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

OTHER ISOTOPES

**RICHLAND OPERATIONS OFFICE
ATLANTIC RICHFIELD HANFORD COMPANY
BATTELLE-NORTHWEST
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O T H E R I S O T O P E S P R O G R A M

June 30, 1968

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MISSION 5
OTHER ISOTOPES

1. INTRODUCTION

The Hanford production complex is potentially capable of producing and processing a wide variety of isotopes. Some of these have been identified as specific R&D missions (i.e., plutonium-238, transplutonium elements and tritium), and the technical programs supporting these efforts are described elsewhere in this series of Five-Year Program documents. Capability to produce other isotopes has been demonstrated in the successful production of "clean" uranium-233, in the recovery of large quantities of selected fission products, and in the development of process technology for polonium-210. Active R&D programs are underway to improve the technical bases for producing these isotopes, and for demonstrating the capability of producing others.

Formerly, several R&D missions were identified and conducted to define the technical factors for initiating production of these isotopes. Either the efforts on these missions have declined because of decreased emphasis (polonium-210) or because of the advanced status of the technology (uranium-233), or the efforts have generally been modest (cobalt-60). These several efforts are now combined into one R&D mission, Other Isotopes, and they are described in the following pages of this document.

2. SCOPE AND OBJECTIVES

The purpose of this program is to develop the technology for producing and recovering radioisotopes (not a part of another R&D mission) such that a production proposal can be made with confidence. Emphasis will be placed on definition of production capability, on minimum cost, and on characterization of the radiation product.

Activities performed under sponsorship of this mission include the following:

Uranium-233 - Continue development of technology to support sustained low-cost production of uranium-233 at Hanford. Evaluate thoria target fabrication, irradiation and separation processes and identify methods for increasing production and reducing costs.

Polonium-210 - Evaluate reduced cost concepts and establish the technical bases for irradiating bismuth, separating the polonium product, and recovering the bismuth for recycle.

Cobalt-60 - Demonstrate Hanford production capability and evaluate production and cost data for cobalt-60 in forms and specific activities not currently commercially available.

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Thulium-170 - Demonstrate the conversion of natural thulium to thulium-170 in the Hanford reactors and provide physics and conversion data for determining production capability and cost.

Special Fission Products - Develop the technology to characterize the products and demonstrate the recovery of fission products for which uses may or could be developed. Fission products of current interest include xenon, krypton-85, antimony-125, rhodium, palladium, technetium and ruthenium-106.

Isotope Separation - Develop methods which may be applied to the separation of isotopes, enhancing the capability to produce new radioisotopes, such as thulium-171 and thallium-204.

Encapsulation Technology - Develop and demonstrate techniques for encapsulating a variety of radioisotopes as energy sources. The effort is directed towards a flexible technology which could be applicable principally to alpha sources (plutonium-238, polonium-210, curium-244) but also versatile enough to have application to beta sources requiring little or no shielding (promethium-147, thulium-171 and thallium-204). These encapsulation studies were consolidated in this mission, although the technology for producing some of the isotopes are parts of other missions. With decreasing fund availability, encapsulation technology is now limited to studies supporting waste management in Mission 9.

3. INCENTIVES

Interest is increasing in the application of the energy of radioisotope decay as heat and power sources for space missions, land and sea stations and submersible vessels, and as radiation sources for food sterilization, chemical polymerization and plant growth modification. There is also the potential widespread use of suitable isotopes for biomedical applications, e.g., as energy sources for heart pacers (millions) or as implanted artificial human hearts (hundreds of thousands). In addition interest continues for the use of krypton-85 and tritium in special light sources. Production and availability of suitable radioisotopes for these applications are limited, either because of availability of the source material or because of the radiation characteristics of the isotope. For example, fission products produced are a direct function of the total nuclear reactor output; plutonium-238 is limited by the availability of neptunium-237; polonium-210 and thulium-170, while intense sources of energy, have short half-lives. Therefore, the development of advanced technologies to define the production capability of and costs for radioisotopes of usable energy is a need of the Commission to establish firm bases for developing marketable uses of the isotopes.

The incentives for the uranium-233 program are unique to this mission. Although produced by reactor irradiation of thorium-232, its value lies in

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its property as a fissile material, rather than as an energy producer. (As indicated above, it is included in this mission since the total R&D effort to support its production is small). As the power reactor industry develops, a large market may potentially exist for tonnage quantities of uranium-233 to provide the basic technology for using and recycling uranium-233 in reactor applications using the thorium cycle. Since much if not most of the uranium-233 will be used to supply fissile material for the development of the reactor technology, emphasis is placed on the production of "clean" uranium-233; i.e., low in uranium-232 content. However, since a mature technology employing uranium-233 will use recycled uranium-233 from long exposure of thorium in a neutronic environment, the program is not limited to "clean" uranium-233 production, but includes the development of advanced technology for optimum production of uranium-233 at minimum costs.

Although currently of limited extent, there exists a continuing interest in the use of uranium-233 as a raw material for nuclear weapons.

For the radioisotopes which are the subject matter of this R&D mission it is to the balanced, long-range interest of the Atomic Energy Commission to define the production capability and product characteristics of reactor products produced in its production complex not only at Hanford, but also at Savannah River. The program described here is geared to providing that knowledge for Hanford. Definition and demonstration of production capability and related costs (both capital and operating) will provide the Commission and other government agencies a basis for planning for new production programs, consistent with and compatible to market and/or use development. Development of the advanced technology for these isotopes' production will also provide the Commission with a firm basis for judging and evaluating proposals for producing these isotopes in facilities external to the Commission's production complex as, for instance, in power reactors.

The program, as scoped in this document, attempts to be responsive to the development of technology to the extent possible as limited by fund availability prior to the need to implement a production program. Changes in emphasis of the various activities are anticipated to change as information becomes available to provide increased definition of product goals.

4. URANIUM-233

a. Purpose of Effort

A modest, continuing effort is supporting the development of technology for sustained production of uranium-233 at Hanford, with emphasis on fringe loadings for biological shield protection and on recycle of thorium, and increase fuel dissolution rates.

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b. Progress During Report Period

Optimization of the thoria target fabrication process was continued through the year during routine fabrication of thoria targets in support of fringe reactor loading for biological shield protection. About 33 tons of thoria were charged in fringe loading of the reactors with no failures being experienced during the period.

Test quantities of thoria were produced in the deactivated uranium calcination pots at Hanford. A portion was processed to a gel, and the powders used for target fabrication and irradiation studies. Compaction studies included tests in which a portion of the thoria was simply calcined at 1200 C and vibratory compacted into aluminum cans to a density of 5.6 g/cc. Tests and development of pelletizing techniques yielded a process by which another portion of the light thoria was mixed with water, pressed, and sintered at 1200 C to produce wafers of 8.2 g/cc density; the pellets met the standard dissolution specifications. The moisture and gas content of both the wafers and the calcined powder were comparable with thoria previously used in production campaigns. One column of thoria targets containing elements with vibratory compacted powder and elements with thoria wafers was charged in C Reactor, the irradiation of which is continuing at year end.

A sustained thoria irradiation program was initiated late in the year to provide high quality uranium-233 product in support of light water breeder demonstration core. Full core loadings of thoria in the K Reactors extending into the first quarter of FY-1970 are being scheduled to permit delivery of uranium-233 to begin during the fourth quarter of FY-1970. The full uranium-233 program involves a total production campaign of 430 Kg of uranium-233 product to meet the quality-time requirements of 360 Kg delivered by the first quarter of FY-1971.

c. Evaluation of Effort

A well defined technology for thoria target fabrication and the limited demand for uranium-233 until late in the year have supported only a low level of development effort. Refinements to the target fabrication process were realized through routine fabrication and fringe loading of thoria targets. Present technology is judged adequate to meet the requirements of the current sustained reactor core irradiation program. Significant advancements were made in establishing a technical capability for conducting all the steps required to recycle thoria on site. The potential savings obtainable by on-site recycle of thoria warrant a small continuing development effort to define the pertinent costs and capabilities.

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d. Budget Period Plans

Routine fabrication of thoria targets for fringe loading as well as the current larger-scale uranium-233 production program will provide continued optimization and refinement of uranium-233 production. Evaluation of the internal recycle thoria processing will be complete with post-irradiation examination of the test elements now being irradiated. Engineering studies of potential applications will be completed.

5. POLONIUM-210

a. Purpose of Effort

The purpose of the polonium-210 studies is, first, to maintain a small continuing effort for updating estimates of capital funding required and of operating cost for producing polonium-210 in amounts greater than about three kilograms per year (at launch), refining past studies by application of emerging technology; and second, to explore alternative approaches to the production of polonium-210 which have potential in offering substantial reductions in both capital and operating costs.

b. Progress During Report Period

PCTR experiments and measurements were completed, data from which are being evaluated for use for physics lattice calculations. Work was initiated on developing the physics characteristics of an enlarged channel lattice to permit an evaluation of operating characteristics and safety analysis of reactor operation.

Laboratory work on development of flowsheets for solvent extraction purification of irradiated bismuth following recovery of polonium was completed. The most promising solvent extraction process studied involves extraction of bismuth from a nitric acid - aluminum nitrate feed with 30 volume percent TBP in a normal paraffin diluent. Good separation of bismuth from virtually all impurities expected to be present appears feasible. The work was discontinued due to a shift of interest to a more promising fused sodium hydroxide extraction process for recovering polonium from irradiated bismuth.

Process feasibility of the fused sodium hydroxide extraction process has been demonstrated in laboratory-scale experiments. Transfer of 99 percent of the polonium-210 to the caustic phase can be accomplished.

The laboratory work involved contacting 2-16 grams of irradiated bismuth (3 to 90 MCi polonium-210/gram bismuth) with sodium hydroxide in a graphite boat at 400 to 600 C. Agitation of the melt in a horizontal plane was provided by a rocking furnace (48 cycles/min through a two-inch amplitude). Contact times varied from 30 to 60 minutes.

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At the end of the contact period the molten phases were allowed to cool, and the sodium hydroxide phase was dissolved in hydroxyacetic acid to separate it from the bismuth.

Ratios of bismuth to sodium hydroxide evaluated ranged from 0.25 to 8.0. At bismuth to sodium hydroxide ratios of 4.0 or less transfer levels were 99 percent or greater, falling to 96 percent at a ratio of 8.0. Equipment designed for better agitation would probably extend the 99 percent transfer level to the higher bismuth to sodium hydroxide ratios.

It was found necessary to blanket the melt with an inert gas during contacting. Both argon and nitrogen were found adequate for this purpose. If air is allowed to diffuse into the inert gas blanket during contact, the polonium-210 transfer level falls to 30-40 percent, probably due to formation of bismuth oxide at the melt interface. However, nitrogen containing 1 percent air showed no deleterious effect. Technical grade sodium hydroxide (available at 8¢/lb) gave results equivalent to reagent grade sodium hydroxide.

A statistically designed experiment was performed to determine action and interaction of time, temperature, and bismuth to sodium hydroxide ratio. Contact time was shown to be a primary variable indicating a kinetically controlled process.

Other experiments demonstrated that polonium-210 was readily extracted into a dibutyl butyl phosphonate-normal paraffin hydrocarbon solvent from a nitric acid solution of the sodium hydroxide phase. Also, gamma energy analyses of bismuth metal before and after contact with sodium hydroxide showed 99+ percent of the antimony-124 and zinc-65 transferred to the sodium hydroxide phase. Essentially all the silver-110 remained in the bismuth phase.

c. Evaluation of Effort

The laboratory work has furnished convincing evidence of the simplicity and feasibility of the molten sodium hydroxide extraction process for removing polonium-210 from irradiated bismuth. This work also suggests strongly that plant-scale operation of the process on a batch basis would provide adequate polonium-210 separation and recovery. Such a scheme would avoid the operation of large-scale solvent extraction or vacuum distillation equipment.

d. Budget Period Plans

Calculations and evaluation of data from the PCTR experiments of sample bismuth elements under the Polonium-210 Program will be completed early in FY-1969. This evaluation will provide a basis upon which to develop physics characteristics for super-bore or enlarged channel lattice can be developed to establish characteristics and safety analysis of reactor operation. A large-scale demonstration

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of the sodium hydroxide extraction process utilizing 1-2 pounds of irradiated bismuth will be completed in the near future. Work will be discontinued pending identification of large-scale need for polonium-210.

6. COBALT-60

a. Purpose of Effort

Irradiations are to be performed to provide data for quantifying the basic reactor parameters for producing cobalt-60 in forms and specific activities not commercially available. Effort will be limited to minor process technology funding.

b. Progress During Report Period

One column of target elements containing BNL strips was discharged from C Reactor late in the year when the strips reached calculated activity levels of about 100 Ci/gm. The strips are undergoing calorimeter measurements. Irradiation of a second column of BNL strip targets is continuing in order to provide additional specimens at higher activity levels for calorimeter measurements.

Two cobalt-plated reactor process tubes and two process tubes fabricated from Haynes-25 alloy (nominal 51 weight percent cobalt) have been prepared for installation in C Reactor. Evaluation of this concept as a technique for preparing cobalt-60 sources is based on indications that there is a potential demand for tubular radiation sources. The process tubes under test are scheduled to remain in the reactors for periods from 9 months to 2 years. Details and schedules for irradiation and evaluation of the tubes are being prepared.

The multi-lattice design-analysis program DCODE has been utilized in evaluating the production capability of N Reactor to produce various isotopes. Initial emphasis has been on the production of Co-60.

Prototypical irradiations of Co-59 in the moderator cooling system in N Reactor have been initiated to confirm the conclusions of the analytical studies. Approximately 1050 gm of cobalt, in the form of BNL strips, were charged during the fiscal year.

c. Evaluation of Effort

Work has been directed toward demonstration of Hanford production capability and the compiling of production and cost data. Process tube irradiations are expected to provide data on basic reactor parameters as well as evaluations of modifications to the form or structure of the cobalt source.

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Preliminary calculations using DCODE indicate that with present N Reactor system 500,000 curies of Co-60 can be produced in the channels of the moderator cooling system per year at a cost of less than \$0.20 per curie. For the production of 30 Ci/gm material the irradiation time is approximately 150 days at an overall efficiency of 75%. For 50 Ci/gm material the irradiation period is 250 days. Production in this mode is limited by the heat removal capacity of the moderator cooling system.

Testing of the thermal hydraulic characteristics of the cobalt targets and analysis of the operational parameters associated with the loading have shown the production of Co-60 in this manner to be compatible with the basic operational modes of the reactor.

Estimates of the production capability using primary system channels indicate that between 70 and 100 MCi of cobalt-60 can be produced per year, depending upon the operating mode of the reactor. The specific activity would be in the range 90 to 110 Ci/gm.

d. Budget Period Plans

Preparations for irradiation and evaluation of cobalt-plated or impregnated reactor process tubes will be completed. Confirmation of market interest and availability of funds will establish the irradiation schedule of this material. Calorimetric measurements of the BNL strips now undergoing irradiation will be completed.

The material charged in N Reactor during Fy-1968 will be discharged, and the specific activity will be determined. Irradiation of Co-59 initiated in Fy-1968 to confirm analytical studies, will be brought to an orderly conclusion. The production data and analytical studies will be evaluated and final reports will be written. The completion of these final studies will provide N Reactor with an essentially on-the-shelf capability for the production of Co-60 in the low to intermediate specific activity range and will provide a firm basis upon which technology could be developed in support of production commitments not readily satisfied by commercial facilities.

7. THULIUM-170

a. Purpose of Effort

Irradiations are performed to establish physics and conversion data as a basis for preparation of a production proposal for thulium-170 in the Hanford reactors.

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b. Progress During Report Period

The fabrication of thulium oxide microspheres was studied and demonstrated, during which work, test quantities of microspheres were prepared. To evaluate the properties and behavior of the microspheres, ten stainless steel capsules were prepared, each containing three grams of the thulium oxide spheres. These capsules are now undergoing irradiation in a test facility in K Reactor. Post irradiation evaluation will include examination of the microspheres and calorimeter measurements.

Calorimeter measurements were made on thulium oxide capsules which were provided by BNW for irradiation in support of other programs and studies. Data are thus being compiled to provide a calorimetric assay of thulium-170 sources.

Prototypical irradiations of thulium-170 in the moderator cooling system were initiated to confirm conclusions of analytical studies. Approximately 175 grams of thulium in the form of Tm_2O_3 compacts were charged to N Reactor during the fiscal year.

c. Evaluation of Effort

Current irradiations and evaluations are intended to confirm activity levels and exposure time relationships to permit definition of a production mode as well as calorimetric assay data of thulium-170 sources.

d. Budget Period Plans

Post-irradiation examination and subsequent calorimetric measurements of thulium oxide microspheres will be completed and included in summary reports of the work.

8. SPECIAL FISSION PRODUCTS

a. Purpose of Effort

Among the fission products are several which are of special interest and of potential value, and whose recovery and purification are not part of another program. Development of the technology for the recovery of strontium-90, cesium-137, promethium-147 and cerium-144 is a part of the Waste Management Program (Mission 9). In addition, investigation of recovery and separation methods for technetium, rhodium and palladium is being supported by other funding. The purpose of this part of the Other Isotopes R&D Mission is to explore the technical and economic factors needed to make authoritative judgments on the recovery of other fission products.

b. Progress During Report Period

Development of technology for the recovery and purification of special

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fission products was primarily limited to rhodium, palladium and technetium under separate funding.

Two techniques were discovered for removing palladium from Purex supernatant wastes. The first technique utilizes methyl tricaprylyl ammonium chloride sorbed on a substrate which acts as ion exchange media to selectively remove rhodium and palladium from the high salt wastes. The second technique calls for the ammonium hydroxide elution of palladium from strongly-basic anion exchange resin subsequent to the nitric acid elution of rhodium. In fact the nitric acid elution apparently is a prerequisite to palladium elution with ammonium hydroxide.

Technetium is recovered together with rhodium from Purex supernatant wastes. Technetium and rhodium are eluted together from anion exchange resin with a decontamination factor for radioactive impurities of about 100. Technetium is separated from rhodium in the next process step, yielding a technetium stream free from major amounts of impurities.

c. Evaluation of Effort

Efforts pertinent to this sub-mission were funded almost entirely from other sources. The chemistry of rhodium in Purex wastes was elucidated, and a feasible process has been identified for the recovery of rhodium, palladium and technetium from the stored wastes at Hanford.

d. Budget Period Plans

Principal effort will be applied to characterize the radiochemical properties of rhodium, palladium and technetium under this mission. Limited process development studies will be performed to scope out processes for recovering specific fission products that are determined to have market potential.

9. ISOTOPE SEPARATION

a. Purpose of Effort

Among all the radioisotopes there is an impressive lack of large-scale, long-range availability of those characterized by low shielding requirements and relatively low biological toxicity. The future supply of such desirable isotopes could be increased to a great extent by the discovery and development of practical techniques of isotope separation. Conventional techniques such as gaseous diffusion, electromagnetic and ultra-centrifugation, appear to be clearly outside the economic range for the materials and magnitudes that can be reasonably projected. It is the purpose of this part of the mission to conduct scouting tests in the laboratory on prospective unconventional isotope separation techniques which may have large-scale and

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economical applicability to the isotopes of interest; e.g., neodymium-146 for promethium-147 production, erbium-170 for thulium-171 production, and the recovery of thallium-204 from irradiated natural thallium.

b. Progress During Report Period

Work was continued on the separation of isotopes by electromigration. In experiments with lithium-7/lithium-6, the lithium-7/lithium-6 isotope ratio was changed by 14.1 percent (from the starting ratio of 12.70 to 10.91) by migrating lithium through silica gel along a membrane for 12 hours using a direct current of 60 ma.

Pulsed current migration experiments were continued in an effort to determine the effect one might expect from pulsed current migrations as compared to direct current migrations. The results of these experiments indicated the following:

1. A pulsed current was more effective for migrating ions than was an equivalent direct current.
2. A symmetrical wave produced a faster rate of migration than did an unsymmetrical wave.
3. The rate of migration was not a function of the pulse frequency.
4. The amount of isotope separation taking place appeared to be a function of both the rate and length of migration. (Consequently, a longer length of migration did not necessarily produce an increase in isotope separation.)

c. Evaluation of Effort

Separation of lithium and neodymium isotopes has been demonstrated both by electromigration along a membrane and by electromigration through silica gel. However, the actual amount of separation achieved in the case of neodymium was so small that it is of little practical interest.

In general, pulsed current produced better ionic migration through silica gel. An optimum pulse frequency for maximum migration and/or maximum isotope separation, however, was not found. The results did indicate that better isotope separation might be achieved using low frequency pulses (<10 kcps) as opposed to very high frequency pulses (>100 kcps). The increase in isotope separation achieved by using a pulsed current is attributed to the partial stripping of the ionic atmosphere from the central atom.

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d. Budget Period Plans

No further work is planned to separate isotopes by electromigration. Work in this sub-mission is suspended pending identification of another technique having sufficient potential to warrant investigation.

10. ENCAPSULATION TECHNOLOGYa. Purpose of Effort

This effort is intended to provide the technology required to support the design concept for a simple, flexible and versatile facility for encapsulating energy sources which require not more than modest shielding. Isotopes which could be encapsulated in such a facility include not only low energy beta-emitting isotopes such as promethium-147 and the alpha-emitting isotope, polonium-210, but also other alpha-emitting isotopes such as plutonium-238 and possibly curium-242 and -244. Development of technology supporting the production of the latter isotopes is the subject of Missions 3 and 4.

b. Progress During Report Period

Techniques and equipment were demonstrated for welding end caps on aluminum right circular cylinders containing simulated neptunium alloy target elements. A special welding process known as the PROGRAMMED INERT GAS MULTIPLE ELECTRODE (PIGME) process was shown to have promise for application in welding capsules in remote installation such as glove boxes and shielded cells.

c. Evaluation of Effort

The equipment and techniques required to automatically weld an aluminum jacketed neptunium aluminum alloy target element have been demonstrated. The weld quality is satisfactory.

The initial evaluation of the PIGME process is encouraging and should have distinct advantage for remote welding applications.

d. Budget Period Plans

Evaluation of the PIGME process and development of remote welding techniques for various metals will be discontinued because of a lack of funds. The encapsulation of cesium-137 and strontium-90 will be developed under Mission 9.

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12. MILESTONES

a. Uranium-233

Complete development of technology for recycle using PT funding.

April, 1970

b. Polonium-210

Evaluate PCTR data and issue report using PT funding.

January, 1969

Complete reactor safety analysis for enlarged reactor channels.

January, 1971

c. Cobalt-60

Evaluate production of tubular cobalt sources.

December, 1971

d. Thulium-170

Evaluate irradiated miscospheres.

January, 1971

Complete calorimetric assays.

April, 1971

Prepare production proposals.

June, 1971

e. Special Fission Products

Characterize radiochemical properties of Rh, Pd, Tc.

January, 1969

f. Encapsulation Technology

Develop prototype job-shop welder for use in remote applications.

April, 1972

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14. MANPOWER AND BUDGET SUMMARY

<u>Dollars (In Thousands)</u>	<u>1969</u>	<u>1970</u>	<u>1971</u>	<u>1972</u>	<u>1973</u>	<u>5 Year Total</u>
Douglas United Nuclear, Inc.						
Conventional Reactors						
N Reactor						
ARHCO	50	50				
Pacific Northwest Laboratory*						
Total	50	50	150	300	500	1050

Man Years

Douglas United Nuclear, Inc.		
Conventional Reactors		
N Reactor		
ARHCO	0.5	0.6
Pacific Northwest Laboratory	0.5	0.4
Total	1.0	1.0

Equipment (in thousands)

Douglas United Nuclear, Inc.		
Conventional Reactors	0	0
N Reactor		
ARHCO	0	0
Pacific Northwest Laboratory*	0	0
Total	0	0

* Pacific Northwest Laboratory funds included in operating contractor totals.

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FY 68	FY 69	FY 70	FY 71	FY 72	FY 73
<u>URANIUM-233</u> FRINGE LOADING LIGHT OXIDE POWDER CHEM. PROCESS'G DEDAUGHTERIZING	<div>EFFORT TO BE SUPPORTED BY PT. FUNDING</div>	<div>AS REQUIRED</div>			
<u>POLONIUM-210</u> ENLARGED CHANNELS CHEMICAL PROCESSING	<div> <div>EVALUATE PCTR DATA</div> <div>ISSUE REPORT</div> <div>USING PT FUNDING</div> </div>	<div>ENGINEERING FEASIBILITY STUDY MOLTEN NaOH EXTRACTION PROCESS</div>	<div>COMPLETE REACTOR SAFETY ANALYSIS</div> <div>EVALUATE PRODUCTION INCENTIVES</div> <div>DEVELOP PLANT SCALE PILOT PRODUCTION PROCESS</div>	<div>DEVELOP PRODUCTION PROCESS AS JUSTIFIED</div>	

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PROGRAM CHART - OTHER ISOTOPES

FY 68	FY 69	FY 70	FY 71	FY 72	FY 73
COBALT-60	<div> <div>EVALUATE DATA</div> <div>ISSUE REPORT</div> <div>USING PT FUNDING</div> </div> <div> <div>N REACTOR</div> <div>100 CI/GR PRODUCTION PROPOSALS</div> </div>	<div> <div>EVALUATE TUBULAR SOURCES</div> <div>ISSUE REPORT</div> <div>100 CI/GR CAPABILITY ESTABLISHED</div> </div> <div> <div>C REACTOR</div> </div>			
THULIUM-170	<div>IRRADIATE TARGETS</div>		<div> <div>EVALUATE IRRADIATED MICROSPHERES</div> <div>CALORI-METRIC MEASURE</div> <div>PRODUCTION PROPOSALS</div> </div> <div> <div>ISSUE REPORTS</div> </div>		
SPECIAL FISSION PRODUCTS	<div>CHARACTERIZE RADIOCHEM. PROP. of Rd, Pd, Tc, ETC.</div>	<div>PROCESS DEVEL. AS NEED IS IDENTIFIED</div>		<div>DEMONSTRATE CAPABILITY FOR ATTRACTIVE PRODUCTS</div>	
ISOTOPE SEP		<div>MARKET STUDIES TO IDENTIFY POTENTIAL USES & APPLICATIONS</div>		<div>LAB & PILOT STUDIES AS JUSTIFIED</div>	
ENCAPSULATION TECHNOLOGY			<div>EQUIP. & PROCESS ENGR. STUDIES</div>	<div>DEMONSTRATE CAPABILITY</div>	
			<div>PROTOTYPE JOB SHOP WELDER</div>		

PROGRAM SCHEDULE - OTHER ISOTOPES

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