monorail crane and placed in a shipping cask in the adjacent area. The shipping casks are then sealed. The crane is next used to pick up the fuel element shipping cask for transfer to the shipping area. For loading, casks may be placed on dollies to facilitate truck or railway car loading. Shipping containers will have adequate shielding (approximately 3 inches of lead) for normal production schedules. Provision for temporary additional shield is incorporated for unusual radiation buildup due to long storage.

#### B. Site Requirements

Where a nuclear operation is contemplated, the plant location is of prime importance. Site selection is affected by all of the following:

- a. radiation hazards
- b. population factors
- c. radioactive waste storage
- d. transportation accessibility
- e. site cost and availability
- f. geology and seismology factors
- g. meteorology
- h. hydrology
- i. construction factors
- j. market location

The reaction hazards that are inherent in a  $U^{233}$  fuel element re-fabrication plant include:

1. sizeable quantities of alpha, beta and gamma emitting materials (approximately 100 kg fuel)
2. criticality
3. inhalation, ingestion and skin absorption.  
Process, equipment design, and conceptual design reflect adequate safety considerations.

Most certainly those site factors that dictate design for general chemical plants will apply. The design presented reflects location at a site similar to the NMI site in Concord, where related fuel element fabrication has been and is being done.

#### C. Building Considerations

The building is arranged in two segments: the office wing with a low ceiling and finished interior surfaces; and the work area wing with a high ceiling, exposed roof construction, and slight interior finish. The difference in roof height provides space for the heating and ventilation equipment. Expensive and unsightly roof structures for this purpose are eliminated.

Architecturally, the division between office and work area is visually expressed by the projecting stone veneered wall.

Cavity walls are specified for greater insulating value. This type wall will also lessen the possibility for condensation on interior wall surfaces. The filler panels above the masonry wall provide for addition of windows if this is desired. The use of sandwich panels shortens the apparent height of the masonry wall and gives a more human scale to the building.

#### Outline Specifications

- |                    |  |
|--------------------|--|
| Site Preparation   | - Grading: filling and backfilling as required.  |
| Soil Capacity      | - Footings and grade beams located on sound rock or soil of adequate bearing quality.  |
| Foundations        | - Reinforced concrete column footing and continuous wall footings (grade beam).  |
| Floor Construction | - Reinforced concrete slab on grade over compacted fill and gravel base with vapor barrier.  |
| Framing            | - Structural steel skeletal frame, riveted or welded connections.  |
| Walls, Exterior    | - Non-bearing masonry cavity walls or brick with concrete block back-up and filler panels of insulated cement asbestos sandwich panel. Entrance wall of native fieldstone on concrete block back-up. |
| Walls, Interior    | - Concrete or gypsum block, with offices to receive two coats of hard white gypsum plaster; other surfaces to be painted. Toilet room to receive glazed tile wainscot 5' - 4" high.                  |
| Ceilings           | - Hung acoustical tile in offices, corridor and vestibule. All other to have exposed roof construction.  |

- Windows - Aluminum fixed sash in office wing.
- Roof - Metal decking on steel bar joists spanning steel girders, with rigid insulation and 4-ply built-up roofing.
- Floor Finish - Vinyl asbestos in offices, corridor, and vestibule. Quarry tile in toilets and liquid hardened cement finish in all other areas.

An area allowance is made in design for alternate unit operations, such as vibratory compaction or swaging rather than pelletizing.

D. Equipment Arrangement and Specifications

Several different equipment arrangements were considered. The recommended design that has been developed reflects these advantages:

1. Straight-through flow of material.
2. Consecutive unit operations.
3. Semi-continuous flow conditions. (Batches are limited to less than 1/2 critical mass.)
4. Parallel equipment and material flow control where necessary.
5. Dual purpose areas where possible.
6. Adequate supply storage areas.
7. Double radiation-type safety enclosures.
8. Air-locks separating each unit operation and parallel equipment line.
9. Flexibility in material quantity flow.
10. Adequate product storage area and handling techniques.
11. Multi-purpose equipment.

<u>Pump No.</u>	<u>Application</u>	<u>Process Fluid</u>	<u>Sp.G.</u>	<u>GPM</u>	<u>Type</u>	<u>Material of Construction</u>
J.1	SX-Feed	UO <sub>2</sub> (NO <sub>3</sub> ) <sub>2</sub> +HNO <sub>3</sub> +HCl Sol'n	1.088	0.044	Posit. Displ.	Plastic Diaphr.
J.2	Stripping Sol'n Feed	0.5% NH <sub>4</sub> Cl Sol'n	1.00	0.066	Posit. Displ.	Rubber
J.3	Raffinate Pump	HNO <sub>3</sub> +HCl Sol'n	1.06	0.043	Posit. Displ.	Plastic Diaphr.
J.4	Eluate Transfer	UO <sub>2</sub> Cl <sub>2</sub> Sol'n	1.0	1	Horiz. Centrif.	Polyeth.
J.5	Waste Pump	HNO <sub>3</sub> +HCl Sol'n	1.06	1	Horiz. Centrif.	Polyeth.
J.6	Th-Nitrate Transfer	Th(NO <sub>3</sub> ) <sub>4</sub> +HNO <sub>3</sub> Sol'n	1.18	15	Horiz. Centrif.	304 S.S.
J.7	Th-U Mix Transfer	Th+U in HNO <sub>3</sub> Sol'n	1.16	15	Subm.Vert.Centr.	304 S.S.
J.8	Filter Feed	Th(OH) <sub>4</sub> +ADU Slurry	1.09	5	Horiz. Centrif.	304 S.S.
J.9	Waste Feed to I-X	NH <sub>4</sub> NO <sub>3</sub> +NH <sub>4</sub> Cl+NH <sub>3</sub> Sol'n	1.016	1	Horiz. Centrif.	304 S.S.
J.10	Waste Transfer	NH <sub>4</sub> NO <sub>3</sub> +NH <sub>4</sub> Cl+NH <sub>3</sub> Sol'n	1.016	15	Horiz. Centrif.	304 S.S.
J.11	Ammonia Transfer	28% Aqueous NH <sub>3</sub>	0.897	15	Horiz. Centrif.	C.S.
J.12	Eluate Transfer	UO <sub>2</sub> (NO <sub>3</sub> ) <sub>2</sub> Sol'n	1.0	1	Horiz. Centrif.	304 S.S.

1. Pump Schedule



J.1 Solvent Extraction Feed Pump

Number required: 1

Product handled: aqueous solution  $\text{UO}_2(\text{NO}_3)_2 + \text{HNO}_3 + \text{HCl}$

Corrosion due to  $\text{HNO}_3 + \text{HCl}$

Pumping temperature:  $68^\circ\text{F}$

Sp. Gr. at pump temperature: 1.088

Capacity: 2.64 GPM

Normal GPM at pump temperature: 2.64

Motor Drive

Manufacturer: Builders Providence Proportioneer

Size and Type: Chlo-O-Feeder

Number of Cylinders: 1

J.2 Stripping Solution on Feed Pump

Number required: 1

Product handled: 0.5%  $\text{NH}_4\text{Cl}$  solution

Corrosion due to  $\text{NH}_4\text{Cl}$

Pumping temperature:  $68^\circ\text{F}$

Sp. Gr. at  $60^\circ\text{F}$ , 1.0; at pump temperature, 1.0

Viscosity: 1.0 Cps

Capacity: 2.96 GPM

Normal GPM at pump temperature: 3.96

Motor drive

Manufacturer: Builders Providence Proportioneer

Size and Type: Chlo-O-Feeder

Number of Cylinders: 1

J.3 Raffinate Pump

Number required: 1

Product handled: aqueous  $\text{HCl} + \text{NH}_3$

Corrosion due to  $\text{HCl} + \text{HNO}_3$

Pumping temperature:  $68^\circ\text{F}$

Capacity: 2.60 GPH

Normal GPM at pump temperature: 2.60

Motor drive

Manufacturer: Builders Providence Proportioneer

Size and type: Chlor-O-Feeder

J.4 Eluate Transfer Pump

Number required: 1

Normal capacity GPM: 1

Product handled:  $\text{UO}_2\text{Cl}_2$  solution

Corrosion due to chloride

Pumping temperature:  $68^\circ\text{F}$

Sp. Gr. at pump temperature: 1.0

Suction pressure: ATM

Casing: polyethylene lined

Impeller: polyethylene lined

J.5 Waste Pump

Number required: 1

Normal capacity GPM: 1

Product handled:  $\text{HNO}_3 + \text{HCl}$  solution

Corrosion due to  $\text{HNO}_3 + \text{HCl}$

Pumping temperature:  $68^\circ\text{F}$

Sp. Gr. at pump temperature: 1.06

Suction pressure: ATM

Casing: polyethylene-lined C.S.

Impeller: polyethylene-lined C.S.

J.6 Th-Nitrate Transfer

Number required: 1

Normal capacity GPM: 15

Product handled:  $\text{Th}(\text{NO}_4)_3 + \text{HNO}_3$  solution

Corrosion due to  $\text{HNO}_3$

Pumping temperature:  $68^\circ\text{F}$

Sp. Gr. at pumping temperature: 1.18

Service: intermittent  
Casing: 304 stainless steel  
Impeller: 304 stainless steel

J.7 Th-U Mix Transfer

Number required: 1  
Normal Capacity GPM: 15  
Product handled: Th + U in  $\text{HNO}_3$  solution  
Corrosion due to  $\text{HNO}_3$   
Pumping temperature:  $68^\circ\text{F}$   
Sp. Gr. at pumping temperature: 1.16  
Service: intermittent  
Type: submerged vertical  
Casing: 304 stainless steel  
Impeller: 304 stainless steel

J.8 Filter Feed

Number required: 1  
Normal capacity GPM: 5  
Product handled:  $\text{Th}(\text{OH})_4$  + A.D.U. slurry  
Corrosion due to  $\text{NH}_4\text{NO}_3$  +  $\text{NH}_4\text{Cl}$   
Pumping temperature:  $90^\circ\text{F}$   
Sp. Gr. at pumping temperature: 1.09  
Service: intermittent  
Casing: 304 stainless steel  
Impeller: 304 stainless steel

J.9 Waste Feed to IX

Number required: 1  
Normal capacity GPM: 1  
Product handled:  $\text{NH}_4\text{NO}_3$  +  $\text{NH}_4\text{Cl}$  +  $\text{NH}_4\text{OH}$  solution  
Corrosion due to  $\text{NH}_4\text{Cl}$   
Pumping temperature:  $90^\circ\text{F}$

Sp. Gr. at pumping temperature: 1.01  
Service: 4 -hour continuous pumping period in 8 hr  
Casing: 304 stainless steel  
Impeller: 304 stainless steel

J.10 Waste Transfer

Number required: 1  
Normal capacity GPM: 15  
Product handled:  $\text{NH}_4\text{NO}_3 + \text{NH}_4\text{Cl} + \text{NH}_4\text{OH}$  solution  
Corrosion due to  $\text{NH}_4\text{Cl}$   
Pumping temperature:  $70^\circ\text{F}$   
Sp. Gr. at pumping temperature: 1.016  
Service: intermittent  
Casing: 304 stainless steel  
Impeller: 304 stainless steel

J.11 Ammonia Transfer

Number required: 1  
Normal capacity GPM: 15  
Product handled: 28% aqueous ammonia  
Pumping temperature:  $68^\circ\text{F}$   
Sp. Gr. at pumping temperature: 0.897  
Viscosity Cps: 1.2 at  $68^\circ\text{F}$   
Service: intermittent  
Casing: carbon steel  
Impeller: carbon steel

J.12 Eluate Transfer

Number required: 1  
Normal capacity GPM: 1  
Product handled:  $\text{UO}_2(\text{NO}_3)_2$  solution  
Pumping temperature:  $68^\circ\text{F}$   
Sp. Gr. at pumping temperature: 1.0  
Casing: 304 stainless steel  
Impeller: 304 stainless steel

<u>Tank No.</u>	<u>Application</u>	<u>Process Fluid</u>	<u>Work'g Vol. Gal.</u>	<u>Total Vol. Gal.</u>	<u>Agitator</u>	<u>Material of Construction</u>
F.1	Demin. Water Head Tank	Demin. Water	50	55		304 S.S.
F.2	HCl Head Tank	36% HCl	50	55		R.L. - C.S.
F.3	HCl Measuring Tank	36% HCl	0.8	1		Polyeth.
F.4	SX-Feed Adjustment	UO <sub>2</sub> (NO <sub>3</sub> ) <sub>2</sub> +HNO <sub>3</sub> +HCl Sol.	3.1	5	Req'd	Polyeth. C.S.
F.5	Stripping Sol'n Make-up	0.5% NH <sub>4</sub> Cl Sol'n	160	200	Req'd	R.L. - C.S.
F.6	Raffinate Receiver	HNO <sub>3</sub> +HCl Sol'n	3.1	5		Polyeth.-C.S.
F.7	Wash Sol'n Make-up	20% HCl	1.6	2		Polyeth.
F.8	Eluate Receiver	UO <sub>2</sub> Cl <sub>2</sub> Sol'n	4	5		Polyeth. -C.S.
F.9	Waste Receiver	HNO <sub>3</sub> +HCl Sol'n	3.1	5		Polyeth. -C.S.
F.10	Th-Nitrate Storage	Th(NO <sub>3</sub> ) <sub>4</sub> +HNO <sub>3</sub> Sol'n	4,900	6,000		304 S.S.
F-11	Th-U Mixing Tank	Th+U in HNO <sub>3</sub> Sol'n	133	175		304 S.S.
*F-12	Precipitation Tank	Th(OH) <sub>4</sub> +A.D.U. Slurry	170	250	Req'd	304 S.S.
F-13	Filtrate Receiver	NH <sub>4</sub> NO <sub>3</sub> +NH <sub>4</sub> Cl+NH <sub>3</sub> Sol'n	266	300		304 S.S.
F-14	Eluate Receiver	UO <sub>2</sub> (NO <sub>3</sub> ) <sub>2</sub> Sol'n	4	5		304 S.S.
F-15	Waste Receiver	NH <sub>4</sub> NO <sub>3</sub> +NH <sub>4</sub> Cl+NH <sub>3</sub> Sol'n	266	300		304 S.S.
F-16	Ammonia Storage	28% Aqueous NH <sub>3</sub>	1,600	2,500		C.S.

\* Cooling jacket required.

2. Tank Schedule

**F.1 Demineralized Water Head Tank**

Number required: 1  
Process fluid: demineralized water  
Working volume, gal.: 50  
Max. weight, lb.: 460  
Sp. Gr.: 1  
Viscosity, Cps: 1  
Working temperature: 68<sup>o</sup>F  
Operating pressure, psig: ATM  
Heat transfer: None  
Heads: 304 stainless steel  
Shell: 304 stainless steel

**F.2 HCl Head Tank**

Number required: 1  
Process fluid: 36% HCl  
Working volume, gal.: 50  
Max. weight, lb.: 540  
Sp. Gr.: 1.18  
Viscosity, Cps: 2  
Working temperature: 68<sup>o</sup>F  
Operating pressure, psig: ATM  
Heat transfer: None  
Agitation: None  
Heads: rubber-lined C.S.  
Shell: rubber-lined C.S.

**F.3 HCl Measuring Tank**

Number required: 1  
Process fluid: 36% HCl  
Working volume, Gal.: 0.77  
Max. weight, lb.: 10  
Sp. Gr.: 1.18

Viscosity, Cps: 2  
 Working temperature: 68°F  
 Operating pressure, psig: ATM  
 Heat transfer: none  
 Agitation: none  
 Heads: polyethylene  
 Shell: polyethylene

#### F.4 SX Feed Adjustment Tank

Number required: 1  
 Process fluid: aqueous solution  $\text{UO}_2(\text{NO}_3)_2 + \text{HCl}$   
 Working volume, gal.: 3.1  
 Always safe geometry: 3.7" diameter  
 Max. weight, lb.: 45  
 Sp. Gr.: 1.08  
 Working temperature: 68°F  
 Operating pressure, psig: ATM  
 Agitation: required  
 Heads: polyethylene-lined C.S.  
 Shell: polyethylene-lined C.S.  
 Vessel No.: F-4  
 Shape of tank: vertical cylinder  
 Internal working pressure, psig: ATM  
 Total tank capacity: gal: 5  
 Working capacity, gal.: 3.1  
 Shaft: polyethylene-lined C.S.  
 Impellers: polyethylene-lined C.S.

Components	Rel.%	Sp. Gr. at	°F
6.3% $\text{UO}_2(\text{NO}_3)_2$ Sol'n	75 Vol %	1.058	68
36% HCl	25 Vol %	1.180	68

Sp. Gr. of final product: 1.09

F.5 Stripping Solution Make-Up and Feed Tank

Number required: 1

Process fluid: 0.5%  $\text{NH}_4\text{Cl}$  solution

Working volume, gal.: 160

Max. weight, lb.: 1330

Sp. Gr.: 1.0

Viscosity, Cps: 1.0

Working temperature: 68°F

Operating pressure, psig: ATM

Heat transfer: None

Agitation required

Heads: rubber-lined C.S.

Shell: rubber-lined C.S.

Vessel No.: F-5

Shape of tank: vertical cylinder

Internal working pressure, psig: ATM

Total tank capacity, gal.: 200

Working capacity, gal.: 160

Shaft: rubber-lined C.S.

Impellers: rubber-lined C.S.

Components	Rel. %	Sp.Gr.	Viscosity at	°F
0.5% $\text{NH}_4\text{Cl}$ Sol'n	100	1.0	1.0	68

Sp. Gr. of final product: 1.0

Viscosity of finished mix: 1.0 Cps at mixing temp. of 68°F

Solids handled: crystalline

Sp. Gr. of solids: 1.53

F.6 Raffinate Receiver

and

F.9 Waste Receiver

Number required: 2

Process fluid:  $\text{HNO}_3$  + HCl solution

Working volume, gal.: 3.1

Max. weight, lb.: 440

Sp. Gr.: 1.06

Working temperature: 68°F



Operating pressure, psig: ATM  
Heat transfer: none  
Heads: polyethylene-lined C.S.  
Shell: polyethylene-lined C.S.

F.7 Wash Solution Make-Up Tank

Number required: 1  
Process fluid: 20% HCl  
Working volume, gal.: 1.6  
Max. weight, lb.: 18.3  
Sp. Gr.: 1.10  
Working temperature: 68°F  
Operating pressure, psig: ATM  
Heat transfer: none  
Heads: polyethylene  
Shell: polyethylene

F.8 Eluate Receiver

Number required: 1  
Process fluid:  $\text{UO}_2\text{Cl}_2$  solution  
Working volume, gal.: 4  
Max. weight, lb.: 42  
Sp. Gr.: 1.0  
Viscosity, Cps: 1.0  
Working temperature: 68°F  
Operating pressure, psig: ATM  
Heat transfer: none  
Agitation: none  
Heads: polyethylene-lined C.S.  
Shell: polyethylene-lined C.S.

F.10 Thorium-Nitrate Storage Tank

Process fluid:  $\text{Th}(\text{NO}_3)_4 + \text{HNO}_3$  solution  
Working volume, gal.: 4,900  
Max. weight, lb.: 59,000  
Sp. Gr.: 1.18

Working temperature: 68°F  
 Operating pressure, psig: ATM  
 Heat transfer: none  
 Agitation: none  
 Heads: 304 stainless steel  
 Shell: 304 stainless steel

F.11 Th-U Mixing Tank

Number required: 1  
 Process fluid: Th + U-233 in HNO<sub>3</sub> solution  
 Working volume, gal.: 133  
 Max. weight, lb.: 1,700  
 Sp. Gr.: 1.16  
 Working temperature: 68°F  
 Operating pressure, psig: ATM  
 Heat transfer: None  
 Heads: 304 stainless steel  
 Shell: 304 stainless steel

F.12 Precipitation Tank

Number required: 1  
 Process fluid: Th(OH)<sub>4</sub> + ADU slurry  
 Working volume, gal.: 170  
 Max. weight, lb.: 2,300  
 Sp. Gr.: 1.095  
 Working temperature: 68°F  
 Operating pressure, psig: ATM  
 Heat transfer: jacket required  
 Agitation required  
 Heads: 304 stainless steel  
 Shell: 304 stainless steel  
Precipitation Tank Agitator

Number required: 1  
 Vessel No.: F-12  
 Internal working pressure, psig: ATM  
 Total tank capacity, gal.: 250

Working capacity, gal.: 171

Shaft: 304 stainless steel

Impeller: 304 stainless steel

Components	Vol. Rel. %	Sp. Gr.	at	°F
Th(NO <sub>3</sub> ) <sub>4</sub> + Cl <sub>2</sub> solution	78	1.16		68
Aqueous NH <sub>3</sub> (14M)	22	0.897		68

Sp. Gr. of final product: 1.095

Solids handled: insoluble

Lb. solids/gal. mixt: 0.75

Class of operation: batch

Degree of mixing: complete

F.13 Filtrate Receiver  
and

F.15 Waste Receiver

Number required: 2

Process fluid: NH<sub>4</sub>NO<sub>3</sub> + NH<sub>4</sub>Cl + NH<sub>3</sub> solution

Working volume, gal.: 266

Max. weight, lb.: 2540

Sp. Gr.: 1.016

Working temperature: 68°F

Operating pressure, psig: ATM

Heat transfer: none

Agitation: none

Heads: 304 stainless steel

Shell: 304 stainless steel

F.14 Eluate Receiver

Number required: 1

Process fluid: UO<sub>2</sub>(NO<sub>3</sub>)<sub>2</sub> solution

Working volume, gal.: 4

Max. weight, lb.: 42

Sp. Gr.: 1.0

Viscosity, Cps: 1.0

Working temperature: 68°F

Operating pressure, psig: ATM

Heat transfer: none  
 Agitation: none  
 Heads: 304 stainless steel  
 Shell: 304 stainless steel

F.16 Aqueous Ammonia Storage

Number required: 1  
 Process fluid: 28.6% NH<sub>3</sub>  
 Working volume, gal.: 1,600  
 Max. weight, lb.: 20,800  
 Sp. Gr.: 0.897  
 Viscosity, Cps: 1.2  
 Working temperature: 68<sup>o</sup>F  
 Operating pressure, psig: ATM  
 Heat transfer: none  
 Agitation: none  
 Heads: carbon steel  
 Shell: carbon steel

G.1 Th(OH)<sub>4</sub> + ADU Filter

Number required: 1  
 Filter for separating out the insoluble product in the following feed and thoroughly washing all soluble salts out of the cake.

Feed Batch:

		<u>Lb.</u>	<u>Weight %</u>
Insoluble) Products )	Th(OH) <sub>4</sub>	125.1	8.05
	(NH <sub>4</sub> ) <sub>2</sub> U <sub>2</sub> O <sub>7</sub>	3.83	0.25
	NH <sub>4</sub> NO <sub>3</sub>	141.7	9.11
	NH <sub>4</sub> Cl	2.12	0.14
	NH <sub>3</sub>	50.3	3.24
	H <sub>2</sub> O	<u>1,230.</u>	<u>79.21</u>
	TOTAL	1,553.	100.00

Sp. Gr.: 1.095  
 Volume: 170 Gal.

### D.1 Ion-Exchange Column

Number required: 1

Ion-exchange column containing 1-1/2 liter of Rohm and Haas Resin IRA-400 of 50-100 mesh size to recover uranium from the following solution:

Feed Solution:

U	.003g/l
HNO <sub>3</sub>	0.15 Molar
HCl	2.93 Molar
Th-228, Ra-224, Pb-212 Impurities Present	
Rate	164 ml/mm

### D.2 Ion-Exchange Column

Number required: 1

Ion-exchange column containing 1-1/2 liter of proper type resin to recover uranium from the following alkaline waste solution:

<u>Feed Solution:</u>	<u>Weight %</u>	<u>g/l</u>
U	Trace	
NH <sub>4</sub> NO <sub>3</sub>	6.29	63.8
NH <sub>4</sub> Cl	0.09	0.95
NH <sub>3</sub>	2.24	22.6
H <sub>2</sub> O	<u>91.38</u>	
	100.00	

Feed Rate: 1GPM

Sp.Gr.: 1.016

### L.1 Mixer-Settler

Number required: 1

5-stage mixer-settler made of polyethylene-lined carbon steel or other material resistant to process liquids below. Each mixer-settler cell to have a capacity of about 1 liter. Three stages to be for extraction and 2 for stripping. Explosion-proof agitators.

Feed to Extraction

$UO_2(NO_3)_2$	50 g/l
$HNO_3$	0.15 Molar
HCl	2.93 Molar
Th-288, Ra-224, PB-212 impurities	
Rate	167 ml/min

Solvent

20% Alamine-336 in Xylene

Rate: 190 ml/min

Stripping Solution0.1 Molar  $NH_4Cl$ 

Rate: 250 ml/min

B.1 Rotary Kiln

Number required: 1

Semi-continuous operation

Four-stage combination drying and reduction kiln

Feed Batch:

$Th(OH)_4$	56,800 g
$(NH_4)_2U_2O_7$	1,738 g
$H_2O$	<u>168,700 g</u>
	227,238 gms

## Operating Temperatures

Stage 1 - 300-400°C	Drying
Stage 2 - 500-600°C	Preheat
Stage 3 - 700-900°C	Reduction
Stage 4 - 300°C	Cooling

Material of Construction - Vycor or Inconel

Overall length: 4'-0"

Diameter: 2'-0"

B.2 Sintering Furnace

Number required: 2

Design temperature: 1750°C

Operating temperature: 1600°C

Automatic hydrogen-nitrogen purge cycles at entrance and exit purge chamber.

Fully instrumented for automatic operation.

Overall height: 7'-6"

Overall width: 4'-3"

Overall length: 26'-0"

Hearth opening: 7" x 6"

Zone 1 - 6' - preheat

Zone 2 - 6' - soak

Zone 3 - 6' - cooling

Harper Electric Furnace Corporation Ref. No. 20661 or equal.

### B.3 Brazing Furnace

Number required: 2

Design temperature max.: 2050°F

Operating temperature: 1600-1800°F

Open end, single zone, pusher type.

Temperature indicator and controller.

Overall length: 10'-0"

Overall height: 7'-0"

Overall width: 5'-0"

Hearth opening: 8" x 8"

Electrical Rating: 50 kw.

Sargeant and Wilbur No. PAC080833 or equal.

### L.2 Pelletizer

Number required: 2

Specification:

Output: 12 to 25 tablets per minute

Maximum diameter of tablet: 1-1/2 in. (38.1 mm.)

Maximum depth of fill: 1-7/8 in. (47.6 mm.)

Maximum working pressure: 10 tons for 57 lbs.

per inch on piston

Maximum compression stroke of upper punch:

0 to 1 inch (0-25.3 mm.)

Maximum compression stroke of lower punch:

0 to 9/16 in. (0-14.2 mm.)

Distance punch face to die table (stroke up,

adjustment up): 3 in. (76.1 mm.)

Air supply: 80 lb. per sq. inch.  
 Height of machine: 6 ft. (182.8 cm.)  
 Floor space: 3 ft. 7 in. x 2 ft. 8 in.  
 (109.2 x 81.2 cm.)  
 Case dmen.: 6 ft. 6 in. x 4 ft. 2 in. x  
 3 ft. 3 in. (198.1 x 127.0 x 99.0 cm.)  
 Weight: 31 cwt. (1578 kilos.)  
 Manesty Model 2 C or equal.

### L.3 Grinding Machine

Number required: 3  
 Wet-belt grinder  
 Belt 4" wide x 54" long  
 Regulating wheel drive mechanism  
 Micrometer in feed with indicator dial  
 Contact wheel  
 Carbide work rest 3/16" wide  
 Drive unit: Sanitary motor - 5 HP, 3 phase,  
 220/440V, 60 cycles.  
 Automatic feed mechanism  
 Fully enclosed with plexiglas dry box  
 Series 493 Federal gage or equal.  
 Sundstrand - Engelburg Atomic Model L-4 or equal.

### L.4 Orienting Belt Elevator

Number required: 3  
 Automatic feeding equipment to grinding and loading  
 Loading height: 42" std.  
 Capacity: 5'-3"  
 Elevating angle: 15°  
 Power: 1/2 H, 3 phase, 220/440V, 1800 rm.  
 R C A Model - OBE-350 or equal.



#### L.5 Shipping Container

Number required: 36

Tricompartmented containers as per drawing No. 1382-R-2

Inside diam. max.: 3.7"

Height: 48"

Material of Construction: S/S

Shielding: 2" lead

Container to be permanently fixed to a shipping frame  
24" on each side and approximately 60" high.

Container shall be equipped with fittings and tubing  
as shown on the referred drawing.

#### L.6 Ammonia Dissociator

Number required: 1

Rated output: 1000/cfh.

Electrical rating: 27 kw.

Ammonia consumption: 22.2 lbs./hr.

Floor space requirements: 7'-0" x 4'-0"

Sargeant and Wilbur 1000 cfh Ammonia Dissociator or equal.

### E. Auxiliary Services

#### 1. Heating, Ventilating and Air Conditioning

##### a. Design Conditions

The latest edition of the American Society of Heating, Refrigerating and Air Conditioning Guide shall be the basic reference of design for heating, ventilating and air conditioning systems. Heating and cooling load calculations shall be prepared using the procedure established by the ASHRAC.

The process room and office areas cooling loads shall be calculated to maintain an inside condition of  $80^{\circ}\text{F} \pm 2^{\circ}$  with outside summer design of  $95^{\circ}\text{F}$  DB and  $76^{\circ}\text{F}$  WB. Heating loads shall be calculated to maintain an inside condition of  $70^{\circ}\text{F} \pm 2^{\circ}$  with outside winter design of  $0^{\circ}\text{F}$ . In summer the relative humidity shall be limited to approximately 50% and in winter

no humidification will be provided. Cooling loads shall be based on installed lighting and equipment loads. The equipment room heating and cooling load shall be calculated to maintain  $65^{\circ}\text{F} \pm 5^{\circ}\text{F}$  in winter and a maximum of  $105^{\circ}\text{F}$  in summer. Radiators shall be provided under the windows in the offices and in any other area where sedentary operations are to be performed.

Where fume hoods are provided, a constant inward air flow, equivalent to 150 fpm for a 12-inch high sash opening for the full width of the hood, shall be used.

b. Building Air Flow Pattern

The design of the heating, ventilating and air conditioning systems provides a normal integrated conventional system as well as an air flow pattern to assist in the containment of contamination within each area. Each part of the air handling system must provide complete compatibility under all operating conditions with other parts of the air handling system.

In general, air shall be provided to each area to maintain temperatures. Air that is provided for the offices shall be exhausted through the corridor and toilets. Air that is provided for the process room shall be exhausted through the cell, glove boxes, and hoods.

Static pressure control will keep the process room pressure negative with respect to the offices.

Fresh air intake for the building shall be located above the low roof.

c. Building Systems

One air supply system shall be provided to supply 100% fresh, filtered, heated or cooled air to the offices and hot change room as shown on Drawing 1382-K-1 (Fig. 14) as System K-1. Supply air system shall contain motor-operated inlet damper, non-freeze preheat coil, prefilter, 80% efficient filters as determined by the National Bureau of Standards' Discoloration Test with Underwriters' Class I rating replaceable media, fan with radial inlet vanes, cooling coil and steam reheat coils. Air shall be distributed to the various areas through a low-velocity duct system. All supply air shall be exhausted through high-efficiency filters to the roof.

One air supply system shall be provided to supply 100% fresh, filtered, heated or cooled air to the process room and air lock as shown on Drawing 1382-K-1 (Fig. 14) as System K-2. Supply air system shall contain motor-operated inlet damper, non-freeze preheat coil, 8% efficient prefilters, and 80% efficient filters as determined by the National Bureau of Standards' Discoloration Test with Underwriters' Class I rating replaceable media, fan with radial inlet vanes, cooling coil, and steam reheat coils. Air shall be distributed through a low-velocity duct system.

All air supplied to the process room shall be exhausted through the cell, glove boxes, and hoods as indicated on Drawing 1382-K-1 (Fig. 14) as System K-3. Separate ducts shall run to the exhaust filter plenum for the following:

Sintering furnace offgas.

Cell, hoods, and rooms.

Glove boxes.

All air, except the sintering furnace offgas, entering the exhaust duct shall be filtered as indicated. All air from the exhaust ducts shall pass through high-efficiency filters before release to the stack.

The exhaust fan and standby exhaust fan shall have radial inlet vanes.

d. Heating and Air Conditioning Controls

Control system shall be pneumatic. Local indication of filter pressure drops, system static pressures, flow rates in exhaust headers, chilled water pressure, steam pressure and air temperature shall be provided.

System K-1 shall maintain room temperature by modulating valves for preheat, cooling, and reheat coils in sequence. Hot Change Room static pressure shall be maintained by modulating the fans radial inlet vanes. Supply and exhaust fans shall be electrically interlocked so that if the supply fan fails, the exhaust fan will stop.

System K-2 shall maintain temperature by modulating valves for preheat, cooling, and reheat coils in sequence. Process Room static pressure shall be maintained by modulating the supply fans radial inlet vanes.

System K-3 exhaust fan shall maintain static pressure upstream of the high-efficiency filters by modulating the fan radial inlet vanes.

## 2. Electrical

The power distribution system for process equipment, lighting and miscellaneous electrical loads to the building is shown on Drawing 1382-N-1 (Fig. 15).

Main power distribution in the building will be 480 volt, 3 phase, 60 cycles.

Power distribution for lighting and receptacle circuits, and small motor loads up to 1/2 HP will be 120/208 volts, 3 phase, 4 wire, 60 cycles.

The normal incoming power supply will be provided from a 3-phase, 500 KVA oil-filled transformer. The high voltage winding of the transformer will be suitable for use at the plant primary distribution voltage. The transformer will be located outside of the building adjacent to the Utility Room with a throat connection through the wall to 480-volt switchgear inside of the building.

The power supply for 120/208 lighting receptacles and miscellaneous power circuits will be provided from a 75 KVA, 3 phase, 480-120/208 volt transformer which will be supplied from one of the feeders in the 480-volt switchgear.

Power for emergency lights, radiation monitoring, instrument and safety interlocking systems will normally be supplied from the 75 KVA transformer. In the event of failure of the normal power supply to the building, the feed to this equipment will automatically transfer to an emergency source. The emergency supply will be connected to a firm source which is not affected by loss of the normal supply circuit.

## 3. Instrumentation

Instrumentation for the U-233 Fuel Element Refabrication Facility requires the instruments and controls necessary to monitor and operate the chemical, mechanical and health physics portions of this facility.

The chemical process involves very small batch flows. Instrumentation is basically local indication with supervisory indicating light and alarm monitoring of extreme process conditions. Batch make-up flows will be metered by small measuring tanks or volumetric totalizers which automatically shut off flow after a predetermined volume has been metered. Feed rates of the positive displacement pumps have local manual adjustment for the various batch concentrations.

Remote storage tank level indicators, the supervisory indicating lights and alarms, and the agitator and pump control switches are grouped on a main control panel for convenience and ease of operation. A "tote board" presentation of the operating condition of each piece of equipment shall be provided and located near the SX and IX hot cell. This will permit the operator to follow the progress of each batch readily through the process.

The rotary kiln and sintering furnace are fully automatic with zone temperature control on each of the four stages. An automatic trip, manual reset, safety shutdown on high pressure is provided for the hydrogen feed. This will shut down the hydrogen flow in the event of an explosion or blowback within these equipments.

The doors into the SX and IX hot cell are interlocked so that only one may be opened at a time. This prevents potentially contaminated cell air from mixing with the process area atmosphere.

The mechanical operations of this facility are an assembly of automatic, unit-operation machines. The conveyors which interconnect these various machines and the machines themselves are interlocked to prevent pileup of the pellets in the event of equipment malfunction. Critical running lights and malfunction indication shall be incorporated into the "tote board" on the main control panel.

Health physics encompasses the monitoring of this facility for alpha, gamma and neutron radioactivity. Monitoring shall be largely portable (hand and foot counter in the change room, personnel dosimeter and film badges, alpha and gamma portable survey meters, etc.) except for specific detection points for which alarm and indication or recording shall be on a section of the main control panel. Individual remote

alarms shall be provided in the health physics office. A neutron detector shall be located within the SX and IX hot cell, and its output shall be recorded.

The solvent extraction offgas and the hood and glove box exhaust shall have particulate sample monitoring just prior to its leaving the building. These samples shall be monitored for alpha and gamma radiation. Radiogas monitoring is not required. Individual filters in each hood or glove box exhaust duct shall be monitored for accumulation of radioactivity with portable meters.

An area (gamma) monitor detector shall be located on the inside of each wall of the process area. Additional detectors shall be located in the shipping air lock and in the change room to monitor all movement of personnel and equipment in and out of the process area.

#### 4. Decontamination and Cleanup

The radioactive contamination of the equipment line within the boxes is held to a minimum. This is accomplished by the choice of equipment and by the flow of air through the dry boxes as shown on Drawing 1382-K-1 (Fig. 14), and described in more detail in Section E,1,c. However, radioactive contamination will occur and provision is made for the routine decontamination and cleanup of equipment when necessary, e.g., between different batches of fuel pellets produced.

Each dry box is fabricated with service line penetrations permanently sealed to the dry box wall. The penetrations are of the quick disconnect type with safety valves on either side of the wall. The general services which may be connected when needed are compressed air, water, detergent, and electrical inputs. These connections are made at appropriate times only. Much of the equipment has been selected to allow for ease of cleaning, e.g., sanitary motors. Equipment layout allows the drainage of wash and rinse solutions to waste receivers. Crevices where dust and dirt may accumulate are held to a minimum. Where gross contamination may accumulate or where major equipment overhaul is required, the piece in question may be removed (with its protective dry box) to the decontamination area for more exacting work. Duplicate equipment exists at more troublesome spots to allow continued production during maintenance and cleanup operations.

## 5. Waste Disposal

The wastes generated in this facility will consist of solid, gaseous and liquid wastes that contain the radioactive daughter products. Disposal of only those wastes from the manufacturing building proper will be considered. Those wastes which occur as a result of chemical analysis and quality control will be disposed in accordance with existing practices at the analytical or quality control laboratories associated with the fabrication facility. Solid wastes will be collected and packaged for off-site disposal. Gaseous wastes, which are expected to be only slightly radioactive, will be dispersed to the atmosphere. Liquid wastes are segregated at the source. High level liquid wastes, i.e., those wastes containing more than  $10^{-2}$   $\mu\text{c/ml}$  will consist of small volumes of liquids and will be collected in special, shielded 50-gallon drums.

Low level wastes, containing less than  $10^{-4}$   $\mu\text{c/ml}$  are generated at about the rate of 30 gallons/day. They are collected in 2 tanks, each one of which has a capacity of 200 gallons. The low level liquid wastes will be sampled before disposal. If the wastes can be safely disposed of by dilution and dispersal, as is possible, this will be done.

Temporary radioactive waste storage is provided for. No permanent radioactive waste storage is contemplated. High level wastes, as they accumulate, will be packaged and either returned to Oak Ridge for reprocessing or transported to an established radioactive waste storage area.

## F. Architectural and Engineering Specifications

### Contents

1. Building Description
2. Site Development
3. Architectural and Structural
4. Electrical
5. Sanitary Plumbing and Storm Drainage
6. Fire Protection
7. Piping
8. Heating, Ventilating, and Air Conditioning
9. Special Equipment

## 1. Building Description

A one (1) story building with a gross floor area of approximately 10,000 square feet is designed to enclose the U-233 Fuel Element Refabrication Facility. This building will contain a Hot Cell, Offices, Health Physics, Process, Process Working Area, Decontamination Area, Toilet Rooms, Hot Change Room, Loading and Storage Pits, Equipment Room, and services to support the functions of this facility and the personnel within.

Since the main problem associated with the handling of U-233 revolves about protection of personnel from alpha contamination, the areas in the building are designed to achieve the following:

1. Protection from ingestion or inhalation of alpha-bearing particles: provided by the complete isolation of the process line and the maintenance of a negative pressure relative to inhabited areas.
2. Localization of any contamination: provided by control of air flow in all areas.

Cracks, crevices, ledges and any particle and dust catching items must be avoided to facilitate decontamination. Materials such as stainless steel, vinyl-asbestos, vinyl tape, as well as special finishes and protective paints are used in accordance with requirements of the particular operating functions.

Spreading of airborne contamination is eliminated by directing air flow to potentially contaminated areas and then to stack.

A radioactive waste system is provided which channels radioactive liquid wastes to underground stainless steel storage tanks for temporary storage pending further disposal.

## 2. Site Development

### a. Excavation

Excavate for slab on grade and for footings, storage and loading pits below grade.

Footings and grade beams shall be located on sound rock or soil of adequate bearing quality.



b. Grading and Drainage

The site shall be graded so that ground surface will slope away from all the sides of the building.

Storm water will be controlled on surface by proper grading.

c. Roads and Walks

Macadam road from shipping and receiving area and connecting to main entry road.

Reinforced concrete sidewalk around perimeter of building.

d. Underground Piping (Exterior to Building)

Sanitary and storm drainage piping shall be extra heavy cast iron soil pipe and fittings with lead calked joints. Piping shall run to existing manhole or existing drainage systems.

Fire and potable water supply shall be 150 lb. cast iron corporation bell spigot piping, with calked joints. Piping shall terminate at connection to plant water supply main. Compressed air piping shall be black steel, seamless piping (ASTM A53) with steel welding fittings and welded joints. Piping shall terminate at connection to plant air piping system.

Steam piping shall be black steel pipe ASTM A53 with welding fittings and welded joints and shall be run to existing plant steam main. Piping shall be insulated with Gilsonite as per the manufacturer's recommendation for the service.

Exhaust ductwork when necessary to run underground shall be fabricated of black steel pipe ASTM A53 with welding fittings, welded joints and coated with corrosion-resistant protective coating.

Demineralized water piping shall be polyvinyl chloride rigid pipe with screwed type, polyvinyl chloride fittings.

Piping shall be run to plant demineralized water system.

e. Fencing

The facility will be located within an existing fenced area. Chain line fencing with 4-strand barb wire and gates shall be provided around the waste and solution tanks at the exterior of the building.

### 3. Architectural and Structural

#### a. Exterior Walls and Foundations

(1) Below Grade. 3000 lbs. reinforced concrete for footings, continuous wall footings and pit walls. Exterior surface of pit walls shall be protected with membrane waterproofing. Interior surfaces of pit walls shall receive a four-coat system of carboline or Amercoat protective coatings suitable for decontamination with hot water.

(2) Above Grade. Non-load-bearing masonry cavity walls or brick with concrete block back-up 10-1/2' high with 6' and filler panels of insulated cement asbestos sandwich panel. Entrance wall shall have native field stone facing on concrete block back-up. Interior surfaces of masonry and filler panel walls shall receive two coats of rubber latex base paint.

#### b. Interior Walls

(1) Below Grade. Partition wall between the Shipping Cask Pit and Finished Fuel Element Storage Pit shall be 3000 lbs. reinforced concrete and shall receive the 4-coat protective coating as specified for exterior walls.

(2) Above Grade. Non-load-bearing concrete or gypsum block with reinforcement for an interior wall except shielding wall at the Cask Loading and Fuel Element Storage pits and Hot Cell walls shall be 12"-thick 3000 lbs. reinforced concrete.

Interior surfaces at office walls shall receive two coats of hard finished white gypsum plaster and two coats of semi-gloss enamel paints. Toilet rooms and Change rooms shall have ceramic glazed tile wainscot 5'-4" high and Keene cement plaster finish above the wainscot to hung ceiling. Plaster surfaces shall receive two coats of rubber latex paint.

Interior exposed surfaces of the Hot Cell and Shielding wall facing the pits shall receive a three-coat system of Carboline or Amercoat protective coating.

All other wall surfaces shall receive two coats of rubber latex paint.

Provide 6" high, reinforced concrete curbing around Decontamination Area.

c. Floor Slabs

(1) Below Grade. Pit floors shall be 3000 lbs. reinforced concrete with 4-coat protective coating system as specified for pit walls and shall have abrasive finish.

(2) Above Grade. All floor slabs on grade shall be 3000 lbs. reinforced concrete with liquid hardener and troweled finish.

Floor in the Hot Cell shall be lined with Type 304 stainless steel sheets 1/8" thick extended up side walls 12" high, all seams and joints shall have continuous welds, the liner shall be anchored to the floor and wall by plug welding the sheets to embedded anchor plates.

Floor in offices, corridor and entrance vestibule shall receive vinyl asbestos tile with 4" high, coved base molding.

Floors in Toilet Rooms and Hot Change Room shall receive quarry tile.

Ramp to Loading and Receiving Area shall be 3000 lbs. reinforced concrete with wood float finish.

All other concrete floors shall receive an acidproof coating system with abrasive finish.

Slabs on grade shall be erected on 8" thick course of crushed stone and provided with a vapor barrier on underside of slab and around the continuous footings.

d. Framing

Structural steel skeleton framing, riveted on welded connection. Open web steel joists on steel girders for supporting metal roof decking.

e. Roofing

Corrugated metal roof decking with 1-1/2" Fiberglass rigid insulation board, vapor barrier and 4-ply 20-year bonded built-up roofing with gravel topping.

Provide aluminum continuous gravel stop and fascia.

f. Ceiling

Acoustical tile hung ceilings on mechanical suspension system in all offices, corridor and vestibule.

Aluminum metal pan hung ceilings on suspension system in toilet rooms and change room.

All other ceilings will have exposed metal roof construction, which shall receive a finish coat of paint to match the side walls except Hot Cell which shall have a poured concrete roof with the underside surface coated same as walls.

g. Doors

All exit doors shall be 16-gage steel, Underwriters label approved, double panel doors with upper panel glazed with wire glass except personnel entrance doors shall have full glass panels, all exit doors shall be provided with panic hardware.

All doors entering into air locks shall be 16-gage steel double paneled doors with neoprene gasketing around periphery of the door for air tight fit, and shall be provided with panic hardware.

All other interior doors shall be 18-gage steel double paneled type with glass in upper panel and provided with locksets.

All doors shall be provided with closers.

Doors and frames shall be finished with same type of painting or protective coatings as the walls of the room.

h. Windows

Windows shall be aluminum fixed sash type, with clear sheet glass glazing from the outside.

i. Plumbing Fixtures and Accessories

The following plumbing fixtures shall be provided in the Change Room, Men's and Ladies' Toilet Rooms as per the following schedule.

	<u>Water Closets</u>	<u>Lav.</u>	<u>Urinals</u>	<u>Showers and Stalls</u>
Hot Change Room	1	1	1	2
Men's Toilet Room	2	2	1	
Ladies' Room	1	1		

Four lavatories, 19 in. x 17 in., vitreous china, Standard F-331-31, with R 2041 chrome plate supply and waste fitting, R 2604 supply pipes with stops and 1-1/2 inch "P" trap with nipple to wall and escutcheon.

Four water closets, vitreous china, Standard F-2223-8 with Sloan No. 110 FYV flush valve, No. 9500 black open front seat, cast iron flange and sponge gasket.

Two urinals, vitreous china 18 in. wall hung with extended shield, integral strainer and trap, Standard F 6200-1 with Sloan No. 186 flush valve.

One service sink for each Janitor's Closet 22 in. x 18 in. enameled iron acid-resisting standard P-2700 with P-7298-P trap standard and R-5001 Combination Hot and Cold faucet with pail hook and brace.

Three safety showers in Process Area, and one at Emergency Exit, Speakman S-2075 with floor drains Zurn Z-510 or equal.

One electric-cooled drinking fountain in corridor with foot pedal supply and chrome plated top (5 gal/hr).

Two shower stalls enameled steel with mixing valve shower and adjustable shower, Fiat Cadet or equal.

Four toilet partitions, flush type, headrail braced, double wall with doors, hardware, coat hooks and toilet paper holders, Fiat "Duro" or equal.

Two urinal partition shields, wall hung double panel type, Fiat or equal.

Four mirrors 18 in. x 24 in.

Four soap dispensers, wall type.

Six steel lockers in Change Room, with sloped top, mounted on concrete base, with one wooden bench.

#### 4. Electrical

##### a. Power Feeders

Three power feeders, one for each motor control center, will be run through separate breakers from existing 500 KVA transformers.

b. Building Voltages

The power will be furnished at 480 v, 3 $\phi$ , 60 cps.

Utilization voltage will be as follows:

1/2 hp motors and over: 440 v, 3 $\phi$

1/3 hp motors and lower: 115 v, 1 $\phi$

Lighting will be 120 v; convenience outlets and low voltage power receptacles will be 208 v and 120 v, 1 $\phi$

c. Building Loading

The maximum power demand is estimated to be 500 kva.

d. Motor Control Centers

Motor control centers, housing breakers and combination starters, will be located in the H and V equipment room. Control components for cell equipment will be located outside the cells.

e. Low Voltage Power

Low voltage power for lighting and receptacles and small motors shall be furnished by 480 - 208/120 v, 3 $\phi$ , 60 cps, 4 wire transformer fed from the 480 volt switchgear.

There will be separate panels for lighting and power receptacles. All panels to be circuit breaker type with 1.2 or 3  $\phi$  breakers as required and will have at least 20% spares.

f. Power Receptacles

An adequate number of receptacles shall be provided throughout the building. All receptacles outside the Hot Cell shall be of the duplex type.

The Hot Cell module will be provided with the following number of receptacles:

Inside Cell

3 - 15 amp, 120 v, 1 $\phi$

1 - 10 amp, 208 v, 1 $\phi$

Cell Face

5 - 15 amp, 120 v, 1 $\phi$

1 - 20 amp, 208 v, 1 $\phi$

g. Cell Lighting

Lighting intensities in the Hot Cell will be 50 footcandles effective at the viewing side of the window. The fixtures shall be located inside the cell (over the window) and shall be designed for remote relamping and replacement. In addition to the window light, push-through 1000 w mercury vapor lights will be located in the hot cell ceiling for general illumination and to illuminate walls and equipment above the cell fixtures. Ballast for mercury vapor lights will be located outside the cell.

h. Building Lighting

Light intensities and type of fixtures shall be as follows:

Hoods	50 fc	Fluorescent
Process Area	50 fc	Fluorescent
Glove Boxes	50 fc	Fluorescent
Air Locks	50 fc	Fluorescent
Offices	60 fc	Fluorescent
Change Rooms, Toilet Rooms	30 fc	Fluorescent
H and V Equipment Room	20 fc	Incandescent
	35 fc	Fluorescent
Receiving Area	20 fc	Incandescent

Lighting level in process areas shall be adjustable from 60 fc to 2 fc for optimum viewing.

i. Grounding

A ground grid connected to driven ground rods and the incoming water main will be provided. All electrical equipment, building steel, cell liners, cranes, monorails, etc., will be connected to this ground grid. Minimum size of ground wires shall be as recommended by the N.E.C.

Motors and other 440 v equipment will be grounded either by a wire run directly to the ground network or by a grounding wire run with the circuit in the same conduit.

j. Conduit and Wire

All conduit and wire will be sized and installed according to N.E.C. with the following exceptions:

Minimum conduit size to be 3/4 inch exposed and 1 inch concealed.

Minimum wire size to be No. 12. Type Tw insulation to be used, up to and including No. 8; RH/RW will be used for No. 6 and larger. Wiring in radioactive areas shall have polyethylene insulation. Wire size No. 10 and smaller may be solid, No. 8 and above will be standard.

All conduits will be standard rigid galvanized, aluminum or EMT. EMT will be used for lighting and receptacle circuits only, and where run above hung ceiling. Only standard rigid galvanized will be run in concrete walls, floors or underground in concrete envelope.

k. Emergency power will be required for the emergency lighting and essential control and instruments. The emergency incoming service shall be run from existing transformers.

1. Communications and Alarm Systems

(1) Fire Alarm System. The fire alarm system shall be master-auxiliary Gamewell type and includes annunciator and Duralarm cabinet. The master box, located at main entrance, is to be Gamewell No. 9000 surface of No. 9103 flush, three-fold, non-interfacing for 48 v. supply. Auxiliary boxes are to be Gamewell No. 9114 surface or No. 9114B flush. The entire installation is to be in accordance with NBFU requirements.

(2) Intercom System. Master and slave stations shall be installed as required for the efficient operation of the facility.

(3) Telephone System. A conduit system (with pull wires installed) for the Bell Telephone system will be installed with outlets in offices, process areas, and at other strategic points. Conduit of sufficient size to allow for normal personnel shift and growth shall be provided.



## 5. Sanitary Plumbing and Storm Drainage

### a. Sanitary Plumbing

Plumbing fixtures shall be furnished and installed as specified under Architectural and Structural Work, in accordance with National Plumbing Code.

Water piping for plumbing fixtures shall be type L copper tubing with bronze solder type fittings. Connect to potable water system.

Sanitary waste and vent piping above ground shall be galvanized steel pipe ASTM A120 with galvanized malleable screwed fittings for vents and galvanized drainage fittings for waste. Waste and vent piping below ground shall be extra heavy cast iron soil pipe and fittings with caulked joints. Waste piping shall be connected to the existing sanitary sewage system.

Floor drains shall be connected to the sanitary sewer.

### b. Storm Drainage

The storm drainage system shall include six roof drains, Zurn Z-100 or equal, with internal leaders of galvanized steel pipe ASTM A120 and galvanized drainage screwed fittings above-ground, cast iron extra heavy soil pipe and fittings with caulked joints below ground.

## 6. Fire Protection

All interior spaces in the building, except cell interiors, shall be served by fire protection facilities as follows:

(1) A wet type sprinkler system with spray heads in compliance with requirements of "ordinary hazard". This shall conform to the latest published standards of the National Fire Protection Association, NFPA No. 13 and No. 14, including a monitor and connections to the existing fire alarm system. Armored protective devices are required for all sprinkler heads.

(2) A system of standpipes and hose cabinets shall be provided to serve all areas. The hose cabinets shall contain one hose 1-1/2 inch size

and 75 feet long and shall be located throughout the building so as to serve floor areas at not to exceed 95 feet from the nearest cabinet. All standpipes and hose cabinets shall conform to the latest standards of the National Board of Fire Underwriters, NBFU pamphlet No. 214.

(3) Auxiliary fire alarm boxes shall be installed immediately adjacent to each hose cabinet along with the required alarm equipment for the wet type system (see Section 4 "Electrical Work").

(4) A Fire Department connection, Siamese type, shall tie into the standpipe and sprinkler systems above ground level where the main enters the building. This location must be accessible to the Fire Department. A valve with post indicator shall be installed in the incoming supply main at least 20 feet from the face of the building. No other pipe connections shall be permitted downstream from this valve except for fire protection purposes. All outside water supply shall conform to Factory Mutual Engineering Division Specifications.

(5) Potable water from the existing plant system shall supply this fire protection system. The potable water main pressure is approximately 80 psig.

## 7. Piping

### a. Codes and Standards

All piping shall conform to the following standards.

- a. American Standard Code for Pressure Piping B-31-1.
- b. National Plumbing Code ASA-A40.8.
- c. National Fire Code, Volumes 1-7, National Fire Protection Association, applicable sections.
- d. American Gas Association, ASA Z-21.30.
- e. American Water Works Association applicable standards for ferrous and non-ferrous pipe, valves and fittings.
- f. American Standards Association applicable standards for ferrous and non-ferrous pipe, valves and fittings.

b. Pipe Lines Designation

All pipe lines shall be identified. In general, each designation will include the following information:

- a. Code for flowing medium
- b. Line number
- c. Line size

c. Color Code for Piping

All piping shall be clearly identified by color as set forth in ASTM standards.

d. Valve Designation

All valves shall be identified by attachment of metal tags upon which will be embossed the applicable service. The metal tags used will be of light gauge Type 304 stainless steel tape approximately 1/2 inch wide and 1-1/2 inches long. The tags will be perforated on each end and bonded securely to the valve by means of Type 304 stainless steel wire. In all cases the tags will be bound in such a position as to present maximum visibility and minimum opportunity for loss. In no case will the tags be attached to handwheels.

e. Piping

(1) Process Piping. All process piping (except radioactive waste) shall be polyvinyl chloride or polyethylene; 1/2 inch and smaller shall be flexible tubing with compression fitting; 3/4" and larger, rigid Schedule 40 with welding fitting and solvent welded joints. All valves shall be of the same material as the piping and suitable for the service.

(2) Radioactive Waste. Radioactive waste piping shall be Schedule 10, Type 304 stainless steel pipe with welding fittings, valves and welded joints.

(3) Air and Steam Piping. Carbon steel Schedule 40 ASTM A-53 with welding fittings and welded joints.

## 8. Heating, Ventilation and Air Conditioning

Outside air intakes shall be provided with fixed aluminum weatherproof louvres and with automatically adjustable dampers interlocked with the supply fan.

Preheat coils shall be of the non-freeze steam distributing tube type and shall be suitable for 15 psig steam. Coils shall be installed vertically with the steam supply connection at the top and with the condensate connections at the bottom.

Steam valves shall be suitable for operations with steam at 15 psig.

Filters shall be America Air Filter Company type PL-24 with 1/16 inch thick Type G Airmat Media. For system design, a final pressure drop of 0.5" W.G. across the filter at the rated air flow before the media is replaced, will be allowed. Filters shall have NBFU Label Class I. High Efficiency Filters shall be Cambridge Filter Corporation absolute Series E. For system design, a final pressure drop of 3.0" W.G. across the filter at the rated air flow before the media is replaced, will be allowed. Filters will be constructed of fire-resistant materials.

Fan shall be selected for efficiency and will be constructed and rated in accordance with AMCA Standards and procedures. Fan drives shall be multiple V-belt and will be sized for 120% of fan motor nameplate horse-power and shall have one adjustable pitch sheave. Fan motors shall be sized for 105% of the fan brake horsepower requirement. Fans for System K-3 shall have radial vanes, motor operated.

Cooling coils shall be selected for chilled water at 42°F entering and 54°F leaving temperatures.

Ductwork construction and testing shall be in accordance with engineering standards; ductwork for process exhaust shall be coated carbon steel, and for glove box and fume hood exhaust shall be polyvinyl chloride, all other ductwork shall be galvanized steel for supply and room exhaust.

Ductwork shall include dampers, test connections and access panels as required. Fire dampers will be installed in supply ducts.

All controls, except for electrical interlocks, shall be pneumatic.

All controllers shall be of the indicating-recording type. Inclined manometer draft gauges shall be provided to indicate the pressure drop across the air filters.

All supply ductwork shall be insulated with fire-resistant materials.

Control valves on Exhaust system K.3 shall be butterfly type motor operated; all other butterfly valves shall be manually operated.

Complete operating instructions including a system description, control settings, operating temperatures, flow rates, lubrication and maintenance schedules, etc., shall be furnished as a guide for operating personnel prior to operating the system.

## 9. Special Equipment

### a. Viewing Windows

The viewing windows for Hot Cell and Shielding Wall shall be non-browning, multi-light oil filled, metal-encased assemblies fitting into a steel liner, and will be embedded in the concrete wall; window shall have protective glass cover plates, oil expansion tank and dryer. Protective plates will be approximately 1 inch thick, non-browning on the hot side and commercial grade on the viewing side. Windows shall be set in 12 inch thick concrete walls.

### b. Monorail Hoists

The monorail hoist in the Shipping and Receiving Area shall be 10-ton capacity with lift of 30 feet, and shall be complete with pendant controls, totally enclosed electric motor, limit switches and takeup reel.

The monorail hoist in the Carrier Unloading Area shall be same as above except 1-ton capacity and 12 foot lift.

### c. Carrier Cask for U-233 Feed Solution Handling and Transfer

The cask shall be fabricated of Type 304 stainless steel plate with sealed cover, lead shielded carrier, stainless steel drain and outlet connection, water connection and polyethylene lead

gloves for internal operation of the water valves. Complete as shown on Drawing 1382-R-2 (Fig. 13).

d. Manipulators

Manipulators for the Hot Cell shall be Central Research Model No. 8 with roller tube mounting assembly.

e. Fume Hoods

Fume hoods shall be of the air flow type with stainless steel interior and steel exterior surfaces with acidproof finish. Cabinet shall be steel with drawers and doors. Hood shall be provided with air, steam, water services and duct connection. Sash shall be glazed with 1/4 inch safety glass. Provide fluorescent lighting fixture in roof of the hood and switches on the exterior.

f. Glove Boxes

The remote operation glove boxes shall consist of compartments with interlocked air locks at each end compartment. Each compartment shall be fitted with Cambridge Filter units and stainless steel enclosures. The compartments shall be fabricated of Type 304 stainless steel plates, 2B finish inside and out and shall have two hinged and gasketed windows with 3/8 inch thick lucite glazing and two pairs of gloves for remote manual operation. All corners and edges shall be coved.

G. COST ESTIMATE1. Summary of Estimate

	<u>Material</u>	<u>Labor</u>
Site Work	\$ 7,200	\$ 7,800
Structural and Architectural	77,160	53,880
Equipment	379,490	37,820
Piping	20,000	10,000
Electrical	46,960	16,570
H.V.A.C.	66,780	18,300
Instruments	<u>12,000</u>	<u>8,000</u>
Direct Totals	(609,590)	(152,370)
G.C. Indirects		
Material Burden M x 5%	30,500	
P.R.T. and Insur. L x 13%		19,800
G.C. Plant O.H. S.T. x 10%		81,200
G.C. Fee S.T. x 7%		62,500
Total Estimated Cost		955,960
Contingency at 15%		143,400
Exclusive of Development Costs		
PRELIMINARY CONSTRUCTION COST ESTIMATE		\$1,099,360
Engineering Cost		
Preliminary Design 4%	44,000	
Detail Design 8%	88,000	
Construction Supervision 2%	22,000	154,000
		\$1,253,360

2. Detailed Cost Estimates

In order that the reader may obtain an idea of the basis on which the cost estimates have been made and that he may appreciate the degree of detailed consideration given to the cost estimate, the following schedules are presented.

VITRO ENGINEERING COMPANY  
A DIVISION OF VITRO CORPORATION OF AMERICA  
**CONSTRUCTION COST ESTIMATE**

Estimate No. 1382

Item SUMMARY OF ESTIMATE

Date \_\_\_\_\_

Signed L.C.W.

		MATERIAL		LABOR	
		UNIT	COST	UNIT	COST
SITE WORK			7 200		7 800
STRUCT. & ARCH.			77 160		53 880
EQUIPMENT			379 490		37 820
PIPING			20 000		10 000
ELECTRICAL			46 960		16 570
H.V.A.C.			66 780		18 300
INSTRUMENTS			12 000		8 000
<b>DIRECT TOTALS</b>			<b>(609 590)</b>		<b>(152 370)</b>
<b>G.C. INDIRECTS</b>					
MATERIAL BURDEN	M x 5%		30 500		
P.R.T. & INSUR.	L x 13%				19 800
G.C. PLANT O.H.	ST x 10%				81 200
G.C. FEE	ST x 7%				62 500
<b>TOTAL EST. COST</b>					<b>955 960</b>
<b>CONTINGENCY @ 15%</b> (EXCLUSIVE OF DEVELOPMENT COSTS)					<b>143 400</b>
<b>PRELIMINARY CONSTR. COST EST.</b>					<b>→ (\$1 099 360)</b>
<b>ENGINEERING COST</b>					
PRELIMINARY DESIGN		4%	44 000		
DETAIL DESIGN		8%	88 000		
CONSTRUCTION SUPERVISION		3%	32 000		
					<b>154 000</b>
					<b>(\$1 253 360)</b>





VITRO ENGINEERING COMPANY  
 A DIVISION OF VITRO CORPORATION OF AMERICA  
**CONSTRUCTION COST ESTIMATE**

Estimate No. 1382

Item STRUCT. & ARCH.

Date \_\_\_\_\_

Signed L.C.W.

					MATERIAL		LABOR	
	UNIT	COST	UNIT	COST	UNIT	COST	UNIT	COST
<u>ELEMENT STORAGE PIT</u>								
<u>SUB-STRUCTURE</u>								
EXCAVATION (MACHINE)	CY	800			-0-	150		1200
BACKFILL (TAMPED)	CY	500			-0-	200		1000
<u>CONCRETE -</u>								
MAT	CY	20	24	-	480	18		360
WALLS	CY	108	33		3560	24		2590
EMBEDDED IRON	TON	1			400			300
WATERPROOFING (EXT.) 2 PLY & INSUL. BD.	SY	200	1	10	220	175		350
PROTECTIVE COATINGS OF PIT INTERIOR	SF	1400		30	420	20		280
PIT COVERS 3/8" CK PL	LBS	3000		.15	450	15		450
SUMP PITS & DRAINS	AC	-			1000			300
					(6530)			(6830)



VITRO ENGINEERING COMPANY  
 A DIVISION OF VITRO CORPORATION OF AMERICA  
**CONSTRUCTION COST ESTIMATE**

Estimate No. 1382

Item STRUCT. & ARCH.

Date \_\_\_\_\_

Signed \_\_\_\_\_

			MATERIAL		LABOR	
			UNIT	COST	UNIT	COST
				(40 140)		(27 050)
<u>ROOFING SYSTEM</u>						
METAL ROOF DECK 18 GA.	}	SF 10M	90	9 000	65	6 500
VAPOR BARRIER						
RIGID INSUL. 2"						
BUR. - R & T 20 YA.						
GRAVEL STOP EXT. AL.	LF	510		1 500		1 000
FLASHING CU 16 OZ.	Lb	1000		700		600
ROOF DRAINS AL W/60' LEADER	EA	8		3 200		1 600
<u>MASONRY ITEMS</u>						
<u>EXTERIOR WALLS (BRICK ON BLOCK)</u>						
8" CB + 4" BRICK	SF	5100	49	2 500	55	2 800
FIELD STONE (VENEER)	SF	160		200		300
PRECAST STONE	LF	460	2	920	50	230
FILLER PANELS	SF	1800	1 40	2 520	1 00	1 800
WINDOW WALLS	SF	500	4	2 000	1 50	750
WALL FLASHINGS	Lb	800	70	560	60	480
				(63 340)		(43 110)

VITRO ENGINEERING COMPANY  
 A DIVISION OF VITRO CORPORATION OF AMERICA  
**CONSTRUCTION COST ESTIMATE**

Estimate No. 1386

Item STRUCT. & ARCH.

Date \_\_\_\_\_

Signed L. C. W.

		MATERIAL		LABOR		
		UNIT	COST	UNIT	COST	
			(63,340)		(43,110)	
<u>MASONRY (CON'T.)</u>						
INTERIOR WALLS -						
6" x 8" x 16" C.B.	SF.	5100	30	1530	34	1740
CERAMIC TILE	S.F.	900	70	560	110	990
<u>MILLWORK -</u>						
PLATE-GLASS ENTRY DOOR SETS -	PR.	2		1200		400
HOLLOW - METAL DOORS 3' x 7' - 21 INCLDS. HARDWARE	SF.	440	550	2420	170	750
TRUCK ENTRY DOOR 12' x 12' - MTR.	SF	144	3 -	430		150
SEAL DOORS & FUEL STORAGE	EA.	4		800		200
WINDOWS - COST SHOWN IN WINDOW WALLS				-0-		-0-
CLOTHES LOCKERS	AL	20		600		40
BINS	}	AL		1500		1000
BENCHES						
FINISH						
				(72,380)		(48,380)

VITRO ENGINEERING COMPANY  
 A DIVISION OF VITRO CORPORATION OF AMERICA  
**CONSTRUCTION COST ESTIMATE**

Estimate No. 1382

Item STRUCT. & ARCH.

Date \_\_\_\_\_

Signed L.C.W.

		MATERIAL		LABOR	
		UNIT	COST	UNPT	COST
			(72 380)		(48 380)
<u>ROOM FINISHES -</u>					
<u>FLOORS -</u>					
STEEL TROWEL W/ LIQUID HARD-	SF 8500	02	170	06	510
VINYL ASBESTOS 6"	SF 1100	32	350	15	160
RUBBER BASE 4"	LF 700	20	140	20	140
QUARRY TILE	SF 1200	90	1 080	70	840
<u>WALLS -</u>					
PLASTER 3 CT. HWE.	SY 620	45	280	1 60	990
PAINTING 3 CT. LATEX	SF 20M	06	1 200	08	1 600
<u>CEILINGS -</u>					
PAINTING 3 CT	SF 8000	06	480	09	720
HUNG CEILINGS ACCOUSTICAL TILE	SF 1800	06	1 080	30	540
			(77 160)		(53 880)



VITRO ENGINEERING COMPANY  
 A DIVISION OF VITRO CORPORATION OF AMERICA  
**CONSTRUCTION COST ESTIMATE**

Estimate No. 1396

Item H.V.A.C.

Date \_\_\_\_\_

Signed L.C.W.

			MATERIAL		LABOR	
			UNIT	COST	UNIT	COST
SYSTEM K-1				4 670		800
DUCTWORK INSULATED	Lb	5000		3 000		1 500
CONTROLS						
REFRIGERATION SYSTEM COMPLETE				24 000		2 000
SYSTEM K-2				6 460		1 000
DUCTWORK & INSUL.	Lb	5000		3 000		1 500
CONTROLS						
SYSTEM K-3				11 850		2 500
DUCTWORK GALV.	Lb	5000		3 000		1 500
DUCTWORK S.S.	Lb	15000	90	13 500	50	7 500
				(66 780)		(18 300)





VITRO ENGINEERING COMPANY  
 A DIVISION OF VITRO CORPORATION OF AMERICA  
**CONSTRUCTION COST ESTIMATE**

Estimate No. 1382

Item EQUIPMENT

Date \_\_\_\_\_

Signed \_\_\_\_\_

TANKS			MATERIAL		LABOR	
	UNIT	COST	UNIT	COST		
DEMIN H <sub>2</sub> O HEAD TANK						
55 GAL. S.S. 30" 4" (F-1)	EA	1		250		
HCL HEAD TANK (F-2)	EA	1		100		
C.S. RUBBER LINED 55 GAL.						
HCL MEASURING TANK						
1 GAL. P.V.C. (F-3)	EA.	1		50		
S-X FEED ADJUSTMENT						
5 GAL. CAP C.S. RL						
W/ AGITATOR (F-4)	EA	1		100		
STRIPPING SOLU - MAKE-UP						
200 GAL. C.S. RL						
W/ AGITATOR (F-5)	EA	1		600	400	
RAFFINATE RECVR 5 GAL.	EA.	1		50		
PVC. C.S. (F-6)						
WASH SOL'N MAKE-UP						
2 GAL PVC (F-7)	EA	1		50		
ELUATE RECVR 5 GAL						
PVC - C.S. (F-8)	EA	1		50		
WASTE RECVR 5 GAL						
PVC/ c.s. (F-9)	EA	1		50		
TH- NITRATE STORAGE						
6000 GAL 304 S.S. (F-10)	EA	1		8 000	3 000	
TH-U MIXING TANK						
175 GAL. 304 S.S. (F-11)	EA	1		1 200	100	
PRECIPITATION TANK 250 GAL.						
304 S.S. JACKETED (F-12)	EA	1		2 500	400	
FILTRATE RECEIVER						
300 GAL 304 S.S. (F-13)	EA.	1		1 500	100	

VITRO ENGINEERING COMPANY  
 A DIVISION OF VITRO CORPORATION OF AMERICA  
**CONSTRUCTION COST ESTIMATE**

Estimate No. \_\_\_\_\_

Item EQUIPMENT

Date \_\_\_\_\_

Signed \_\_\_\_\_

TANKS			MATERIAL		LABOR	
	UNIT	COST	UNIT	COST		
ELUATE RECR 5 GAL 304 S.S. (F-14)	EA	1		50		20
WASTE RECEIVER 300 GAL 304 S.S. (F-15)	EA	1		1 500		100
AMMONIA STORAGE (F-16) 2500 GAL C.S.	EA	1		1 600		500
<b>TOTAL TANKS</b>				<b>(17 650)</b>		<b>(4 670)</b>
TH (OH <sub>4</sub> ) FILTER (G-1) VOL 170 GAL	EA	1		5 000		500
ION-EXCHR COL (D-1)	EA	1		1 000		
(D-2)	EA	1		1 000		400
MIXER SETTLER (L-1)	EA.	1		2 000		500



VITRO ENGINEERING COMPANY  
 A DIVISION OF VITRO CORPORATION OF AMERICA  
**CONSTRUCTION COST ESTIMATE**

Estimate No. 1382

Item EQUIPMENT

Date \_\_\_\_\_

Signed L. C. W.

		MATERIAL		LABOR	
		UNIT	COST	UNIT	COST
L-5 SHIPPING CONTAINER	EA	86	12M	43 200	-0-
L-6 AMMONIA DISSOCIATOR	EA	1		7 500	2 000
GLOVE BOXES				59 400	6 000
FILTERS				7 840	
MONO-RAIL & HOISTS 10 TON				4 500	500
MONO-RAIL & HOISTS 1 TON				1 500	500
				(379 490)	(37 820)



VITRO ENGINEERING COMPANY  
 A DIVISION OF VITRO CORPORATION OF AMERICA  
**CONSTRUCTION COST ESTIMATE**

Item JOB # 1382  
NUCLEAR

Estimate No. \_\_\_\_\_

Date 5-31-61

Signed \_\_\_\_\_

		MATERIAL		LABOR	
		UNIT	COST	UNIT	COST
POWER TRANSFORMER 500 KVA 4160/480V - 3P - 60 ~	1	10,000	10,000	700	700
480 V SWITCHGEAR					
CIRCUIT BREAKER 100A - 480V - 3P	2				
CIRCUIT BREAKER 150A - 480V - 3P	3		6000		500
CIRCUIT BREAKER 225A - 480V - 3P	2				
MCC. #1					
SIZE 1 COMB. MAG. STARTER W/ CIRCUIT BREAKER	10	230	2300	25	250
CIRCUIT BREAKER 100A - 480V - 3P	4	200	800	25	100
MCC. #2					
SIZE 1 COMB. MAG. STARTER W/ CIRCUIT BREAKER	10	230	2300	25	250
CIRCUIT BREAKER 100A - 480V - 3P	3	200	600	25	75
A.C. M.C.C.					
SIZE 1 COMB. MAG. STARTER W/ CIRCUIT BREAKER	4	230	920	25	100
SIZE 2 COMB. MAG. STARTER W/ CIRCUIT BREAKER	4	300	1200	35	140
SIZE 3 COMB. MAG. STARTER W/ CIRCUIT BREAKER	3	450	1350	60	180
SIZE 4 COMB. MAG. STARTER W/ CIRCUIT BREAKER	2	750	1500	80	160

VITRO ENGINEERING COMPANY  
 A DIVISION OF VITRO CORPORATION OF AMERICA  
**CONSTRUCTION COST ESTIMATE**

Estimate No. \_\_\_\_\_

Item JOB # 1382

Date 5-31-61

Signed \_\_\_\_\_

		MATERIAL		LABOR	
		UNIT	COST	UNIT	COST
TRANSFORMER 75 KVA-3 $\phi$ -60~ 480/120/208 V.	1	900	900		200
TRANSFORMER 3-1 $\phi$ 15 KVA-60~ 480/120/208 V.	1	600	600		180
CIRCUIT BREAKER-100A-3P-220V	1	100	100	30	30
DISTRIBUTION PANEL-3 $\phi$ -4W 120/208V W/ 6-100A-3P BRKRS	1	300	300	250	250
AUTOMATIC TRANSFER SWITCH 50A-220V-3P	1	300	300	.50	50
LIGHTING PANEL 3 $\phi$ -4W 120/208V 24 CKT	4	230	920	100	400
LIGHTING PANEL 3 $\phi$ -4W 120/208V 12 CKT	1	180	180	.70	70
1/2 500 MCM CABLE	300'	1.90	570	.40	120
1/2 # 2/0 CABLE	1200'	.55	660	.23	300
1/2 # 4/0 CABLE	600'	1.20	720	.45	270
1/2 # 2 CABLE	1500'	.30	450	.10	150
1/2 # 6 WIRE	2400'	.14	340	.08	200
1/2 # 12 WIRE	27,000'	.25	675	.04	1080
1/2" THRU 1/2" RIGID STEEL CONDUIT	10,000'	.30	3000	.50	5000
2" RIGID STEEL CONDUIT	500'	1.00	500	1.40	650
2 1/2" RIGID STEEL CONDUIT	200'	1.40	300	1.80	280
3" RIGID STEEL CONDUIT	200'	1.90	400	2.40	300
5" RIGID STEEL CONDUIT	50'	4.00	200	3.00	150
			38 085		12 135
					38 085
TOTAL LABOR & MATERIAL					(51 220)



H. Bibliography

Not all of the listed references that follow were consulted for design. Certain of the more pertinent reports were used. The list is included for general information and future use.

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WAPD-PWR-PMM-466

MCW-404

TID-7530 Part 2

TID-7546 Book 2

WAPD-MRP-68

BMI-1117

CRL-45

KAPL-1918

CRFD-762

KAPL-1702

BMI-1205

BMI-1294

ARCL-836

SCNC-290

NYO-2684

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\* These references are cited in full in the General Bibliography at the end of this report (p. 157, ff).

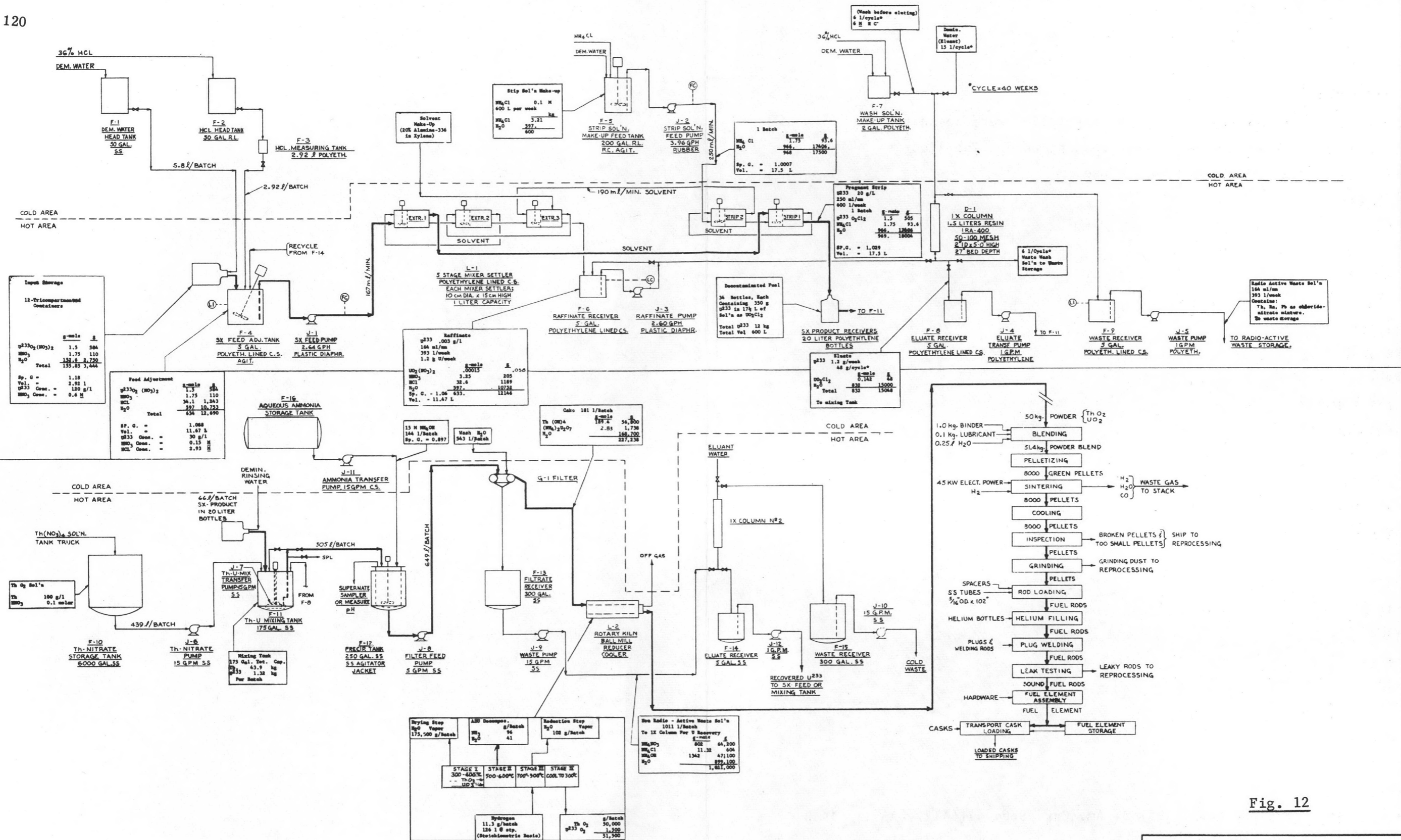


Fig. 12

**NUCLEAR METALS INC.**  
CONCORD, MASS.

SUBCONTRACT NR. 61-1      UNDER CONTRACT NR. AF(30-1)-1963

**VFC ENGINEERING COMPANY**  
A DIVISION OF VFC CORPORATION OF AMERICA  
100 CHURCH STREET - NEW YORK 7, N. Y.

**PROPOSED U-233 FUEL ELEMENT  
REFABRICATION FACILITY  
ENGINEERING FLOW DIAGRAM  
& MATERIAL BALANCE**

DRAWN BY: W.M. JONES	DATE: 5-12-61	SCALE: NONE
CHECKED: 5/17	DRAWING NO. 1382-M-1	
APPROVED: 5/17	APPROVED:	

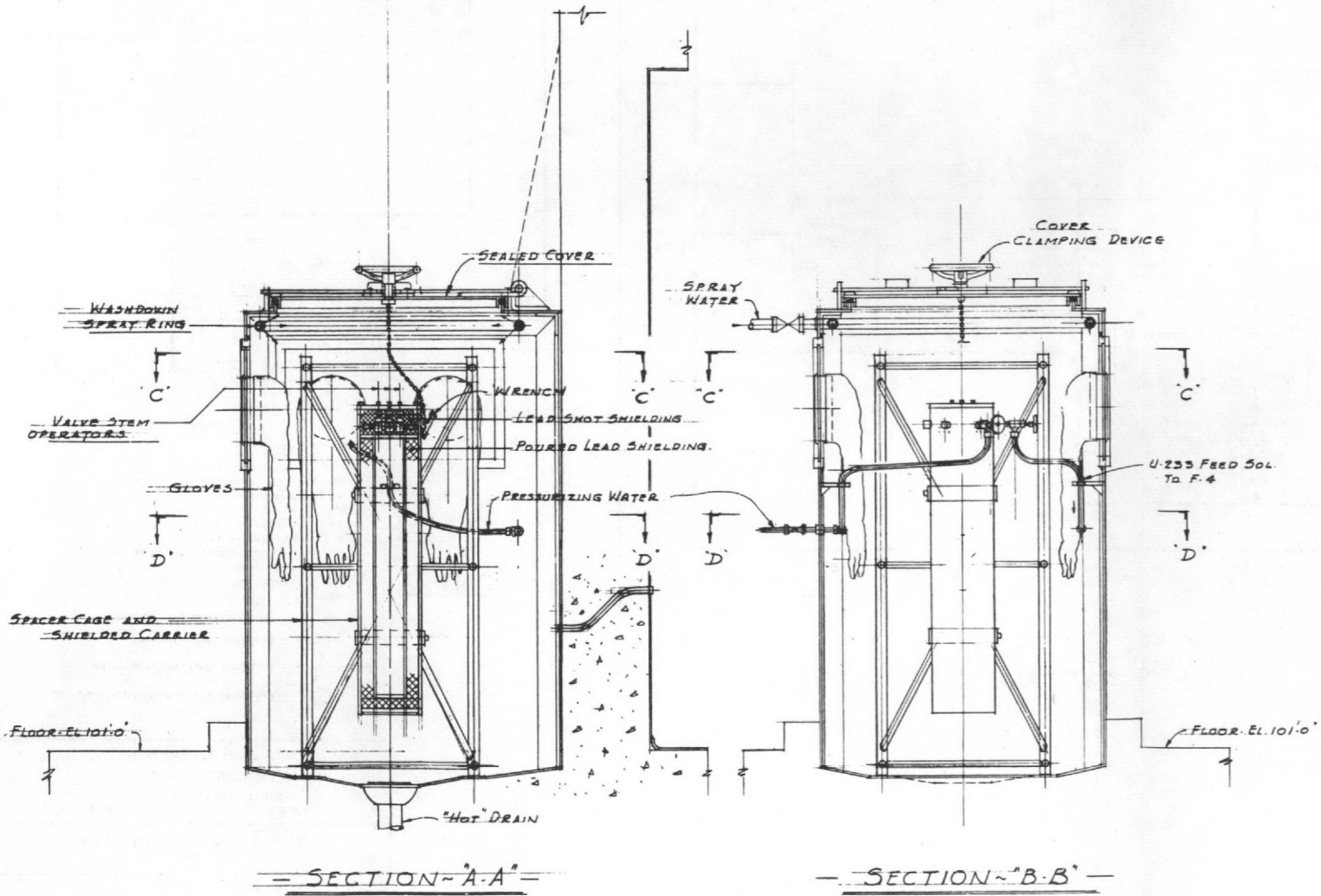
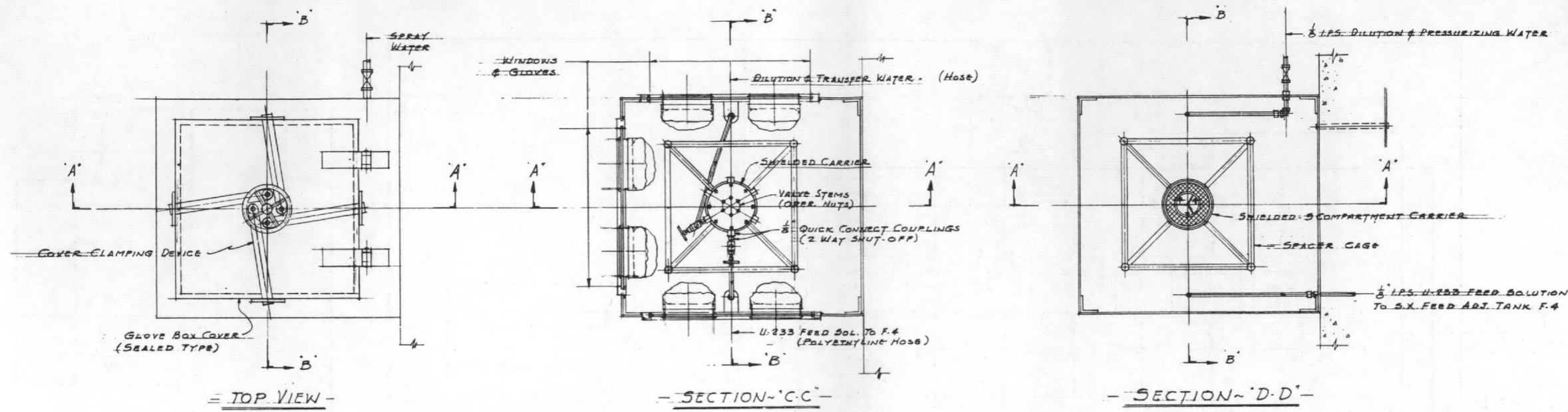


Fig. 13

NUCLEAR METALS, INC.  
 CONCORD, MASS.  
 SUB CONTRACT # G1-1  
 UNDER CONTRACT NO. AF (30-1) 1565

ENGINEERING COMPANY  
 A DIVISION OF THE COMPANY OF AMERICA  
 100 CHURCH STREET - NEW YORK 7, N. Y.

PROPOSED U-233 FUEL ELEMENT  
 REFABRICATION FACILITY  
 PROPOSED METHOD FOR U-233 FEED  
 SOLUTION HANDLING & TRANSFER

NO.	DATE	REVISIONS	APPROVED	DATE 5-12-61	SCALE 1/8" = 1'-0"	DRAWING NO.
						1382-R2

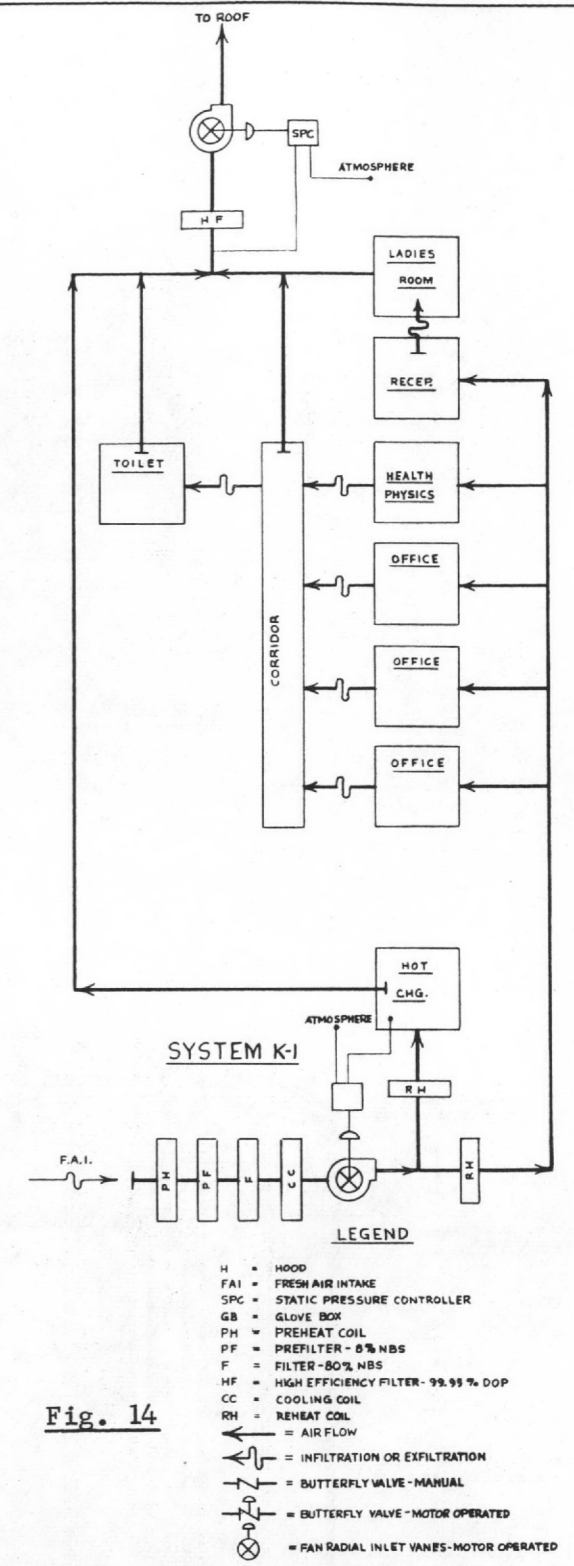
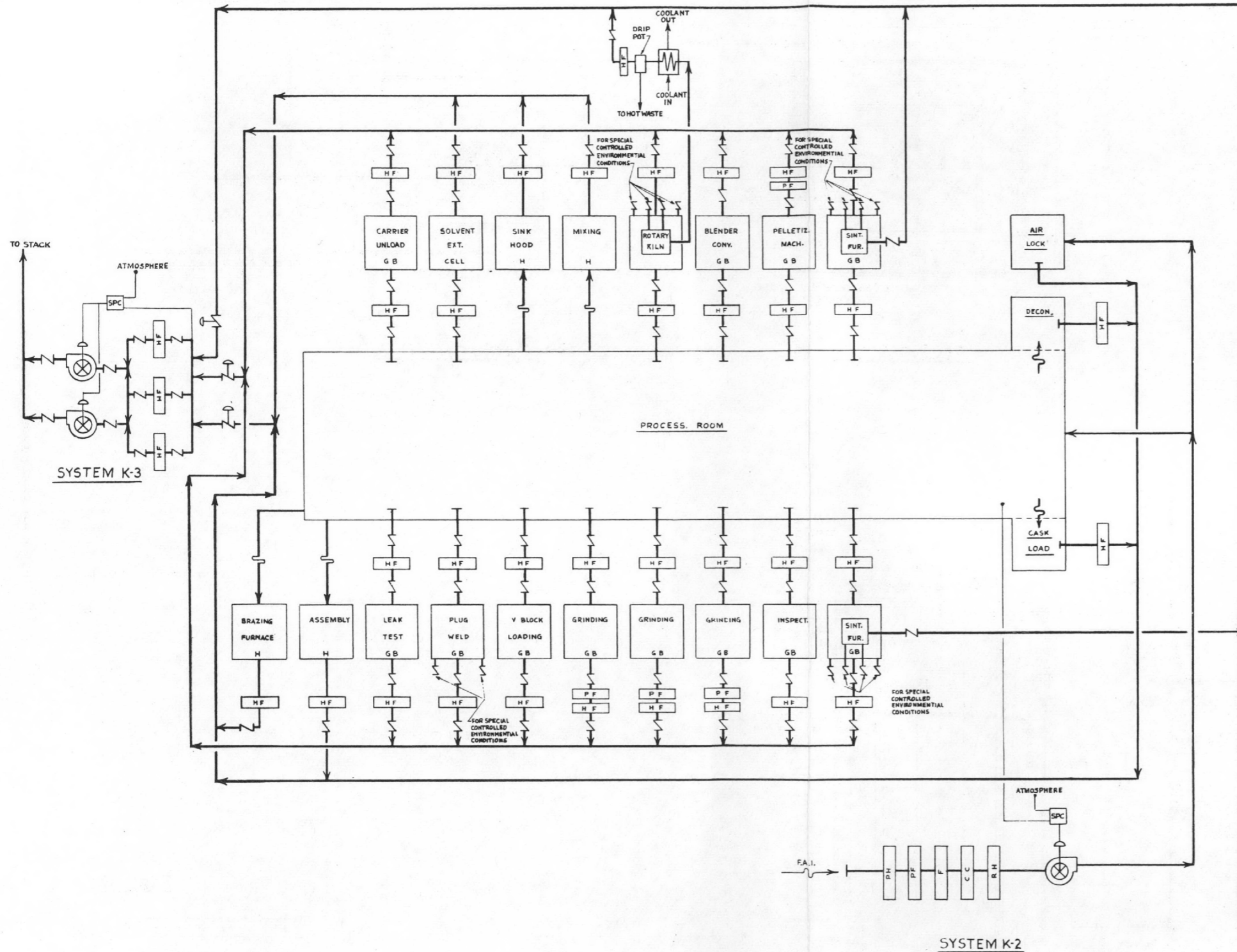


Fig. 14

- LEGEND**
- H = HOOD
  - FAI = FRESH AIR INTAKE
  - SPC = STATIC PRESSURE CONTROLLER
  - GB = GLOVE BOX
  - PH = PREHEAT COIL
  - PF = PREFILTER - 8% NBS
  - F = FILTER - 80% NBS
  - HF = HIGH EFFICIENCY FILTER - 99.99% DOP
  - CC = COOLING COIL
  - RH = REHEAT COIL
  - = AIR FLOW
  - ↔ = INFILTRATION OR EXFILTRATION
  - ⌵ = BUTTERFLY VALVE - MANUAL
  - ⌵ = BUTTERFLY VALVE - MOTOR OPERATED
  - ⊗ = FAN RADIAL INLET VANES - MOTOR OPERATED

NUCLEAR METALS INC.  
 CONCORD, MASS.  
 SUBCONTRACT # G1-1  
 UNDER CONTRACT NO AF (30-1)-1565

**ENR ENGINEERING COMPANY**  
A DIVISION OF VITKO CORPORATION OF AMERICA  
 100 CHURCH STREET - NEW YORK 7, N. Y.

PROPOSED U-233 FUEL ELEMENT  
 REFABRICATION FACILITY

**H<sub>2</sub>V AIR FLOW  
 DIAGRAM**

DRAWN BY	D. R. BERLUCCI	DATE	5-11-61	SCALE	NONE
CHECKED					
APPROVED					
NO.	BY	DATE	REVISIONS	APPROVED	

DRAWING NO. 1382-K-1

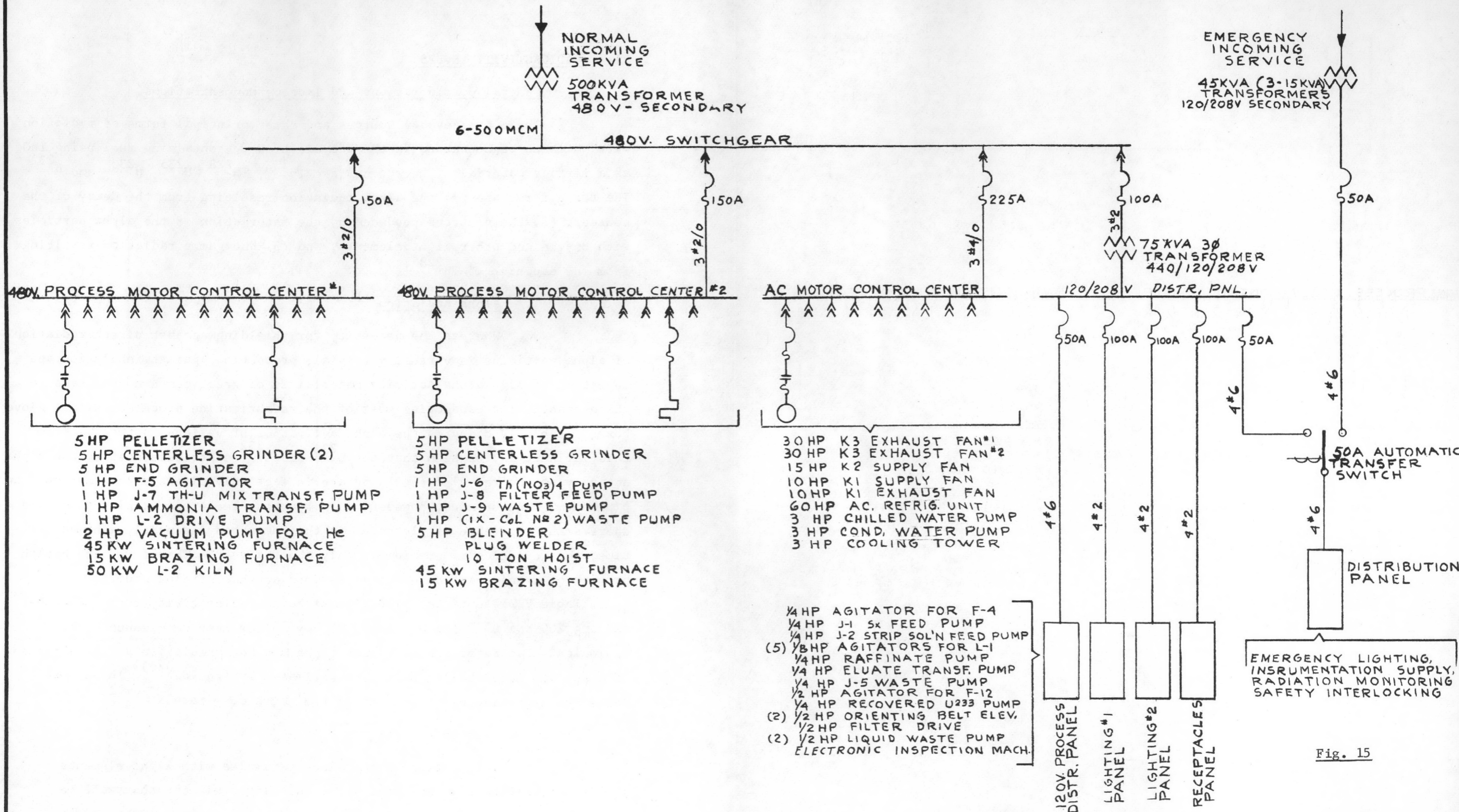


Fig. 15

				<b>Vitro ENGINEERING COMPANY</b> A DIVISION OF VITRO CORPORATION OF AMERICA 100 CHURCH STREET • NEW YORK 7, N. Y.				NUCLEAR METALS INC	
				SCALE	NONE	APPD.	JR	ELECTRICAL ONE LINE DIAGRAM	
NO	DATE	DESCRIPTION	BY CH AP	DRAWN	A.C.	APPD.	JWU 5/13		
REVISIONS				CHKD.		APPD.	JWU 5/16	DWG. NO. 1382-N-1	

## VI. RADIOACTIVITY LEVELS

### A. General Considerations and Isotope Concentrations

There are several sources and three principal forms of radiation with which to be concerned in the recycle of  $U^{233}$ -containing fuel using  $ThO_2$  as a fertile material. The sources are  $Th^{232}$ ,  $Th^{228}$ ,  $U^{235}$ ,  $U^{233}$ , and  $U^{232}$ . The three forms are (1) the alpha radiation resulting from the decay of the sources, (2) the neutrons produced by the interaction of the alpha particles with oxygen and other light elements, and (3) the gamma radiation resulting from the daughters of  $Th^{228}$ .

#### 1. Alpha Radiation

Whereas the necessity for shielding against direct radiation of alpha particles is virtually trivial, protection against inhalation and ingestion of high alpha-activity material is of great concern and leads to the necessity for conducting most of the refabrication processes within glove boxes. Table IV compares the alpha activities of various isotopes. From this Table it may be seen that the  $U^{233}$  and  $U^{232}$  contribute most to the alpha activity of the recycled fuel, and are in fact only slightly less active than plutonium. The uranium, however, is not a bone-seeker as is plutonium, and whereas the residence of plutonium in the human body is nearly a human lifetime, the residence of uranium is only about a year. Therefore, the health problem of uranium is not quite so serious as that of plutonium.

Table V presents the allowable airborne radioactivity from fuel containing 200 ppm  $U^{232}$  (in uranium), 90 days after uranium cleanup. The biological dose rates due to direct alpha and beta radiation at the surface of a thorium metal billet have been estimated by Arnold.<sup>(10)</sup> These dose rates are approximately 4 percent of the gamma dose rate.

#### 2. Neutron Radiation

Interaction of the alpha particles with light elements yields penetrating fast neutrons. The only light element that will be associated in high concentration with the fuel is oxygen. Other light elements are present in low concentrations and are neglected here. The threshold of the  $(\alpha, n)$  reaction with  $O^{16}$  is 15 Mev and hence the production

of neutrons is not energetically possible. The Q for the  $(\alpha, n)$  reaction in  $O^{17}$  and  $O^{18}$  is approximately zero, however, and neutrons will be produced.

One can estimate an upper limit of the neutron production from the  $O^{17}$   $(\alpha, n)$  and  $O^{18}$   $(\alpha, n)$  reactions by considering the production of neutrons from radium-beryllium neutron sources. The Q for the  $Be^9(\alpha, n)C^{12}$  reaction is +6.7 Mev and hence is much more probable than either the  $O^{17}$  or  $O^{18}$   $(\alpha, n)$  reactions. The yield of neutrons from the  $Be^9(\alpha, n)C^{12}$  reaction is  $1.5 \times 10^7$  n/curie or 400 neutrons per  $10^6$  alphas. Since the abundance of  $O^{17}$  and  $O^{18}$  is only 0.24%, the upper limit of neutron production from oxygen would be 0.96 neutrons per  $10^6$  alpha particles. A value of 0.1 neutron per  $10^6$  alphas would be more probable. We understand that the Oak Ridge National Laboratory has some plans to measure the yield of neutrons from  $U^{232}$  (and its daughters) in oxygen-containing media.

At the upper limit of 1 neutron per  $10^6$  alphas the neutron dose is a factor of  $10^3$  lower than the gamma dose. In some instances where thick gamma shielding is required, the neutron dose may be significant. However, if concrete were used for gamma shielding, the neutron dose would be correspondingly reduced.

### 3. Gamma Radiation

The gamma-ray activity from natural thorium is significantly greater than the activity from  $U^{232}$  during the first few days of fuel reprocessing, and is therefore worthy of consideration.

The activity of natural thorium may vary by a factor greater than 2, depending upon the extent of radium removal during refining and the time elapsed since processing. Section a, following, considers these aspects of thorium activity.

Another variable contributing to the thorium activity is the  $Th^{230}$  content. There are substantial deposits of thorium which occur in uranium ores, and thorium is economically recoverable as a by-product of uranium recovery. This thorium will probably contain up to 100 ppm of  $Th^{230}$  as a result of the  $U^{234}$  decay. The  $Th^{230}$  chain, decaying through  $Ra^{226}$ , also emits some energetic photons, and could cause serious trouble from background

activity. However, since there are substantial quantities of Th<sup>230</sup>-free thorium available, we do not feel that the Th<sup>230</sup> problem warrants quantitative consideration here.

Although the activity of natural thorium is of importance, it is of only secondary concern. Of primary concern for U<sup>233</sup> fuel refabrication is the buildup of activity due to the presence of U<sup>232</sup> in the fuel. Here, it can be assumed that Th<sup>228</sup> and its daughters are almost completely removed from the fuel immediately prior to processing. The buildup of Th<sup>228</sup> and its daughters from U<sup>232</sup> during processing is of prime importance and is considered in Section b, following.

a. Th<sup>228</sup> Concentration as a Function of Ra Removal During Refining, and Time Elapsed Since Refining

Consider one mol of natural thorium in radiochemical equilibrium with its daughters. We are concerned principally with Th<sup>232</sup>, Ra<sup>228</sup>, and Th<sup>228</sup>; their significant properties and natural abundance relative to one another are described in the table below:

<u>Isotope</u>	<u>Half-Life</u>	<u>Abundance/Mol Th<sup>232</sup></u>
Th <sup>232</sup>	1.4 x 10 <sup>10</sup> yrs.	6 x 10 <sup>23</sup> atoms
Ra <sup>228</sup>	6.7 yrs.	2.9 x 10 <sup>14</sup> atoms
Th <sup>228</sup>	1.9 yrs.	8.1 x 10 <sup>13</sup> atoms

During the processing of Th, a great deal of the Ra is removed, so that for a few years after refining at least, the activity of natural thorium actually declines. The Th<sup>228</sup> decay can be described by the equation:

$$\frac{dn_{08}}{dt} = (\lambda n)_{88} - (\lambda n)_{08} \quad (19)$$

where

- λ is the disintegration rate constant (0.693 divided by the half life in years),
- n is the number of atoms,
- 08 refers to Th<sup>228</sup>
- 88 refers to Ra<sup>228</sup>, and
- t is the time in years.



Ra<sup>228</sup> buildup can be expressed by a similar equation:

$$\frac{dn_{88}}{dt} = (\lambda n)_{02} - (\lambda n)_{88} \quad (20)$$

where 02 refers to Th<sup>232</sup>.

Substituting the values:

$$\begin{aligned} (\lambda n)_{02} &= 3 \times 10^{13} \text{ disintegrations/yr. and} \\ \lambda_{88} &= 0.104 \text{ years}^{-1} \end{aligned}$$

$$\frac{dn_{88}}{3 \times 10^{13} - .104 n_{88}} = dt$$

Solving and integrating between limits,

$$\ln \left| \frac{3 \times 10^{13} - .104 n_{88}}{3 \times 10^{13} - .104 f (2.9 \times 10^{14})} \right| = -.104 t \quad (21)$$

$$n_{88} = 2.9 \times 10^{14} \left[ 1 - (1-f)e^{-.104t} \right] \quad (22)$$

where

f is the fraction of Ra<sup>228</sup> remaining after refining.

Substituting the  $\lambda$  values and Eq. (22) in Eq. (19),

$$\frac{dn_{08}}{dt} = 3 \times 10^{13} \left[ 1 - (1-f)e^{-.104t} \right] - 0.37 n_{08} \quad (23)$$

Transposing, and solving the resulting linear equation,

$$n_{08} e^{.37t} = 3 \times 10^{13} \left[ 2.7 e^{.37t} - (1-f)(3.8) e^{.26t} \right] + C$$

When  $t = 0$ ,  $n_{08} = 8.1 \times 10^{13}$ , so solving for the integration constant,  $C = 1.1 \times 10^{14} (1-f)$  and substituting back into the equation,

$$n_{08} = 3 \times 10^{13} \left[ 2.7 - 3.8(1-f)(e^{-.104t} - e^{-.37t}) \right] \quad (24)$$

Substituting values for t and f, Th<sup>228</sup> concentrations are listed as a function of time lapsed after refining and radium removal in Table VI and the results are plotted in Figure 16.

b. Buildup of Th<sup>228</sup> and Daughters From Decay of U<sup>232</sup>

Figure 17 presents the decay schemes for Th<sup>232</sup> and U<sup>232</sup>. Consider one mol of thorium containing one ppm of U<sup>232</sup>, initially clean. The buildup of Th<sup>228</sup> can be expressed by the equation

$$\left(\frac{dn}{dt}\right)_{08} = (\lambda'n)_{22} - (\lambda'n)_{08} \quad (25)$$

For a time period of days, the decay of Th<sup>228</sup> is of insignificant effect upon its concentration, so decay may be ignored. Therefore,

$$n_{08} = (\lambda'n)_{22} t \quad (26)$$

where  $\lambda'$  is the decay constant (0.69 divided by the half-life in days,)

$t$  is the time in days

$n$  is the number of atoms per mol of Th,

08 refers to Th<sup>228</sup>, and

22 refers to U<sup>232</sup>.

Since

$$\begin{aligned} \lambda'_{22} &= 2.6 \times 10^{-5} \text{ days}^{-1}, \text{ and} \\ n_{22} &= 6 \times 10^{17} \text{ a/ppm,} \\ n_{08} &= 1.6 \times 10^{13} t \end{aligned} \quad (27)$$

for Ra<sup>224</sup>,

$$\left(\frac{dn}{dt}\right)_{84} = (\lambda'n)_{08} - (\lambda'n)_{84} \quad (28)$$

$$\lambda'_{08} = 10^{-3} \text{ days}^{-1}$$

$$\lambda'_{84} = 0.19 \text{ days}^{-1}$$

so,

$$\begin{aligned} \left(\frac{dn}{dt}\right)_{84} &= 1.6 \times 10^{10} t - .19 n_{84} \\ \left(\frac{dn}{dt}\right)_{84} + .19 n_{84} &= 1.6 \times 10^{10} t \end{aligned} \quad (29)$$

Solving the linear equation,

$$n_{84} e^{.19t} = 1.6 \times 10^{10} \int t e^{.19t} dt$$

$$n_{84} e^{.19t} = 1.6 \times 10^{10} \left( \frac{e^{.19t}}{.036} \right) (.19t - 1) + c$$

when  $t = 0$ ,  $n_{84} = 0$ , so

$$c = 4.4 \times 10^{11}$$

$$n_{84} = 4.4 \times 10^{11} [(.19t - 1) + e^{-.19t}] \quad (30)$$

The half-lives of  $\text{Em}^{220}$  and  $\text{Po}^{216}$  are so short, that they are essentially in equilibrium with  $\text{Ra}^{224}$ . Both are essentially pure  $\alpha$  emitters.

However,  $\text{Pb}^{212}$  has a significant half-life ( $\lambda' = 1.57$ ), and it is a source of trouble, so its concentration warrants calculation.

$$\left( \frac{dn}{dt} \right)_{22'} = (\lambda' n)_{84} - (\lambda' n)_{22'} \quad (31)$$

where 22' refers to  $\text{Pb}^{212}$

Substituting for  $\lambda'$  and  $n_{84}$  from eqn (30)

$$\left( \frac{dn}{dt} \right)_{22'} + 1.57 n_{22'} = .19 (4.4 \times 10^{11}) [(.19t - 1) + e^{-.19t}] \quad (32)$$

Solving the linear equation,

$$n_{22'} e^{1.57t} = 8.4 \times 10^{10} \int [ .19t e^{1.57t} - e^{1.57t} + e^{1.38t} ] dt$$

$$n_{22'} e^{1.57t} = 8.4 \times 10^{10} [ .077 e^{1.57t} (1.57t - 1) - .64 e^{1.57t} + .72 e^{1.38t} ] + c \quad (33)$$

when  $t = 0$ ,  $n_{22'} = 0$ , so  $c \cong 0$

Therefore,

$$n_{22'} = 8.4 \times 10^{10} [ .077 (1.57t - 1) - .64 + .72 e^{-.19t} ]$$

$\text{Bi}^{212}$  and  $\text{Tl}^{208}$ , the principle sources of  $\gamma$  radiation, are in equilibrium with  $\text{Pb}^{212}$  since their half-lives are short. The concentrations and half-lives are listed for the various isotopes as a function of time in Table VII. and are plotted in Figure 18.

After a period of 90 days, the isotopic concentration is almost directly proportional to the half-lives. However, the decay of  $\text{Th}^{228}$  has become important, so that Th concentration can more properly be represented by the solution to equation (25):

$$n_{08} = 1.6 \times 10^{16} \left[ 1 - e^{-10^{-3}t} \right] \quad (34)$$

The activity shown in Figure 18 for time after 90 days is based on the  $\text{Th}^{228}$  concentration as calculated from equation (34).

#### B. Gamma Radiation Dose Rates From Fuel Containing $\text{U}^{232}$ And Its Daughter Products

The buildup of radioactivity from  $\text{Th}^{228}$  and its daughters in fuels containing  $\text{U}^{232}$  has been discussed in Section A3a. The radiation dose rates due to gamma rays resulting from the radioactive disintegration of these isotopes are investigated here for several configurations which may be encountered in a fabrication plant as a function of time after cleanup of the uranium. These are:

1.  $\text{UO}_2 (\text{NO}_3)_2$  in a 5 gm U/l solution for a 3 inch column.
2. Activity of a 3 inch diameter ion exchange column used to clean up the  $\text{UO}_2 (\text{NO}_3)_2$  solution.
3.  $\text{UO}_2$  in bulk form.
4.  $\text{UO}_2$  in  $\text{ThO}_2$  of natural activity in bulk form.
5.  $\text{UO}_2$  in  $\text{ThO}_2$  of natural activity in the form of an infinite length 1/2-inch rod.

The decay schemes, yield and energies of the various isotopes under consideration are given in Chapter 1.2 of the Reactor Handbook, Vol. I.<sup>(5)</sup> The data on gamma ray production for the  $\text{Th}^{228}$  chain have been reduced in Table VIII in the form of Mev of energy produced per disintegration in each of 7 different gamma energy groups. The energies and yield of alpha particles are shown in Table IX and the yield and energy of beta particles in Table X.

The disintegration rates of  $\text{Th}^{228}$  and its daughters are given in Table VII for 1 mole of thorium containing 1 ppm  $\text{U}^{232}$ . For convenience these data are converted to disintegration rates due to 1 ppm of  $\text{U}^{232}$  in 1 gram of

uranium. The gamma radiation produced by these disintegrations was then determined using Table VIII and is given in Table XI.

The dose rates due to the gamma radiation were calculated using standard methods. Cases (1), (2) and (5) are cylindrical geometry. Spherical geometry was assumed for cases (3) and (4) which involve bulk materials.

The gamma dose rate due to an unshielded cylindrical volume source is

$$D = \sum_i \frac{S_i G_i}{4 R F_i}$$

where,

$S_i$  = Line source strength in gamma group  $i$ , Mev/sec-cm.

$R$  = Distance from center of cylinder.

$F_i$  = Conversion constant Mev/cm<sup>2</sup> sec to Mr/hr for gamma in group  $i$ .

$G_i$  = Self-shielding factor for group  $i$ .

For spherical geometry:

$$D = \sum_i \frac{S_i G_i}{4 R^2 F_i}$$

$S_i$  = Total source strength of energy group  $i$ , Mev/sec.

The self-shielding factor is determined using the results of Storm, Hurwitz and Roe.<sup>(11)</sup>

Results of the calculations are as follows:

Case 1: The dose rate one meter from a 3-inch diameter column containing 5 g U/l as uranyl nitrate solution with 1 ppm U<sup>232</sup> in the uranium is given in Table A for various times after uranium cleanup. The dose rate varies inversely as the distance from the column.

TABLE A

Dose Rate 1 Meter from a 3-inch Column  
Containing 5 g/l Uranium with 1 ppm U<sup>232</sup>

<u>Time After Cleanup (Days)</u>	<u>Dose Rate (mr/hr.)</u>
1	$9.1 \times 10^{-5}$
3	$4.3 \times 10^{-3}$
7	$2.35 \times 10^{-2}$
15	$8.5 \times 10^{-2}$
30	0.216
90	0.745
400	2.53

Case 2: The dose rate at 1 meter due to the activity retained on a 3-inch diameter ion exchange column 3 feet long when saturated by uranium containing 1 ppm U<sup>232</sup> and its daughter products is shown in Table B for various times after cleanup of the uranium prior to passing it thru the ion exchanger. The result is based on the assumption that the resin will retain 4% of the uranium treated before becoming saturated or 250 g U/(1 of resin) plus the Th<sup>228</sup> and its daughter products associated with the uranium processed.

TABLE B

Dose Rate One Meter from a 3-inch Ion Exchange Column  
Saturated with Uranium Containing One ppm U<sup>232</sup> as a  
Function of Time After Previous Cleanup of the Uranium

<u>Time After Cleanup (Days)</u>	<u>Dose Rate (mr/hr)</u>
1	0.11
3	5.4
7	29.
15	106.
30	270
90	930
400	3200

Case 3: The dose rate due to 1 kg of uranium as  $\text{UO}_2$  containing 1 ppm of  $\text{U}^{232}$  is given in Table C for various times after cleanup of the uranium. The dose rate due to neutrons produced by the  $(\alpha, n)$  reaction with the oxygen isotopes was calculated for the 90-day case. The dose rate due to neutrons at 1 meter was  $3.9 \times 10^{-5}$  mr/hr per ppm of  $\text{U}^{232}$  compared to 0.12 mr/hr per ppm of  $\text{U}^{232}$  due to gamma radiation.

TABLE C  
Dose Rate in mr/hr from 1 kg of Uranium  
as  $\text{UO}_2$  Containing 1 ppm  $\text{U}^{232}$

Distance	Time After Uranium Cleanup (Days)						
	1	3	7	15	30	90	400
Surface	$1.2 \times 10^{-2}$	5.3	31	116	302	1030	3500
10 cm	$6.6 \times 10^{-4}$	0.28	1.7	6.2	16	54	185
100 cm	$6.6 \times 10^{-6}$	$2.8 \times 10^{-3}$	$1.7 \times 10^{-2}$	$6.2 \times 10^{-2}$	0.16	0.54	1.85

Case 4: The dose rate due to 10 kg of  $\text{ThO}_2 + 5\% \text{UO}_2$  containing 1 ppm  $\text{U}^{232}$  in the uranium is given in Table D as a function of time after cleanup of the uranium. The thorium is assumed to contain no daughter products in this Table. The dose rate due to the thorium with natural radioactivity is 0.20 mr/hr at 1 m, 20 mr/hr at 10 cm and 78 mr/hr at the surface and should be added to the result obtained by multiplying the dose rates by the  $\text{U}^{232}$  concentration.

TABLE D  
Dose Rate in mr/hr from 10 kg of  
 $\text{ThO}_2 + 5\% \text{UO}_2$  Containing 1 ppm  $\text{U}^{232}$

Distance	Time After Uranium Cleanup (Days)						
	1	3	7	15	30	90	400
Surface	$3.1 \times 10^{-4}$	0.245	1.5	5.5	14.0	47	163
10 cm	$7.7 \times 10^{-5}$	$6.2 \times 10^{-2}$	0.37	1.4	3.5	12	41
100 cm	$7.7 \times 10^{-7}$	$6.2 \times 10^{-4}$	$3.7 \times 10^{-3}$	$1.4 \times 10^{-2}$	$3.5 \times 10^{-2}$	0.120	0.41

Note: After multiplying by  $\text{U}^{232}$  concentration, add 78 mr/hr at surface, 20 mr/hr at 10 cm or 0.20 mr/hr at 100 cm for natural thorium activity.

Case 5: The dose rate due to an infinite length of  $\text{ThO}_2 + 5\% \text{UO}_2$  fuel rod containing 1 ppm  $\text{U}^{232}$  in the uranium is given in Table E as a function of time after cleanup of the uranium. Here again, the dose rate from natural thorium is neglected and should be added after multiplying by the  $\text{U}^{232}$  concentration. The dose rate varies inversely as distance from the rod.

TABLE E

Dose Rate in mr/hr from a 1/2-inch Rod of  $\text{ThO}_2$   
+ 5%  $\text{UO}_2$  Containing 1 ppm  $\text{U}^{232}$  in the Uranium

<u>Distance</u>	<u>Time After Uranium Cleanup (Days)</u>						
	1	3	7	15	30	90	400
Surface	$3.3 \times 10^{-4}$	0.13	0.76	2.8	7.2	25	84
10 cm	$2.1 \times 10^{-5}$	$8.2 \times 10^{-3}$	$4.9 \times 10^{-2}$	$1.8 \times 10^{-1}$	0.46	1.6	5.3
100 cm	$2.1 \times 10^{-6}$	$8.2 \times 10^{-4}$	$4.9 \times 10^{-3}$	$1.8 \times 10^{-2}$	$4.6 \times 10^{-2}$	0.16	0.53

Note: After multiplying by the  $\text{U}^{232}$  concentration in ppm, add 59 mr/hr at surface, 2.7 mr/hr at 10 cm or 0.27 mr/hr at 100 cm for natural thorium activity.

In addition to the above five cases, a study has been made of the re-fabrication process steps described in Sections IV and V to evaluate the dose rates to be expected for contamination levels of both 200 and 400 ppm  $\text{U}^{232}$  in the uranium. The 400 ppm concentration is not expected until the third cycle of Indian Point fuel and would be a very conservative design basis; accordingly, any necessary shielding is designed for the 400 ppm level. However, the concentrations to be expected in the first few years of operations are nearer 200 ppm and so this level is used as a basis for exposure estimates. The first few years of operation can be expected to pinpoint areas of high radiation level and provide data on integrated exposure to the operators. This operating experience will provide a basis for the development of refined handling techniques and the design of any necessary shielding to operate with 400 ppm  $\text{U}^{232}$  fuel.



Continuous whole body dose rates should probably be limited to 11 mr/week or 2.5 mr/hr. Dose rates of 500 mr/week or 12.5 mr/hr can be permitted on hands and forearms. Much higher dose rates can be permitted for intermittent operations such as unloading feed-fuel carriers.

Dose rates and shielding requirements for various stages in the refabrication process are given in Table XII. A discussion of some of the results shown in Table XII follows:

Stage 1. Feed Fuel Shipping Cask.

Cask should be provided with 2.5 in. of lead during transport. Upon arrival at the plant an additional 1.75 in. of Pb should be added if long term storage is anticipated. The connections and valve handles should be shielded from the container by 2.5 in. of Pb.

Stage 13. Subassembly of 14 Con Ed fuel rods.

Some shadow shielding may be required. A 2-inch Pb shield to protect the operator's torso should be adequate to handle 400 ppm 7-day-old fuel rods. Dose rates on hands will not be excessive if tongs are used. Dose rate at the operator's head should be 2.5 - 3 mr/hr with 3-day-old fuel and 10/mr/hr with week-old fuel.

Stage 14. Final Assembly and Brazing.

A 5.8-inch shield of steel should be provided on the brazing jig. Stacking of the 14-rod subassemblies in the jig should be done behind a 2-inch Pb shadow shield.

Stage 15. Final Inspection.

During inspection the final subassembly should be shielded by 3.6-inches of Pb. Inspection will have to be carried out through small openings in the shield. An opening 3 inches wide all the way around the element should be acceptable. A means of moving the element past the opening could be provided.

Stage 16. Fuel Storage Pit.

The dry storage pit requires a 3-foot concrete top for year-old elements. A pool with 6 feet of water over the elements would also be adequate.

### Stage 17. Fuel Shipping Cask

A shipping cask with 5.0-inch Pb (or equivalent) walls will allow shipment of year-old elements with 400 ppm  $U^{232}$ .

A calculation of the radiation doses due to dust in the glove boxes indicates that this is an insignificant problem. Assuming a 0.001-inch coat of  $ThO_2$ -5%  $UO_2$  (100% dense) on the walls of a box constructed of 1/8-inch steel sheet, the dose rate at the outer surface would be 2.1, 7.9, and 20.1 mr/hr, respectively, for 7-, 14-, and 30-day-old dust of 200 ppm  $U^{232}$ . Therefore, radiation levels from reasonable thicknesses of dust can be neglected. Furthermore, it is reasonable to assume that routine decontamination (at least once a week) and properly designed boxes (coved corners, edges, etc.) will result in very low quantities of residual material.

A rough calculation was also made on the dose rate from a layer of dust on the floor and ceiling of a room. Assumptions used were a 0.001-inch thick coat of  $ThO_2$ -5%  $UO_2$  (100% dense), a concentration of 100 ppm  $U^{232}$  in the uranium, and an average age of 15 days. The variation of dose with height above floor is:

Height Above Floor (cm)	Dose Rate (mr/hr) (floor + ceiling)	D(h)/D(60cm)
1	5.7	4.9
10	3.1	2.7
30	1.8	1.5
60	1.2	1.0
122 (center of room)	1.0	0.86

Assuming a 0.001-inch thick coat of 100% dense  $ThO_2$ -5%  $UO_2$  containing 100 ppm  $U^{232}$  in the uranium that covers the walls, ceiling, and floor of an 8-foot cubical room, one may compute the dose rates as a function of time after uranium cleanup with the following results:

Time after U Cleanup (Days)	Dose Rates (mr/hr) from		
	Floor	Walls and Ceiling	Total
1	$1.6 \times 10^{-3}$	$3.2 \times 10^{-3}$	$4.3 \times 10^{-3}$
3	$5.6 \times 10^{-2}$	0.11	0.17
7	0.30	0.60	0.90
15	1.1	2.2	3.3
30	2.7	5.5	8.2
90	9.4	19.0	28.0
400	32.0	64.0	96.0

It is of interest to note that 0.001 inch of 100% dense  $\text{ThO}_2$  on a floor 8' x 8' square is 1.5 kg of  $\text{ThO}_2$ .

Calculations have been made of the dose rates to be expected for shipping containers and a storage pit for various thicknesses of shielding. The results of these calculations are presented in Tables XIII through XVI, inclusive.

TABLE IVRelative Alpha Activities of Various Isotopes

Isotope	Half-life (years)	Relative Concentration	Relative Activity <sup>(1)</sup>
Th <sup>228</sup>	1.9	6.3x10 <sup>-10</sup> (2)	2.3x10 <sup>-10</sup>
Th <sup>232</sup>	1.4x10 <sup>10</sup>	1	4.9x10 <sup>-11</sup>
U <sup>232</sup>	74	5x10 <sup>-7</sup> (3)	4.7x10 <sup>-9</sup>
U <sup>233</sup>	1.6x10 <sup>5</sup>	2x10 <sup>-2</sup> (4)	8.6x10 <sup>-8</sup>
U <sup>235</sup>	7.1x10 <sup>8</sup>	3x10 <sup>-2</sup> (4)	2.9x10 <sup>-11</sup>

- (1) Relative activity = relative concentration x 0.69/half-life.
- (2) At about 2 days after U<sup>232</sup> daughter removal, for 10 ppm of U<sup>232</sup>; 1x10<sup>-10</sup> added for Th<sup>228</sup> in natural Th.
- (3) Assumes that the U<sup>232</sup> is 10 ppm in the U.
- (4) Assumes that 40 percent of fuel is U<sup>233</sup>, and that the U constitutes 5 percent of the Th in the refabricated material.

TABLE V

Allowable Airborne Radioactivity From Fuel With 200 ppmU<sup>232</sup> in Uranium \*

Isotope	$\alpha$ Radioactivity	
	In U Only	5% U + Th
U <sup>232</sup>	$4.63 \times 10^{-3}$ c/g	$2.3 \times 10^{-4}$ c/g
Th <sup>228</sup>	$1.23 \times 10^{-3}$ c/g	$6.15 \times 10^{-5}$ c/g
Th <sup>232</sup>		$1.17 \times 10^{-7}$ c/g

## Tolerance in Air

	40 hr wk (occupational)		Outside Exclusion Area		
	$\mu\text{c/ml}$	gU/ml	$\mu\text{c/ml}$	gU/ml	g(Th+5%U)/ml
U <sup>232</sup>	$3 \times 10^{-11}$	$6.5 \times 10^{-15}$	$9 \times 10^{-13}$	$1.9 \times 10^{-16}$	$3.8 \times 10^{-15}$
Th <sup>228</sup>	$9 \times 10^{-12}$	$8.1 \times 10^{-15}$	$3 \times 10^{-13}$	$2.4 \times 10^{-16}$	$4.8 \times 10^{-15}$
Th <sup>232</sup>	$3 \times 10^{-11}$		$10^{-12}$		$8.5 \times 10^{-12}$

\* 90 days after uranium cleanup.

TABLE VI

Atoms of Th<sup>228</sup> Per Mol of Th<sup>232</sup> as a Function  
of Time After Refining and Ra<sup>228</sup> Removal

Time After Refining (yr)	Fraction of Ra <sup>228</sup> Remaining After Refining			
	0	.1	.3	.5
1	5.7 x 10 <sup>13</sup>	5.9 x 10 <sup>13</sup>	6.4 x 10 <sup>13</sup>	6.9 x 10 <sup>13</sup>
2	4.3	4.7	5.4	6.2
3	3.5	4.0	4.9	5.8
4	3.2	3.7	4.6	5.6
5	3.1	3.7	4.6	5.6
6	3.3	-	-	-
10	4.3	-	-	-

TABLE VII

Abundance and Disintegration Rates of Th<sup>228</sup> and  
Daughters in One Mol of Thorium for One ppm U<sup>232</sup>

Isotope	Half-Life		Abundance and Activity					
			Time After U <sup>233</sup> Cleanup (Days)					
			1.0	3.0	7.0	15	30	90
Th <sup>228</sup>	1.9 yrs	atoms/mol x 10 <sup>13</sup>	1.6	4.8	12	24	48	144
		d/sec-mol x 10 <sup>5</sup>	1.8	5.4	13	27	54	162
Ra <sup>224</sup>	3.6 days	atoms/mol x 10 <sup>9</sup>	7.5	60	270	850	2100	7100
		d/sec-mol x 10 <sup>4</sup>	1.7	13	59	185	460	1570
Em <sup>220</sup>	55 sec	d/sec-mol x 10 <sup>4</sup>	1.7	13	59	185	460	1570
Po <sup>216</sup>	.16 sec	d/sec-mol x 10 <sup>4</sup>	1.7	13	59	185	460	1570
Pb <sup>212</sup>	10.6 hrs	atom/mol x 10 <sup>9</sup>		4.4	26	97	244	850
		d/sec-mol x 10 <sup>4</sup>		8	47	174	440	1540
Bi <sup>212</sup>	60 min	d/sec-mol x 10 <sup>4</sup>		8	47	174	440	1540
Po <sup>212</sup>	sec	d/sec-mol x 10 <sup>4</sup>		5.4	31	116	290	1030
Tl <sup>208</sup>	3.1 min	d/sec-mol x 10 <sup>4</sup>		2.6	16	58	150	510

TABLE VIII

Gamma Radiation Sources from Th<sup>228</sup> and Daughters - Mev/Disintegration

Isotope	Gamma Energy Groups (Mev)						
	0-0.1	0.1-0.3	0.3-0.7	0.7-1	1-2	2-2.5	2.5-3
Th <sup>228</sup>	0.024						
Ra <sup>224</sup>		0.011					
Em <sup>220</sup>	none						
Po <sup>216</sup>	none						
Pb <sup>212</sup>		0.206					
Bi <sup>212</sup>	0.009	0.0047		0.190	0.160	0.102	
Tl <sup>208</sup>		0.028	0.568	0.129			2.62
Po <sup>212</sup>	none						

TABLE IX

Alpha Particle Production from Th<sup>228</sup> and Daughters

Isotope	Alpha Energy (Mev)	Yield/Dis. (%)
Th <sup>228</sup>	5.42	72
	5.338	28
Ra <sup>224</sup>	5.681	95
	5.448	4.6
	5.194	0.4
Em <sup>220</sup>	6.282	100
Po <sup>216</sup>	6.774	100
Bi <sup>212</sup>	6.086	27.2*
	6.047	69.9
	5.765	1.7
	5.5	1.2

\* Note: Yield per alpha disintegration - only 33.7% of Bi<sup>212</sup> disintegrations produce alphas.



TABLE XBeta Particle Production from Daughters of Th<sup>228</sup>

Isotope	Energy (Mev)	Yield/Dis. (%)
Pb <sup>212</sup>	0.36	80
	0.57	20
Bi <sup>212</sup>	2.25	66.3

TABLE XIGamma Radiation Sources in Mev/Sec due to One ppm of U<sup>232</sup> in One Gram of Uranium for Various Times After Clean-up of the Uranium

Time (Days)	Energy Group (Mev)						
	0-0.1	0.1-0.3	0.3-0.7	0.7-1	1-2	2-2.5	2.5-3
1	18	0.8					
3	59	126	63	79	54	35	288
7	152	4.68x10 <sup>2</sup>	3.9x10 <sup>2</sup>	4.7x10 <sup>2</sup>	3.2x10 <sup>2</sup>	2.04x10 <sup>2</sup>	1.81x10 <sup>3</sup>
15	347	1.74x10 <sup>3</sup>	1.42x10 <sup>3</sup>	1.75x10 <sup>3</sup>	1.2x10 <sup>3</sup>	7.65x10 <sup>2</sup>	6.55x10 <sup>3</sup>
30	730	4.4x10 <sup>3</sup>	3.69x10 <sup>3</sup>	4.44x10 <sup>3</sup>	3.04x10 <sup>3</sup>	1.9x10 <sup>3</sup>	1.70x10 <sup>4</sup>
90	2.27x10 <sup>3</sup>	1.53x10 <sup>4</sup>	1.25x10 <sup>4</sup>	1.54x10 <sup>4</sup>	1.06x10 <sup>4</sup>	6.7x10 <sup>3</sup>	5.76x10 <sup>4</sup>
400	7.40x10 <sup>3</sup>	5.17x10 <sup>4</sup>	4.24x10 <sup>4</sup>	5.226x10 <sup>4</sup>	3.59x10 <sup>4</sup>	2.286x10 <sup>4</sup>	1.967x10 <sup>5</sup>

TABLE XII

Radiation Levels To Be Expected At Various Stages  
In the Conceptual Fuel Refabrication Plant

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Stage	Material	U <sup>232</sup>	Fuel Age (days)	Shielding	Dose Rate (mr/hr)					
					Surface	10cm	1 ft	2 ft	1 meter	
1. Feed Fuel  Shipping Cask	1050 gU at 120 g/l in	200 ppm	30	2" Pb	100			33		
	3.7" cylinder	200 ppm	400	3.7" Pb	100			33		
		400	30	2.5" Pb	100			33		
		400	400	4.25" Pb	100			33		
2. Glove Box For Carrier Unload	Same as (1)				100			33		
3. Solvent Extraction Units	9 gU/l in all units	400	400	2' concrete	0.5					
4. Precipitation Tank	1.32 KgU + 10 Kg ThO <sub>2</sub>	200	<2	none					0.20	
		400	<2	none					0.20	
	"	200	14	none					7.6	
	"	400	14	none					15	
	"	200	30	none					20	
	"	400	30	none					39.8	
	1.32 Kg U + 50 Kg ThO <sub>2</sub>	200	<2	none						0.58
		400	<2	none						0.58
	"	200	14	none					4.9	
	"	400	14	none					9.2	
"	200	30	none					12.2		
"	400	30	none					23.8		

TABLE XII cont'd

Stage	Material	U <sup>232</sup>	Fuel Age (days)	Shielding	Dose Rate (mr/hr)					
					Surface	10cm	1 ft	2 ft	1 meter	
5. Filter	Same as (4)									
6. Rotary Kiln	Same as (4)									
7. Pelletizing Machine	Same as (4)									
8. Boat Transfer	1 Kg Pellets	200	3	none		4.4		0.1		
	"	400	3			6.0		0.14		
9. Inspection	1/2" rod of 5% UO <sub>2</sub> + ThO <sub>2</sub>	400	3		94	5.9	1.9			
		200	3		68	4.3	1.4			
		200	7		194	12.5	4.1			
		400	7		346	22.3	7.3			
10. Grinding	Same as (9)									
11. Rod Loding	Same as (9)									
12. Plug Welding	Same as (9)									
13. Sub Assembly	14 Con Ed Rods	200	3			22.8	7.4	3.7	2.4	
		400	3			31.3	10.1	5.1	3.3	
14. Final Assembly	1 Con Ed Element 7% UO <sub>2</sub>	200	3	none			65		19.8	
		400	3	none			81		24.6	
		200	3	2.6" Pb or 4.2" Fe				2.5		
		400	3	2.75" Pb or 4.44" Fe				2.5		

TABLE XII cont'd

Stage	Material	U <sup>232</sup>	Fuel Age (days)	Shielding	Dose Rate (mr/hr)				
					Surface	10cm	1 ft	2 ft	1 meter
14. (cont)	1 Con Ed Element 7% UO <sub>2</sub>	200	7	none			144	72	44
		400	7	none			240	120	73
		200	7	3.2" Pb or 5.2" Fe			2.5		
		400	7	3.6" Pb or 5.8" Fe			2.5		
15. Final Inspection	1 Con Ed Element	Same as (14)							
16. Fuel Element Storage Pit	Array of Con Ed Elements	200	400	2.75' concrete	7.5				
		400	400	3' concrete	7.5				
17. Fuel Element Shipping Cask	4 Con Ed Elements	400	30	3.3" Pb	100				
		400	400	5.0" Pb	100				

TABLE XIII

Dose Rate At Surface of Unshielded 3.7-inch Diameter Shipping  
Container for Uranyl Solution Containing 200 ppm U<sup>232</sup> in Uranium

Age of Fuel (Days)	Concentration (g/l)	Dose Rate at Surface (mr/hr)
30	40	940
30	120	2,820
400	40	11,000
400	120	33,000

TABLE XIV

Thickness of Concrete for Various Dose Rates at Surface of Shielding  
On the Fuel Storage Pit and Extractor with 400-Day-Old Fuel  
Containing 200 ppm U<sup>232</sup> in Uranium

Dose Rate (mr/hr)	Concrete Thickness (feet)	
	Extractor	Storage Pit
100	0.3	1.5
7.5	1.05	2.65
0.75	1.7	3.0

TABLE XV

Dose Rate vs Lead Thickness at Surface of Shielding on 3.7-inch  
ID Shipping Container for Uranyl Solution Containing  
200 ppm U<sup>232</sup> in Uranium

Age of Fuel (Days)	Concentration (g/l)	Dose Rate at Surface (mr/hr)	Pb Thickness (inches)
30	40	100	1.2
30	40	7.5	3.1
30	40	0.75	4.7
400	40	100	2.95
400	40	7.5	4.8
400	40	0.75	6.4
30	120	100	2.0
30	120	7.5	3.8
30	120	0.75	5.4
400	120	100	3.5
400	120	7.5	5.6
400	120	0.75	7.2

TABLE XVI

Dose Rate vs Lead Thickness at Surface of Shielding on Indian  
Point Fuel Elements with 200 ppm U<sup>232</sup> in Uranium

Age of Fuel (Days)	No. of Elements in Cask	Dose Rate (mr/hr)	Pb Thickness (inches)
30	1	3,300	0
30	1	100	2.25
30	4	100	2.75
30	1	7.5	4.2
30	4	7.5	4.7
400	1	38,300	0
400	1	100	4.0
400	4	100	4.5
400	1	7.5	5.9
400	4	7.5	6.4

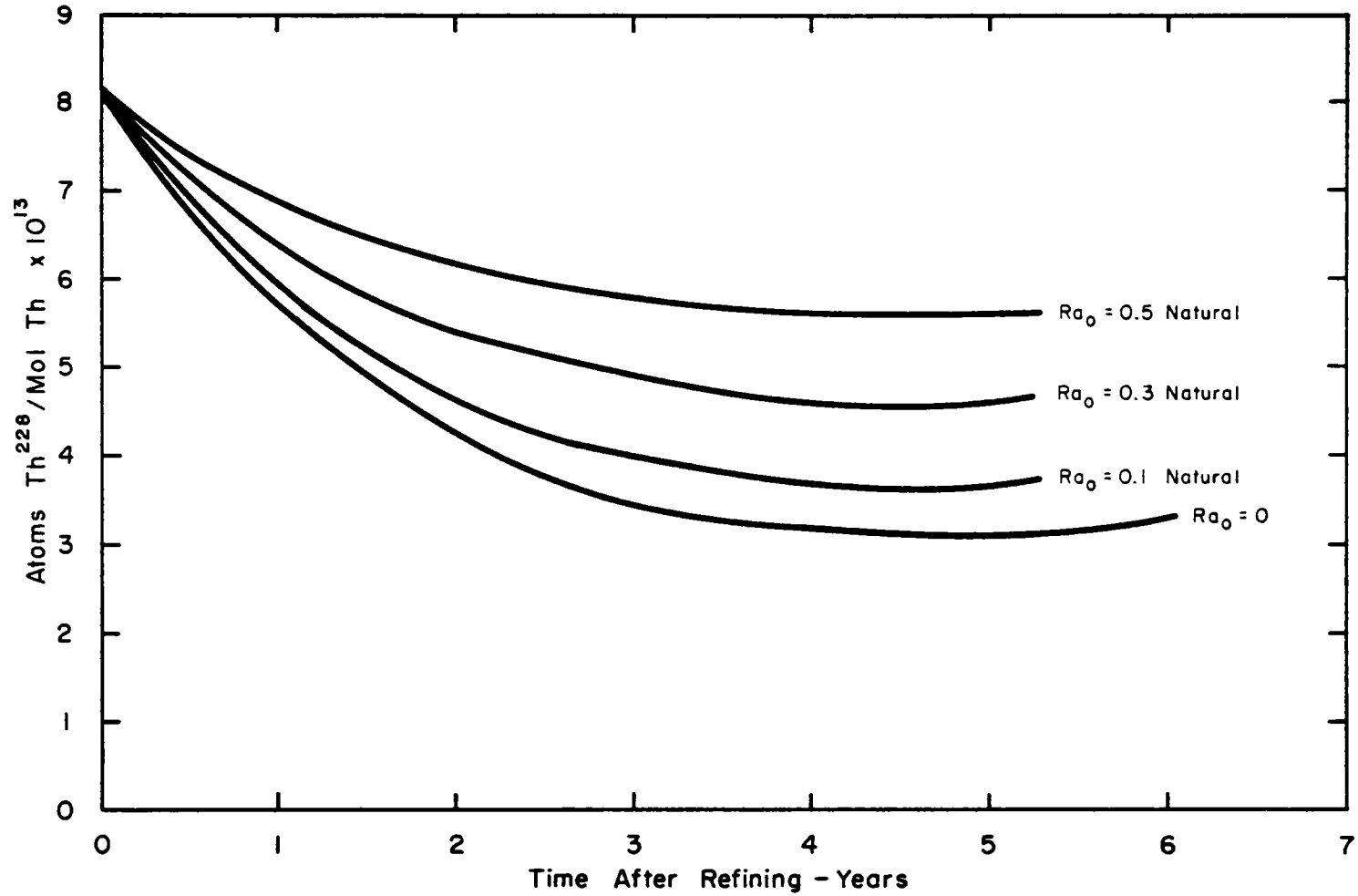


Fig. 16 - Atoms  $\text{Th}^{228}$  / mol thorium as a function of  $\text{Ra}^{228}$  removal and time after refining.  
Drawing No. RA-2356



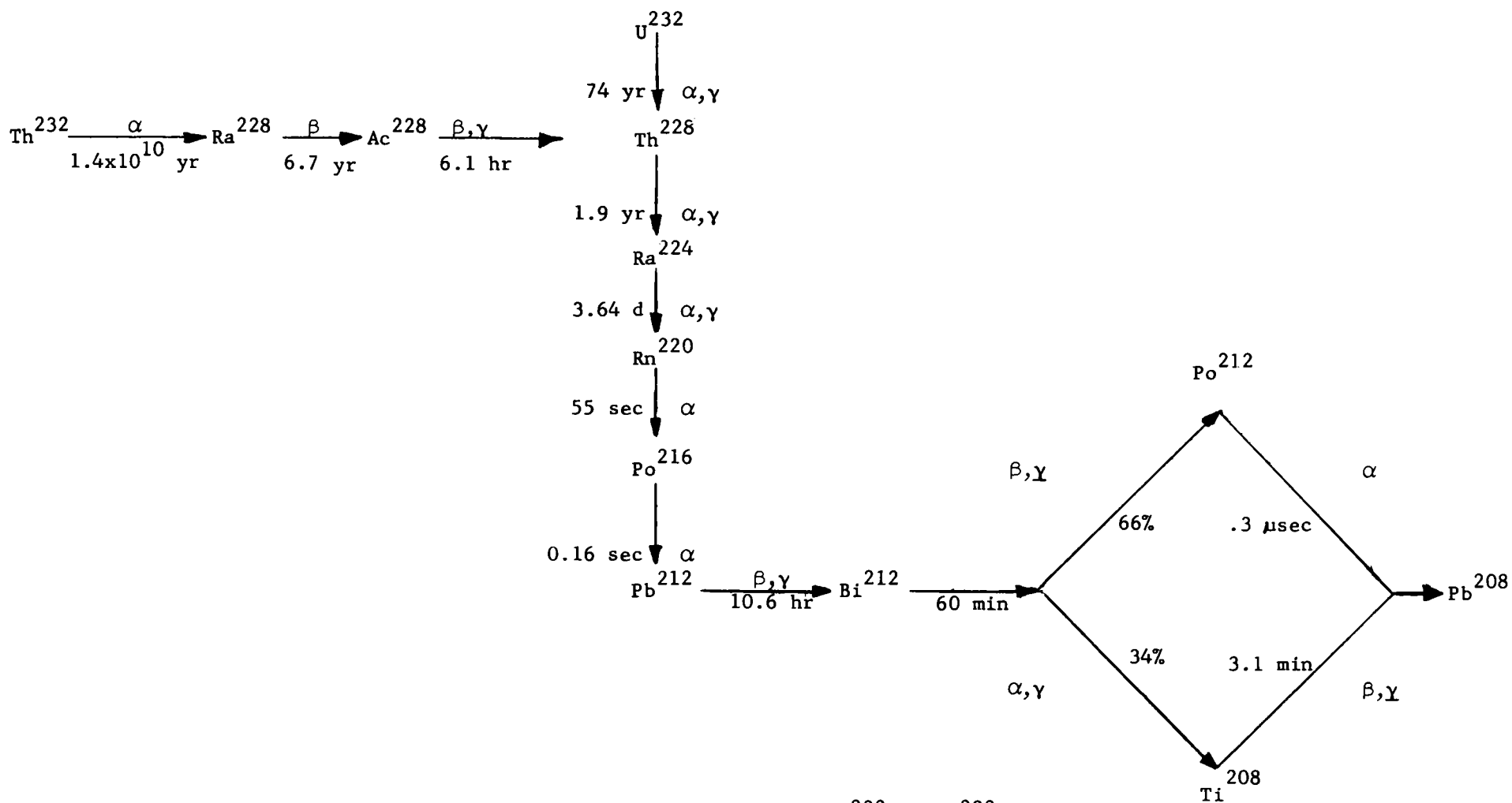


Fig. 17 - Decay Schemes for  $\text{Th}^{232}$  and  $\text{U}^{232}$

Hard gamma activity comes primarily from the  $\text{Bi}^{212} \longrightarrow \text{Po}^{212}$  with energies in the range of 0.8 to 2.2 Mev, and from  $\text{Tl}^{208} \longrightarrow \text{Pb}^{208}$  with 2.6 Mev energies.

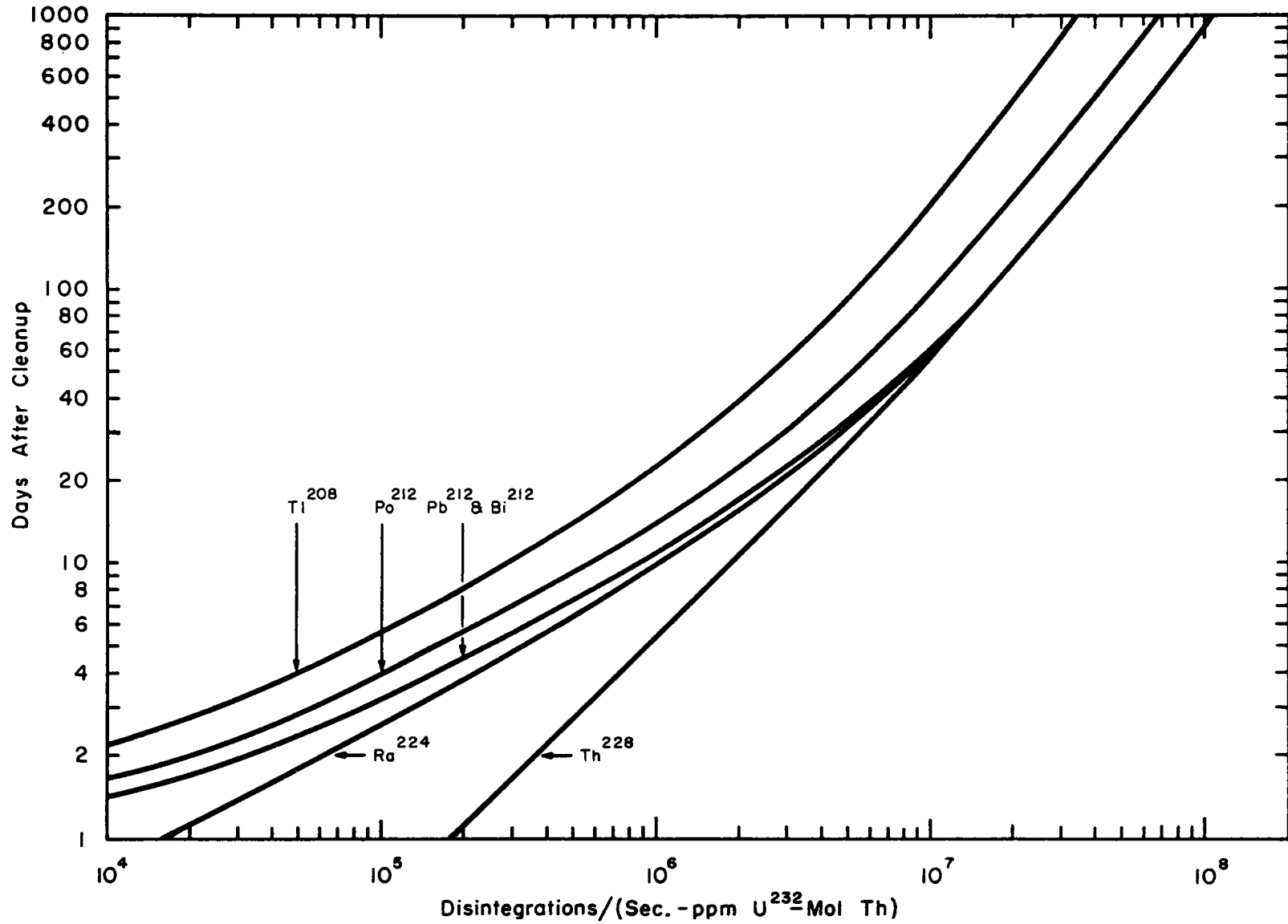


Fig. 18 - Activity of the Th<sup>228</sup> chain vs time after U<sup>232</sup> cleanup.  
Drawing No. RA-2357

## VII. RECOMMENDATIONS FOR FUTURE WORK

The results of this program have been discussed at the American Nuclear Society 1961 annual meeting in Pittsburgh, and at an informal meeting in Washington on August 3, 1961 with representatives of the Oak Ridge National Laboratory. In the first instance, it was suggested that we might be able to adapt a more simple method of oxide preparation, i.e. flame denitration, to our process. We have reviewed this suggestion in some detail, and find that it has merit. In the second instance, contrary experience was cited in the matter of equipment contamination, although the applicability of this experience to the present case is questionable.

As an outgrowth of our program and of these discussions, we have delineated several areas in which further work appears warranted.

### A. Oxide Preparation

There are many possible advantages which could result from an adaptation of the Mallinckrodt Chemical Works flame denitration process to the direct-access fabrication procedure. By using  $H_2$  and  $O_2$  instead of propane and air, it may be possible to eliminate off-gases and produce a suitable oxide product directly from solution. This would eliminate the precipitation, filtration, and firing steps from the process as previously defined. The time saved by eliminating these steps is estimated to be about 7 hours, or nearly 10 percent of the total process time from the output of fuel cleanup to inspection of final element.

Although our study assumed the production of pellets, for the sake of being conservative in demonstrating the feasibility of rapid refabrication, a more flexible approach is advantageous. We understand that it is practical to arc-fuse the freshly prepared oxide to produce material suitable for vibratory compaction, and that the addition of an arc-fusing step to a modest size plant would probably cost in the neighborhood of \$100,000. Thus, with little additional effort, a plant could be designed to produce oxide that would be suitable either for the production of pellets or for vibratory compaction, using essentially one basic process.

B. Separation of U<sup>232</sup> Daughters

On the basis of one-stage separation factors, we have satisfied ourselves that essentially complete separation would be feasible and efficient using simple equipment of small size. However, there have been questions, based on thorex experience (only roughly analogous), as to speed, effectiveness, and adaptability of our solvent-extraction process. Also, the initial feasibility work was based on the use of high chloride-content solutions; we believe it to be desirable and feasible to adapt the process so that the product will be in the form of nitrate solutions.

C. Over-all Feasibility

Questions have been raised with regard to the over-all feasibility of direct fabrication. The validity of our analysis is questioned, not on its own merits or lack thereof, but rather from considerations of possible contrary results based on experience in remote fabrication that may or may not be applicable in direct fabrication.

Doubts have been raised concerning the maintenance of low radiation levels in the working areas over a long period of time. Our contention is that routine, frequent decontamination (as opposed to one-shot cleanup of more heavily contaminated equipment) and use of equipment specially designed for ease of decontamination (as opposed to standard equipment selected for its primary functions only) can maintain radiation levels below the accepted tolerances over lengthy periods.

Theoretical work that is contrary to pertinent experience should always be questioned; however, in this particular case we are not convinced of the applicability of the experience cited. We believe that it would be prudent, nevertheless, to:

1. Review our analysis, calculating doses on various operating personnel in some detail, contemplating in depth the consequences of improbable but possible accidents, examining closely the procedure for unexpected time delays for which provisions might not have been made, etc.

2. Design an experiment which would use  $U^{232}$  to determine experimentally the buildup of activity in the deposits of thoria-urania mixtures passing through simulated processing equipment.

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A. Forcella (Comitato Nazionale Per L'Energia Nucleare) and J. K. Davidson (Allis-Chalmers Manufacturing Company), Nuclear Fuel Cycles and the PCUT Program.

J. K. Davidson and A. C. Schafer (Allis-Chalmers Manufacturing Company), G. Orsenigo and F. Fumagalli (Comitato Nazionale Per L'Energia Nucleare), Chemical Reprocessing in CNEN's PCUT Pilot Plant.

J. K. Davidson, O. Drolshagen and A. C. Schafer (Allis-Chalmers Manufacturing Company), O. Nepi and G. Orsenigo (Comitato Nazionale Per L'Energia Nucleare), The PCUT Facility for Reprocessing and Refabricating  $ThO_2$ - $UO_2$  Fuels.

O. Drolshagen (Allis-Chalmers Manufacturing Company), F. Gazzabin (Bombrini Parodi-Delfino), and S. Russo (Comitato Nazionale Per L'Energia Nucleare), Mechanical Equipment in the PCUT Pilot Plant.

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