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**RICHLAND FIVE-YEAR
02 R&D PROGRAM**

ENRICHED FUEL PROCESSING

**RICHLAND OPERATIONS OFFICE
ATLANTIC RICHFELD HANFORD COMPANY
BATTELLE-NORTHWEST
DOUGLAS UNITED NUCLEAR, INC.**

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W. S. Bryan	AEC
D. H. Curtiss	DUN
A. M. Platt	BNW
H. F. Tew	DUN

COMPILED BY

H. C. Rathvon	ARHCO
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R I C H L A N D F I V E - Y E A R
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ENRICHED FUEL PROCESSINGINTRODUCTION

In anticipation of a continuing trend for reductions in military plutonium requirements, reactor programs at the Richland site are being directed toward other long-term production objectives. For the general case of an alternate reactor product, uranium-238 would be displaced from the reactor by another target material, e.g., lithium (for tritium production) or neptunium-237 (for plutonium-238 production), and the remaining fuel would require higher enrichment (increase of uranium-235 concentration) to maintain reactor reactivity. The objectives of several Richland research and development programs reflect this trend toward new reactor products and are associated with the requirement for operation with higher enrichment fuels, for example:

Mission 2 - Coproduct
Mission 3 - Transplutonium
Mission 4 - Pu-238
Mission 7 - Target Space Enhancement

The operating fuel reprocessing facilities at Hanford were originally designed for the processing of fuels containing less than one percent U-235 (pre-irradiation basis). Today, limited amounts of a "spike" fuel, averaging about 1.15 percent U-235, are included in the production load, and demonstration quantities of 2.1 percent enriched coproduct fuels have been processed under special test support conditions. Over-all, however, the Hanford manufacturing program is primarily associated with military plutonium production.

Anticipated reactor programs requiring higher enrichment fuels pose new problems of reprocessing technology. These problems have their bases in the increased U-235 content of the fuel, and in the material and design features provided to obtain a higher specific power in the reactor. The programs required to develop the technological bases for reprocessing proposed Hanford fuels of greater enrichments, generally in excess of one percent U-235, are described by this document.

SCOPE AND OBJECTIVES**DECLASSIFIED**

The research and development objective for this mission is to provide the scientific and engineering bases for safely and economically reprocessing the enriched fuels being considered for use in the Richland reactors. The direction of reactor-sponsored enriched fuel programs to date has classified most fuels into two principal categories of ARHCO investigation:

1. Fuels of moderately low enrichment include those enriched to greater than one percent U-235. The N-Reactor Coproduct demonstration fuel, enriched to 2.1 percent U-235, is of particular current interest in this group. Also, slightly higher and lower enrichments have been the subjects of various studies.

2. Highly enriched fuels are those containing uranium of a very high U-235 percentage, e.g., Oralloy, at 93 percent, which is alloyed with a non-fertile diluent. Substitution of fissile plutonium for U-235 has also been considered.

The scope and objectives of the ARHCO program are designed to be consistent with the programs and functions of other Richland contractors. An important consideration is the magnitude and diversity of the total reprocessing load, since the associated target reactor products, as well as the driver fuels, may also require chemical processing in the same or similar facilities. Consistent with a growing technology and changing business climate, a chemical processing capability for a large and diverse load is a presumed requirement.

The near-term goals of this mission are to identify the facility changes that would be required to process irradiated fuels for each enrichment of interest, to estimate the time and cost of effecting these changes, and to estimate the processing capability and operating costs of the revised facilities, when processing fuels of the various enrichments. These data, when combined with similar data pertaining to the rest of the fuel cycle, will provide a basis for optimizing the modification and operation of the AEC production complex. Increased enrichment of fuels and associated changes of fuel design can drastically and adversely affect the reprocessing capabilities of the ARHCO and other feed-chain facilities. Potential effects are sufficiently large to be an important consideration to the overall AEC production program. At Richland, the basic separations plant problem results from the increased complexity of receiving and decladding projected fuels, the decreased critical mass and critical concentration values for the higher fuel enrichments, and their effect on the processing capability of existing facilities. Similarly, in the production of UO_3 from the separations plant aqueous nitrate product, the existing facilities, featuring continuous trough calciners, and large equipment for bulk handling of product require modifications to be critically safe at higher enrichments.

The broad scope of the ARHCO R&D programs includes the technology of the chemical reprocessing steps commencing with the dispatch of the irradiated fuels from the reactor sites. The processing steps subsequent to fuel receipt include the separation of desired products from cladding materials and fission products, the production of purified product streams, U, Pu, and Np, and the preparation of waste streams for disposal in accordance with the requirements of acceptable waste management practices. Product forms acceptable for transport and consistent with normal customer requirements are to be produced. Most of the ARHCO effort has stressed the development and application of technology to existing ARHCO facilities or adaptations of existing facilities. However, the advantages and possible need for new technology and facilities concepts are to be considered and developed as justified. A current knowledge of the research and development efforts of others will be maintained.

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INCENTIVES

The incentive for developing ARHCO capability for processing higher enrichment fuels stems from the need for establishing Richland capability for producing new, marketable reactor products. Whereas, ARHCO facilities are basically limited to the routine production processing of uranium fuels containing not greater than one percent U-235, the Richland reactors could effectively employ fuels of greater enrichment and use the developed space to irradiate targets other than uranium-238. A safe and economical way of providing and reprocessing fuels permitting a higher effective enrichment in the Richland reactors would permit the large-scale production of isotopes other than plutonium-239 and extend the useful life of the Richland production complex.

PROGRESS DURING REPORT PERIOD

1. Low Enrichment Fuels

Low enrichment fuel reprocessing development derived substantial benefits from the studies and experience associated with the initial processing of Coproduct demonstration fuels (100 tons of 2.1 percent U-235; Zircaloy-clad), which was completed at the Purex plant during the year. The chemical flowsheet for this campaign featured the Zirflex processing of the Coproduct fuels in a single Purex dissolver equipped with a boron-stainless steel insert for maintaining criticality control. To minimize capital cost requirements for the demonstration, the other two Purex dissolvers were operated in parallel for the processing of normal aluminum-clad 94 Metal and the dissolver solutions from all three dissolvers were blended to an enrichment just below one percent U-235 prior to solvent-extraction processing. Process performance during the operation was excellent and no control problems were experienced.

Dissolver charging of Coproduct fuels was successfully accomplished during the campaign by means of a mechanical charging machine prototype designed and constructed for this purpose. This machine inverts each fuel canister and drops individual elements into the dissolver. The quantity of fuel charged into the restricted annulus averaged about 3/4 of the predicted value (6.8 tons) because of a) an apparent 75 percent void space between the elements as compared with 60 percent void measured in mock-up tests; and b) variations in the stacked fuel height. The problem of variable stacked height was reduced by adjustments to operating practices. The overall processing rate for a single dissolver remained at about 2.18 tons/day as compared with 3.0 tons/day originally predicted.

A criticality analysis of the Coproduct Demonstration processing proposal was completed prior to the start of the campaign. This analysis concluded that the processing was feasible within acceptable safety criteria and risks. The analysis utilized recently developed laboratory measurements of the densities of various uranium precipitates, and concluded that the maximum

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allowable U-235 concentration (or equivalent reactivity) in the Purex Plant, as currently instrumented and operated, is 1.78 percent U-235. On this basis, unusual criticality control requirements (e.g., nuclear blanks) were necessary only at process points upstream of the initial solution blending vessel, where the equivalent reactivities might be greater than 1.78 percent U-235.

Also, during the Coproduct Demonstration campaign, a prototype Purex head-end sequential controller was successfully demonstrated. This prototype was designed to provide the improved assurance of control, principally criticality control, that is necessary when processing higher enrichment fuels in conjunction with the added variations in chemical flowsheets in the head-end (dissolver) system. The prototype controller is an instrument system that a) groups the solution transfer controls required for head-end processing into "blocks" corresponding to specific operations; and b) employs interlocks to permit operation of the desired "block", only.

Corrosion studies were also included as part of the Coproduct Demonstration campaign, to verify corrosion data, in general, for the Zirflex processing, and, more specifically, to investigate the structural integrity of the boron-stainless-steel dissolver insert. Corrosion coupons of 304-L stainless steel, with and without 1.3 weight percent added boron, had been inserted in the Coproduct dissolver prior to the campaign. Analyses of the data are not yet complete; however, average corrosion rates for the entire Zirflex cycle were approximately as predicted by laboratory data. The rate for boron-stainless steel was about 12 mils per month, compared to 2.5 mils per month for ordinary 304-L. These data indicate that the insert integrity is conservatively adequate for the balance of the Coproduct Demonstration load (300 additional tons of fuel) which will be processed during FY-1969. Visual inspection also indicated the insert annulus to be in good condition.

Purex Plant cost-capability relationships for low enrichment fuels (to 2.1 percent U-235) have been further refined primarily as a result of operating experiences with the Coproduct Demonstration processing and with the Zirflex processing of Zircaloy-clad fuels in general. As a result of experiences in the use of a dissolver insert, a more realistic application of criticality safety criteria was developed (e.g., an insert width designed to limit k_{eff} to 0.98 in case of double batching), and fuel element packing density data for the actual combination of charging method and "as built" insert design was gained. This background of experience has been extrapolated to the processing of future fuel types, permitting more reliable estimates of capability (capacity factor) versus capital expenditures corresponding to particular facilities improvements. For example, the sustained processing of Coproduct (2.1%) fuels (Mark V, advanced design) could be accomplished at an estimated 1.18 CF (9.8 tons/day, instantaneous), assuming a limited capital expenditure of \$640,000 for dissolver units and auxiliaries, cask cars, and UNH product shipping cars. The addition of modifications to the Purex Plant E-Cell (centrifugation) equipment for an incremental additional capital expenditure of \$100,000 would be expected to increase this processing rate to 1.29 CF.

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Revisions of cost-capability relationships previously made are continuing; it is expected that a reliable analysis will be completed and published early during FY-1969.

As a possible route to significant improvement in the Purex Plant capability for processing higher enrichment fuels (or normal enrichment, Zircaloy-clad fuels), the development of a crush-leach head-end concept was commenced. Initially, a simple short-stroke press was used to crush single N-Reactor fuels. By this technique, the outer tubes of unirradiated and irradiated fuel elements were crushed by the application of 150 tons force against the outside surfaces. Further work to demonstrate other crushing techniques is in progress. Also, conceptual design studies of crusher-dissolver arrangements have commenced.

Previously initiated studies which had been designed to develop techniques for penetrating the refractory oxide coating on irradiated Zircaloy-clad fuels were cancelled, at least temporarily, as a result of Purex Plant Zirflex processing experience. Apparently, under the conditions of charging N-Reactor Zircaloy-clad fuels at Purex, the oxide coating is damaged sufficiently to permit relatively rapid dissolution of the cladding.

Laboratory scale investigations were commenced to characterize the freezing and melting behavior of uranyl nitrate hexahydrate (UNH) relative to the mechanical problems of loading and unloading tank car volumes of the material, at, or near, the 100% UNH concentration. Two characteristics of the system were investigated--the thermal conductivity of UNH crystals, and the volume changes associated with the solid-liquid phase transition. In addition, the study confirmed that, upon freezing, stoichiometric UNH precipitates quantities of uranyl nitrate trihydrate $[\text{UO}_2(\text{NO}_3)_2 \cdot 3\text{H}_2\text{O}]$ which could be undesirable because of its high melting point. Trihydrate material did not precipitate during freezing of solutions with uranyl nitrate concentrations less than 76 weight-%. Further laboratory investigations will continue as needed to support engineering studies of the UNH shipping system, its problems, and hazards.

2. Highly Enriched Fuels

Concepts for the processing of highly enriched fuels were developed and evaluated on the basis of studies of the available technical literature as applied to fuel alloys and fuel designs considered likely for application in Hanford reactors. Following the initial engineering studies, supporting laboratory investigations were defined to provide a basis for more definitive flowsheet concepts. The over-all studies included preliminary evaluation of the impact of highly enriched fuel processing of the ARHCO waste management program, and the capital and operating cost requirements for typical processing loads of interest.

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For purposes of the ARHCO studies, two basic highly enriched fuel concepts were assumed as follows:

- a) A tubular fuel, approximately 2.5 inches in diameter, fabricated from alloy containing 10 to 40 weight percent Orallo (93% U-235) in Zircaloy (with Zircaloy cladding) and 200 grams of U-235 per foot of element length; and
- b) A tubular fuel, 1.5 to 2.0 inches in diameter, fabricated from alloy containing 10 to 20 weight percent Orallo in aluminum (with aluminum cladding) and 70 to 100 grams of U-235 per foot of fuel length.

For the particular uranium-zirconium alloys of interest, the modified Zirflex and the Alniflex processes were initially considered for Purex dissolution application. However, because of probable corrosion and criticality control problems associated with the modified Zirflex process, particularly for alloys containing greater than 10 percent uranium, the Alniflex process was selected for initial development and as the basis for the engineering studies. This process uses $\text{HF-HNO}_3\text{-Al}(\text{NO}_3)_3\text{-(NH}_4)_2\text{Cr}_2\text{O}_7$ as the dissolver, typically either at 70°C or at boiling. The dissolver construction materials can be 304-L stainless steel for 70°C operation. However, considerable benefit in dissolution rate would be expected by operation at the more corrosive conditions of the higher (boiling) temperature and the economic incentive for this mode of operation in a new, corrosion-resistant dissolver will be studied. On the basis of the data presently available, reliable estimates of the dissolution rate are not possible; the best current guess is 1,000 to 2,000 pounds per day of zirconium, total for the three existing Purex dissolvers fitted with special inserts and operating at 70°C.

For uranium-aluminum alloy fuel elements, the demonstrated mercury-catalyzed nitric acid dissolution flowsheet would be adopted at Purex. Reaction-rate control by controlling the depth of fuel element immersion is an approach of possible merit for application at Purex.

Thus far, studies of the methods for transporting highly enriched fuels from the reactors to the separations plant and for charging the dissolvers have resulted primarily in a definition of problems. For example, by current Hanford fuel handling philosophy, the amount of fissile material contained in a single container that might be accidentally spilled, e.g., a canister or a slug bucket, must be less than a critical mass. For the conceptual fuel designs that have been considered by ARHCO studies, the amount of U-235 per container would be limited to about a kilogram, maximum. Assuming the current system for dissolver charging by canyon crane at the Purex Plant, a processing rate restriction resulting from the charging time requirements is implied. Because of the probable short lengths (less than 12 inches) of Hanford reactor fuels, the criterion of less than one critical mass per container is expected to complicate dissolver charging operations, possibly causing the system to become rate limiting. However, with a

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long fuel element, as a geometrically-safe cylinder, a single crane operation could carry a greater amount of U-235, and the amount of time required to complete a given charge size, would be proportionately decreased. This potential problem and its solution will require further study.

An economical and practicable approach to the adaptation of the current Purex annular dissolvers to highly enriched fuels processing might be accomplished by providing an insert of geometrically favorable tubular fuel receptacles, appropriately spaced within the existing dissolver charging annulus. Each barrel would make a transition at the lower end into a geometrically favorable annular slab, which could be designed to receive any fissile particulates in a nuclearly safe geometry. The barrel diameter and spacing would be based on full moderation of the dissolver charge.

Possible solvent-extraction concepts for application at Purex include a) the use of large-diameter pulse columns with a dilute TBP (1.5 to 3 percent) solvent, for criticality control; or b) the use of small-diameter geometrically favorable pulse columns operating with the normal 30 percent TBP solvent. Either approach could be employed at the Purex Plant, either for U-Al or U-Zr alloy fuels. Studies indicated, however, that the large-diameter columns, with a dilute TBP flowsheet, would have the following disadvantages:

- a) The minimum capacity of the Purex HA Column with dilute TBP would probably be very close to the maximum capacity of the dissolving operation; thus, the HA Column may not be practicably operable without modifications that would be undesirable for other Purex processing requirements.
- b) Desirable processing flexibility would probably be unattainable in the First Cycle Columns; viz., for practicable flowsheet concepts, column capacities do not consistently match.
- c) Product concentration would require a large capital cost for equipment.

The problems associated with the use of a 30 percent TBP flowsheet in geometrically favorable columns appear to be more amenable to economical solution; consequently, this approach has been selected for further study.

The currently favored solvent-extraction flowsheet would include the use of the following equipment systems:

- a) The existing neptunium recovery pulse columns (J-Cell Package) would become the First Uranium Cycle;
- b) The existing second plutonium cycle pulse column (L-Cell) would become the Second Uranium Cycle;
- c) The pulse column package currently being proposed as a third plutonium cycle would become the Third Uranium Cycle;
- d) A new stripper-concentrator and product loadout system would be provided in order to avoid the cross-contamination between uranium and plutonium that would occur in a common system.

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The solvent-extraction processing rate for the system described is estimated to be about 150 kg/day of uranium assuming an aluminum or zirconium fuel alloy of approximately 15 percent uranium, minimum (total diluent basis, including cladding). For "leaner" alloys the processing rate would be expected to be proportionately lower. Verification or refinement of these estimates is one element of the current laboratory development program.

The major uncertainty of any solvent-extraction flowsheet application to the Purex Plant is the method for neptunium recovery. Probably an oxidizing flowsheet (nitrite or dichromate) is preferable to a reducing flowsheet (ferrous iron); further study and flowsheet development will be required prior to process selection.

Current concepts for waste processing and acid recovery for U-Al and U-Zr alloy fuels are summarized briefly as follows:

- a) For U-Zr alloy fuel, the First Cycle raffinate containing essentially all of the fission products probably will not be concentrated in the Purex Plant because of the instability of the solution with respect to precipitation, particularly if heated. A waste of the following approximate composition is visualized:

<u>Component</u>	<u>Concentration, M</u>
Al	0.6
Hg	0.005
Zr	0.3
F	2
HNO ₃	0.6

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At this composition, the volume per ton of Zr processed would be approximately 8,800 gallons. The Second and Third Cycle waste would be treated for acid recovery and for campaign recovery of neptunium and plutonium values.

- b) For U-Al alloy fuel, the First Cycle raffinate would be treated for acid recovery. The Second and Third Cycle wastes would also be treated for acid recovery, and a combined waste concentrate of approximately the following composition would be discharged from the plant:

<u>Component</u>	<u>Concentration, M</u>
Al	1.2
Hg	0.01
Na	0.003
HNO ₃	2.3

At this composition, the volume per ton of aluminum processed would be approximately 7,400 gallons.

Using concepts described above, preliminary studies have been made of facilities requirements for reprocessing conceptual production cases containing highly enriched fuels. For example, one case assumed the operation of the two Hanford K Reactors with 94 metal "spiked" with highly enriched U Al alloy. The highly enriched fuel was assumed to be an alloy containing 10 percent U-235. Allowing for the aluminum cladding, the net U-235 content was 9.1 percent. For two K Reactors, the annual load of highly enriched fuel, delivered to the Purex Plant, contained about 96 tons of aluminum and 8,000 kilograms of U-235. The processing rate at the Purex Plant was assumed to be 100 kg of U-235 per day, instantaneous. Conceptual facilities modifications included the following:

1. Purex Plant facilities requirements were assumed to include a) new railroad cars, casks, and cask inserts; b) a car cleaning station; c) dissolver inserts and storage boxes; d) Purex sump modifications; e) new instrumentation and jumpers; f) a product loadout system. These changes and additions were estimated to cost about \$2.5 million.
2. Assuming the desirability of "closed cycle" operation with respect to the Hanford site, facilities could be provided adjacent to Purex Plant for producing U-Al alloy ingots to be refabricated to fuel elements. Facility costs of \$1.5 million are visualized.
3. Waste management costs would depend on the need to provide new boiling waste storage tanks to handle the relatively larger salt volume associated with the fission products of a U-Al alloy fuel. If existing tanks can be programmed to accommodate the highly enriched fuel processing, Waste Management facilities requirements would be limited to lag storage volume (stainless-steel tanks in a vault) between the Purex Plant and B-Plant, plus instrumentation and piping changes at B-Plant to accommodate the higher flow rate requirements of the conceptual waste. These requirements were estimated to cost about \$3 million.

For the processing of U-Zr alloy fuels, and assuming a reasonable fuel design (high U/Zr ratio) Purex costs for comparable capability requirements might be about the same as for the U-Al fuel. At present, however, considerably more uncertainty exists in the U-Zr alloy processing concepts because of the status of technology. The application of the B-Plant waste management system to U-Zr alloy wastes is also technologically uncertain.

Laboratory efforts in support of highly enriched fuel process development include the following programs:

1. Dissolution studies are being made in support of U-Zr alloy processing. Candidate processes include the modified Zirflex, Alniflex (with modifications), and an electrochemical process. Initial work has indicated that the Alniflex process is probably most readily adaptable to the Purex Plant from the standpoint of costs, control, and safety.

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2. Corrosion studies of dissolution systems were initiated and are continuing, with emphasis on the Alniflex process. The use of 304-L stainless steel as a containment material is probably limited to a process temperature of about 70°C. Although this temperature is practicable for Alniflex processing, the superior corrosion characteristics of HAP0-20 alloy would permit operation at boiling temperatures, with a resulting increase in fuel alloy dissolution rate approximately by a factor of four.
3. Studies are being made of the adaptation of existing ARHCO waste management processes to the types of wastes that are expected. Strontium separation from simulated U-Al processing wastes appears to be practicable, either by lead sulfate carrier precipitation, or by D2EHPA solvent extraction. For a U-Zr waste, maximum strontium removal has been only about 80 percent, thus far.

The evaluation of possible non-aqueous reprocessing schemes was terminated with the conclusion that an incentive is lacking for application to a foreseeable program. With regard to halide volatility concepts, the relatively large amounts of diluent (Al or Zr) associated with conceptual fuels would require several parallel processing lines to assure nuclear safety, resulting in high capital costs. No further studies of non-aqueous systems are presently planned.

EVALUATION OF EFFORT

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1. Low Enrichment Fuels

Engineering and laboratory studies, coupled with the processing experience gained in connection with the Coproduct Demonstration reprocessing campaign, has permitted an improved assessment of required plant modifications and of Purex Plant capability for processing fuels of this type.

- a) The development of improved data has justified the use of less restrictive criticality controls than had been originally anticipated. Further refinement of data could result in additional progress in this direction.
- b) Experience with the Zirflex process, in conjunction with the dissolver insert and charging machine, has yielded data which are the basis for revised estimates of Purex Plant capability for processing variations of this type fuel. The results have been useful in the preparation of computer models of potential production modes (CAGE program).
- c) The prototype head-end sequential controller represents a "first-generation" demonstrated approach to safety devices and may be expected to lead to related and/or improved applications.
- d) The program for evaluating dissolver insert integrity, including evaluation of the corrosion data, should be of value in future design efforts, as well as in safety considerations of insert life.

The processing experience described, above, in conjunction with the laboratory and engineering studies that have been accomplished to date, provides a sound basis for definition of processing requirements for most future low enrichment fuels from current Hanford reactors that might be considered. Completion of a relatively accurate evaluation of the variables and parameters affecting reprocessing capability for various fuel designs should be possible with relatively little additional development effort.

The development of a fuel crushing concept for head-end processing remains as perhaps the single major development effort to be pursued on behalf of low enrichment fuels.

2. Highly Enriched Fuels

Studies accomplished during the past year have indicated that excellent capability for processing highly enriched U-Al and U-Zr alloy fuels in ARHCO separations facilities should be attainable at reasonable cost. Flexibility is available for the recovery of uranium, neptunium, and plutonium products. Areas of required laboratory development support have been defined, and verification of flowsheet concepts and requirements is in progress.

The principal areas of required study, with regard to separations capability and costs, appear to be in (a) the transport and handling of fuels between the reactors and separations plant, (b) the process and equipment to be used for the isolation of neptunium, and (c) the applications of current waste management concepts at reasonable cost, particularly with respect to the waste from a U-Zr alloy fuel.

BUDGET PERIOD PLANS

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1. Low Enrichment Fuels

a. Fiscal Year 1969

Engineering studies of separations cost-capability relationships will be completed and documented. Data obtained during plant processing operations conducted in FY-1968 and scheduled in FY-1969 will be included as applicable. Possible benefits from the use of soluble poisons as a secondary criticality control will be evaluated. Also, the study of hazards or other problems that might be associated with the shipping of the UNH product will be completed. Further studies of the processing of low enrichment fuels will not be planned, except as necessary to support specific Hanford reactor or AEC study requests.

Because of its applicability to the case of Zircaloy-clad fuels, in general, over a variety of enrichments, the development of the fuel crush-leach concept will be continued. Crushing concepts will be compared and the most promising technique will be selected. Design concepts for remote application will be investigated. Potential problems and hazards will be evaluated and a system demonstration with irradiated fuel will be planned.

b. Fiscal Year 1970, and Continuing

Development of the fuel crushing concept will be continued, with the objective of completing the demonstration and design studies required for a prototype system during FY-1970.

2. Highly Enriched Fuels

a. Fiscal Year 1969

Engineering studies of separations processing concepts and capabilities for U-Al and U-Zr alloys will be documented. As additional laboratory data and reactor data become available, concepts will be modified accordingly. A relatively low level of effort is expected, pending the completion of more definitive reactor studies and the definition of fuel design concepts.

Laboratory programs will be restricted to completion of the dissolution, solvent-extraction, and waste studies that were commenced during FY-1968. These studies will be of a preliminary nature, pending the definition of fuel design concepts by reactor personnel.

b. Fiscal Year 1970, and Continuing

On the basis of current concepts of an integrated Hanford schedule, heavy emphasis on chemical processing of highly enriched fuels is expected during FY-1970. Dissolution flowsheets will be defined and demonstrated with fully irradiated fuels as characterized by FY-1969 reactor studies. Solvent extraction reprocessing flowsheets will be defined and demonstrated. Waste management processing concepts will also be demonstrated.

With the development of reactor fuel design concepts, studies of cost-capability relationships will be accelerated. On-site recycle concepts will be studied in more detail. Equipment development requirements will be defined on the basis of the results of these studies.

Results of engineering studies and development programs will be continually responsive to adjustments in reactor fuel concepts. As definition of variables becomes more firm, preliminary facilities scoping will be started. Process and facilities concepts will also be dependent on the results of associated criticality studies. It is expected that by the beginning of FY-1971, process definition will be sufficiently firm to permit the completion of facilities scoping.

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MILESTONES

Schedules of activities for this mission are shown in Figures 1 and 2. Program milestones are as follows:

Complete preliminary studies of processing concepts and capabilities for U-Al and U-Zr alloy highly enriched fuels.	September 1968
Update cost-capability study for low enrichment fuels and publish results.	October 1968
Firm flowsheets available for processing N-Reactor coproduct.	January 1969
Firm flowsheets available for processing low-pressure reactor coproduct fuels.	January 1970
Fuel crush-leach process defined.	June 1970
Complete scoping of preliminary facilities for processing highly enriched fuels.	June 1971

REPORTS ISSUED

1. ARH-142, "Criticality Analysis of the Process of the Coproduct Demonstration 210 Metal Fuel Elements in the Purex Plant", J. B. Kendall and H. C. Rathvon, October 30, 1967 (SECRET).
2. ARH-106, "Processing of 210 Metal Fuel from the N-Reactor Coproduct Demonstration", J. M. Austin and G. A. Nicholson, October 18, 1967, (SECRET)
3. ARH-147, "Properties of Potential Uranium Precipitates as Related to Chemical Processing", C. A. Colvin, October 30, 1967 (UNCLASSIFIED).
4. ARH-407, "Safe U-235 Enrichments for the Purex Facility", G. R. Kiel and K. R. Ridgway, March 1, 1968 (UNCLASSIFIED).
5. ISO-1038, "Press Crushing of Zirconium-Clad Fuel Elements as a Mechanical Decladding Technique", R. M. Smithers, August 14, 1967 (UNCLASSIFIED).
6. ARH-31, "An Investigation of the Solid Liquid Phase Transition of 100% UNH", R. R. Jackson and J. V. Panesko, September 19, 1967 (UNCLASSIFIED).

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MANPOWER AND BUDGET SUMMARY

	<u>Fiscal Year</u>	<u>1969</u>	<u>1970</u>	<u>1971</u>	<u>1972</u>	<u>1973</u>	<u>5-Year Total</u>
<u>Dollars (in thousands)</u>							
ARHCO		63.3	182.4				
Battelle-Northwest		<u>36.7</u>	<u>217.6</u>				
TOTAL		100.0	400	520	530	480	2030
Equipment		10	50				

Man-Years

ARHCO	2.2	4.3
Battelle-Northwest	<u>1.0</u>	<u>5.9</u>
TOTAL	3.2	10.2

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FY 68	FY 69	FY 70	FY 71	FY 72	FY 73
INTERIM COST-CAPABILITY 1.0-2.10% SIMPLE SOL. POISON TECH.	STUDY				
ADVANCE COST-CAPABILITY 1.0->2.10% COMPLEX SOL. POISON TECH.	STUDY		UPDATE WITH ALT. H.E..	CONTINUE AS NECESSARY	
UNH SHIPPING STUDIES	CONCEPTS SELECTED	PROTOTYPE DESIGN	CONSTRUCT.	ACCIDENT ANALYSIS	LICENSE APPLIC.
ALT. HEAD END (MECH. CRUSH. PROCESS)	ENGINEERING STUDIES	DESIGN, FAB TEST PROTOTYPE	DES. DEV.	DECLASSIFIED	
HEAD END PROGRAMMER	EVALUATE 1st. UNIT	STUDY ALT. CONCEPTS	PROTOTYPE ALTERNATE CONCEPT		
NPR COPRODUCT SUPPORT	COPROD. PROCESSING				CONTINUE ENGR. SUPPORT
	EVAL. COPROD. CAMPAIGNS				
LPR COPRODUCT SUPPORT		ENGR. STUDIES	CRITICAL ANAL.	PROCESS DEMO.	CONTINUE ENGR. SUPPORT

PROGRAM SCHEDULE - LOW ENRICHMENT FUEL PROCESSING (1.0 - >2.1% U-235)

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PROGRAM SCHEDULE - HIGHLY ENRICHED FUEL PROCESSING

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