

Appendix C

Medical and Industrial Isotope Target Fabrication and Processing Operations

This appendix describes the technologies used to fabricate targets, process irradiated targets, recycle unconverted materials, and ship the product medical and industrial isotopes.

C.1 TARGET FABRICATION AND PROCESSING FACILITIES

Target fabrication and processing facilities are described in Chapter 2.

C.2 DESCRIPTION OF THE TARGET FABRICATION PROCESS

The production of medical and industrial isotopes involves fabricating specially designed targets, irradiating the targets in the reactor core or the accelerator target caves to generate specific medical isotopes, and processing the targets to prepare medical isotopes for shipment to customers. At reactors, Long-Term Irradiation Vehicles would be used for irradiating materials to produce long-lived isotopes, and Rapid Radioisotope Retrieval Systems would be used for short-lived isotopes. The Long-Term Irradiation Vehicle would consist of a bundle of target pins installed inside a nozzle, duct, and handling socket assembly similar in appearance to a fuel assembly (**Figure C-1**). Depending on the isotopes to be produced, the pin bundle could contain moderator pins and neutron shield pins to provide the desired flux in specific core locations for isotope production. A design that would allow reuse of the Long-Term Irradiation Vehicle assembly nozzle, duct, and handling socket hardware would be considered during the design process in an effort to reduce costs and waste generation. **Figure C-2** is a picture of the cobalt-60 test vehicle prior to installation of the outer hexagonal duct. While cobalt-60 is not on the list of representative candidate medical isotopes, this assembly configuration is typical of the Long-Term Irradiation Vehicles irradiated in the Fast Flux Test Facility (FFTF).

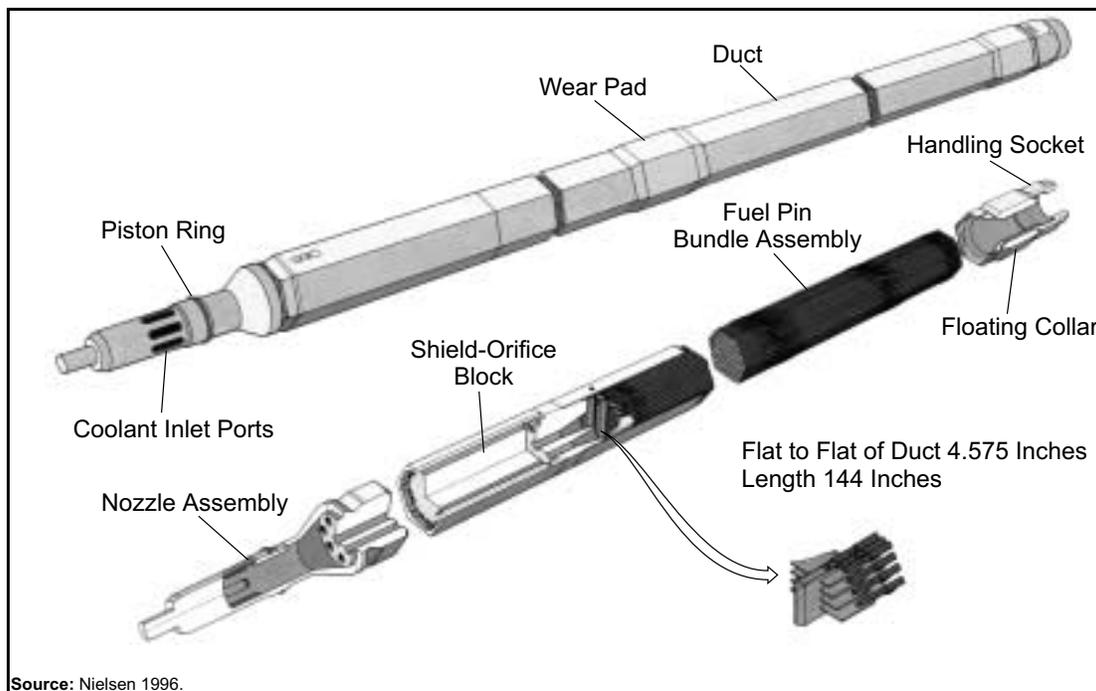


Figure C-1 An Example of an FFTF 12-Foot Fuel Assembly

The Rapid Radioisotope Retrieval System, which would be developed and used for the production of short-lived isotopes, would allow target materials to be inserted and withdrawn from the reactor core with the reactor operating at power. This system is described in Appendix D. Target material would be contained in individual capsules or interconnected carriers; allowing insertion and retrieval from the reactor as a string of targets. In addition to irradiating solid targets in the rapid retrieval system carrier strings, gas targets also could be irradiated to produce short-lived isotopes, as discussed in Appendix D. The system configuration presented for FFTF in Appendix D is typical of Rapid Radioisotope Retrieval System installations in other reactors engaged in medical and industrial isotope production.

The remainder of this section focuses on the first step in the isotope production process: the fabrication of the medical and industrial isotope targets.

Table C-1 presents a representative listing of 30 medical isotopes that are evaluated in this programmatic environmental impact statement. These isotopes were selected as being representative considering the current and future domestic and international demand for individual isotopes and current and future production capabilities for these products. The specific mix of isotopes produced by any irradiation facility will vary depending on the operating characteristics and authorized operating limits of the facility and the changing demands for specific isotopes during the production mission.

C.2.1 Target Materials

Each type of medical isotope would be produced using a target enriched in the appropriate isotope target material for neutron irradiation. With the exception of the radium-226 target and those using recycled materials, all of the targets use nonradioactive materials. Table C-1 presents the type and form of the target material to be used to produce each medical isotope. In cases where the target material and the product isotope are the same element, the target and the product cannot be chemically separated and the target material is shipped with the product. In cases where the target material and the isotope product are different elements, the target and isotope product can be chemically separated. After irradiation of the original target material (which in some cases may not be completely pure), there may be radioactive impurities that remain with the target material after the removal of the medical isotope product. Because of these impurities, the reuse of the material can, in some cases, create targets that are radioactive. As a result, the fabrication of targets using recycled target materials would require special handling. Shielding and special handling also would be required for the radium-226 target material (used to produce actinium-227, thorium-238, and thorium-239); it would be the only target material that would be radioactive before irradiation.



Source: Nielsen 1999.

Figure C-2 A Cobalt-60 Test Assembly

Table C-1 Representative Candidate Medical Isotopes

Product Isotope	Half-Life	Primary Target Isotope ^a	Target Vehicle	Irradiation Time (days)	Primary Target Isotope Mass (grams)	Product Isotope Specific Activity (curies per gram)	Product Isotope Activity Produced (curies)	Product Isotope Activity Delivered (curies per assembly)
Actinium-227 ^b	2.18 years	Radium-226 ^c	LTIV-H	100	1.5×10 ¹	7.2×10 ¹	3.4×10 ¹	3.4×10 ¹
Gold-198	2.69 days	Gold-197 ^d	R3-H	10	3.7×10 ⁻¹	2.1×10 ²	1.7×10 ²	7.9×10 ¹
Cadmium-109	462.0 days	Cadium-108 ^d	LTIV-H	100	6.6×10 ¹	9.9×10 ⁰	6.6×10 ²	6.5×10 ²
Copper-64	12.7 hours	Zinc-64 ^d	R3	10	2.4×10 ³	3.8×10 ⁶	4.8×10 ³	9.6×10 ¹
Copper-67	2.58 days	Zinc-67 ^d	R3	10	7.8×10 ¹	7.5×10 ⁵	8.2×10 ⁰	3.7×10 ⁰
Gadolinium-153	242 days	Natural europium	LTIV-H	100	1.2×10 ³	4.4×10 ¹	1.1×10 ³	1.1×10 ³
Holmium-166	1.12 days	Holmium-165	R3-H	10	4.3×10 ⁻¹	4.0×10 ¹	1.1×10 ²	1.7×10 ¹
Iodine-125	60.1 days	Xenon-124	Gas line	100	1.6×10 ⁰	1.7×10 ⁴	2.6×10 ³	2.4×10 ³
Iodine-131	8.04 days	Tellurium-130 ^d	R3-H	25	1.6×10 ²	1.2×10 ⁵	3.4×10 ²	2.6×10 ²
Iridium-192	73.8 days	Iridium-191	LTIV	100	5.7×10 ¹	6.1×10 ¹	3.7×10 ³	3.5×10 ³
Lutetium-177	6.68 days	Lutetium-176	R3-H	25	2.0×10 ⁻³	2.0×10 ²	5.5×10 ⁻¹	4.0×10 ⁻¹
Molybdenum-99	2.75 days	Molybdenum-98 ^d	R3-H	10	2.7×10 ¹	3.7×10 ¹	2.1×10 ³	1.0×10 ³
Osmium-194	6.0 years	Osmium-192 ^d	LTIV	100	1.6×10 ⁴	1.4×10 ⁻⁴	2.2×10 ⁰	2.2×10 ⁰
Phosphorus-32	14.3 days	Sulfur-32	R3	25	1.0×10 ¹	2.8×10 ⁵	4.1×10 ¹	3.5×10 ¹
Phosphorus-33	25.3 days	Sulfur-33	LTIV	100	1.3×10 ⁰	1.5×10 ⁵	7.8×10 ¹	6.5×10 ¹
Palladium-103	17.0 days	Palladium-102 ^d	R3-H	25	2.0×10 ¹	6.0×10 ¹	1.4×10 ³	1.2×10 ³
Platinum-195m	4.02 days	Platinum-195 ^d	R3-H	25	2.2×10 ¹	5.5×10 ⁰	2.0×10 ²	1.2×10 ²
Rhenium-186	3.78 days	Rhenium-185 ^d	R3-H	25	6.4×10 ⁰	4.7×10 ²	5.2×10 ³	3.0×10 ³
Scadium-47	3.35 days	Titanium-47 ^d	R3	10	4.3×10 ²	8.2×10 ⁵	3.3×10 ¹	1.8×10 ¹
Selenium-75	120 days	Selenium-74 ^d	LTIV-H	100	3.2×10 ⁻²	5.3×10 ²	1.8×10 ¹	1.7×10 ¹
Samarium-145	340 days	Samarium-144 ^d	LTIV-H	100	2.3×10 ⁰	4.7×10 ⁰	1.1×10 ¹	1.1×10 ¹
Samarium,-153	1.93 days	Samarium-152 ^d	R3-H	10	3.4×10 ⁻²	1.0×10 ³	1.0×10 ²	3.5×10 ¹
Tin-117m	13.6 days	Tin-116 ^d	R3-H	25	1.4×10 ⁰	3.1×10 ¹	5.1×10 ¹	4.4×10 ¹
Strontium-85	64.8 days	Strontium-84 ^d	LTIV-H	100	3.5×10 ¹	5.7×10 ¹	2.2×10 ³	2.0×10 ³
Strontium-89	50.5 days	Strontium-88 ^d	LTIV-H	100	1.6×10 ²	8.8×10 ⁻¹	1.5×10 ²	1.4×10 ²
Thorium-228 ^e	1.91 years	Radium-226 ^c	LTIV-H	100	6.3×10 ¹	8.1×10 ²	1.4×10 ²	1.4×10 ²
Thorium-229 ^f	7,300 years	Radium-226 ^c	LTIV-H	100	2.0×10 ³	1.1×10 ⁻²	2.7×10 ⁻²	2.7×10 ⁻²
Tungsten-188	69.4 days	Tungsten-186	LTIV-H	100	1.0×10 ⁴	5.5×10 ⁻¹	5.9×10 ³	5.5×10 ³
Xenon-127	36.4 days	Xenon-126 ^d	LTIV	100	2.2×10 ⁻¹	3.0×10 ²	7.4×10 ¹	6.5×10 ¹
Yttrium-91	58.5 days	Zirconium-91 ^d	LTIV	100	5.3×10 ²	2.5×10 ⁴	1.8×10 ¹	1.7×10 ¹

a. One hundred percent enriched.

b. Actinium-227 will decay to radium-223, which is the isotope that has medical applications.

c. Target form would be either basic element, carbonate, or chloride.

d. Target form would either be the basic metallic element or a metallic oxide or other compound, dependent on availability and on engineering considerations such as material melting point, degradation characteristics and processing methods.

e. Thorium-228 will decay to radium-224, which is the isotope that has medical applications.

f. Thorium-229 will decay to bismuth-213, which is the isotope that has medical applications.

Key: LTIV, Long-Term Irradiation Vehicle; LTIV-H, Long-Term Irradiation Vehicle–hydrided; R3, Rapid Radioisotope Retrieval System; R3-H, Rapid Radioisotope Retrieval System–hydrided.

Source: Nielsen 1999.

For the nonradioactive targets, the target material typically would be acquired from Oak Ridge National Laboratory (ORNL), where enrichment processes are conducted to produce target material of sufficient purity to support the generation of medical isotopes. The target form may be a metal, metallic oxide, or other chemical compound suitable for irradiation, depending on engineering considerations such as material heat transfer characteristics, melting points, and metallurgical properties. The nonradioactive target material would be transported by truck from ORNL to Hanford or to the U.S. Department of Energy (DOE) site having the new DOE low-energy accelerator or new research reactor and support facilities. The transportation mode, the shipment origin and destination, and transportation requirements for medical isotopes to be produced at Hanford are illustrated in **Figure C-3**. With the exception of the transportation of radium-226 to the medical isotope processing facility for target fabrication, the same transportation scenario would be applicable to each isotope.

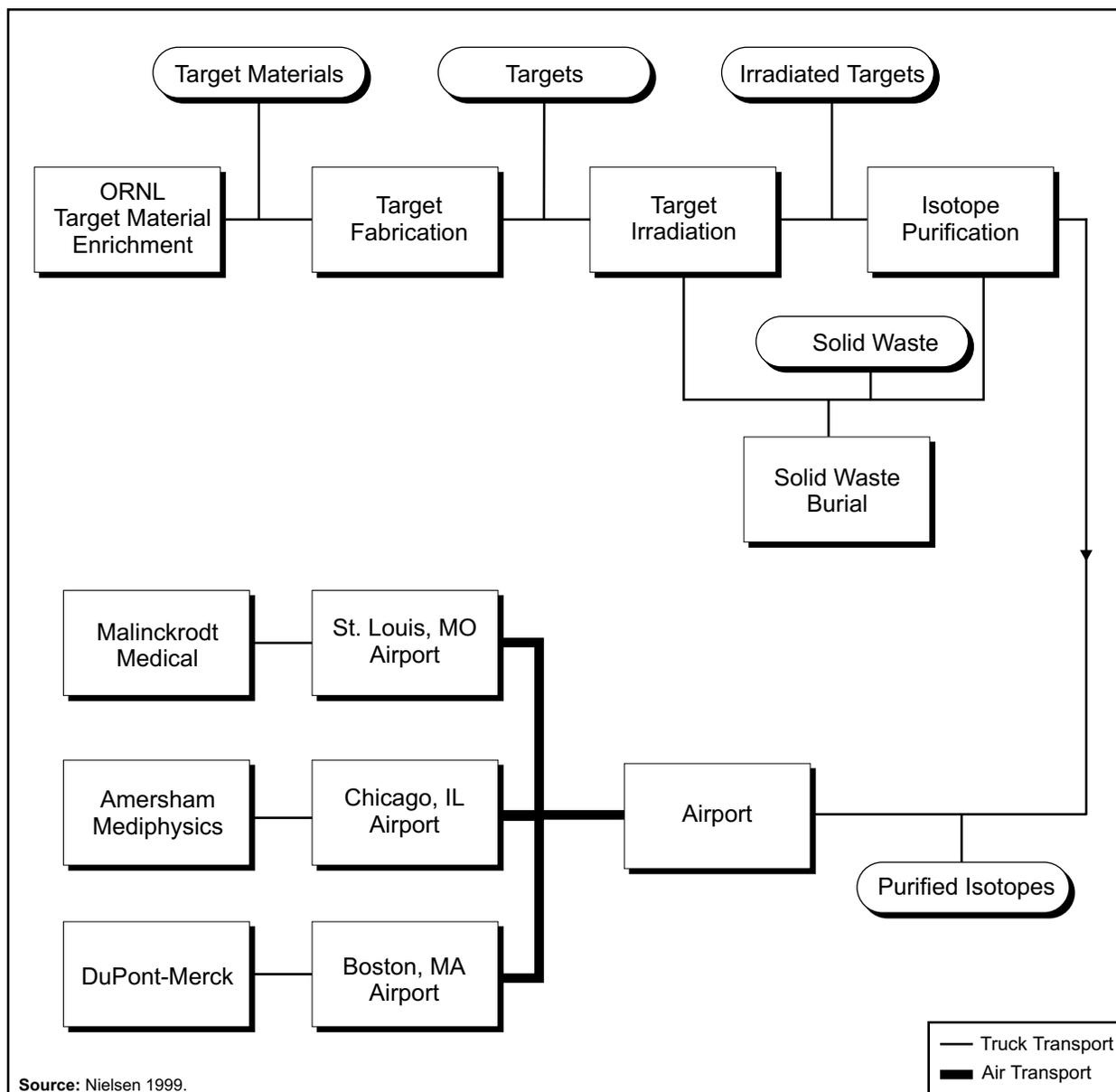


Figure C-3 Isotope Transportation

The radioactive target material, radium-226, would not be supplied by ORNL. However, radium-226 could be supplied by a variety of sources. Once materials for the targets arrive on the site, the materials would be stored in the medical isotope processing facility until needed for fabrication into medical isotope targets. Nonradioactive target materials do not have special storage requirements and most likely would be stored at the fabrication site. Special storage conditions are required because radium-226 generates radon gas. Recycled target materials would be stored in the processing facility.

For most types of targets, a cladding or encapsulation material would be needed to contain the target material during handling and irradiation. The Long-Term Irradiation Vehicle would be submerged in the reactor coolant, and it is anticipated that the target pins would be similar in construction and cladding to fuel pins. In this case, the cladding material must be fully compatible with the reactor coolant and should have low neutron-induced swelling characteristics.

For the Rapid Radioisotope Retrieval System, the targets would be inserted into and be irradiated in one or more fixed thimbles that extend into the reactor core. The individual targets, each composed of different target material, would be encapsulated or clad as necessary to (1) facilitate target handling, (2) contain the target materials and products during irradiation, (3) avoid interactions between the various target and structural materials at the irradiation temperature, and (4) maintain target purity. Dependent on the method finally selected for target insertion and retrieval (pneumatic or mechanical and cable system), the individual targets may be contained in cylindrical target carriers. These carriers would be interconnected to form a target string. To reduce the volume and weight of shielding required on the retrieval and irradiated target handling equipment, consideration would be given to using low-activation material for the target carriers. Studies also would be done to define the impacts on irradiation conversion efficiency, worker dose, processing efficiency, and waste streams of using higher-activation material for the target carriers or cladding (Nielsen 1999).

C.2.2 Target Fabrication Operations

Solid targets for either the Long-Term Irradiation Vehicles or the Rapid Radioisotope Retrieval System would be fabricated in gloveboxes using a series of mechanical and thermal processes. For the solid targets based on a powder, it is unknown at this time if the powder would be loose or if the powder would be pressed and sintered into pellets in some cases, depending on the material. If it is preferable to have pressed and sintered pellets, this option would require separate dies and boats to press and sinter each type of solid target material to reduce the risk of cross-contamination from other target materials. For solid metallic target material, depending on the purchased form, it may only be necessary to machine or cut the material to the required size.

If pellets are used, the first major step in their preparation would be powder conditioning and pressing. This would include weighing, blending, and pressing the powder and binder into slugs. The slugs would be granulated, blended with binder addition, and pressed into pellets. The pellets would be transferred to the sintering and debind station, weighed, and subjected to a series of thermal processes to debind and sinter the pellets. The sintered pellets would be subject to characterization to ensure that specifications were met.

Acceptable pellets would be transferred to the loading and welding station to be visually inspected before inclusion into a capsule or pin. For both powder or pellet target materials, capsules and pins would be cleaned before final closure. The capsules would be leak-tested and inspected before being cleared for use (Nielsen 1999). Normally, the required characteristics and physical configuration of a target for production of a specific isotope define if the powder should be loose in the target or pressed and sintered into pellets. In the event that there is a choice between powder or pellet, all other irradiation considerations and irradiated target processing considerations being equal, the loose powder target would normally be preferred. The target fabrication process for loose powder targets is much simpler, more cost effective, and generates less waste and

emissions due to the fact that there is less material processing. In addition, less equipment would require general housekeeping and cleanup, test, calibration, and maintenance.

C.2.3 Nuclear Research and Development Targets

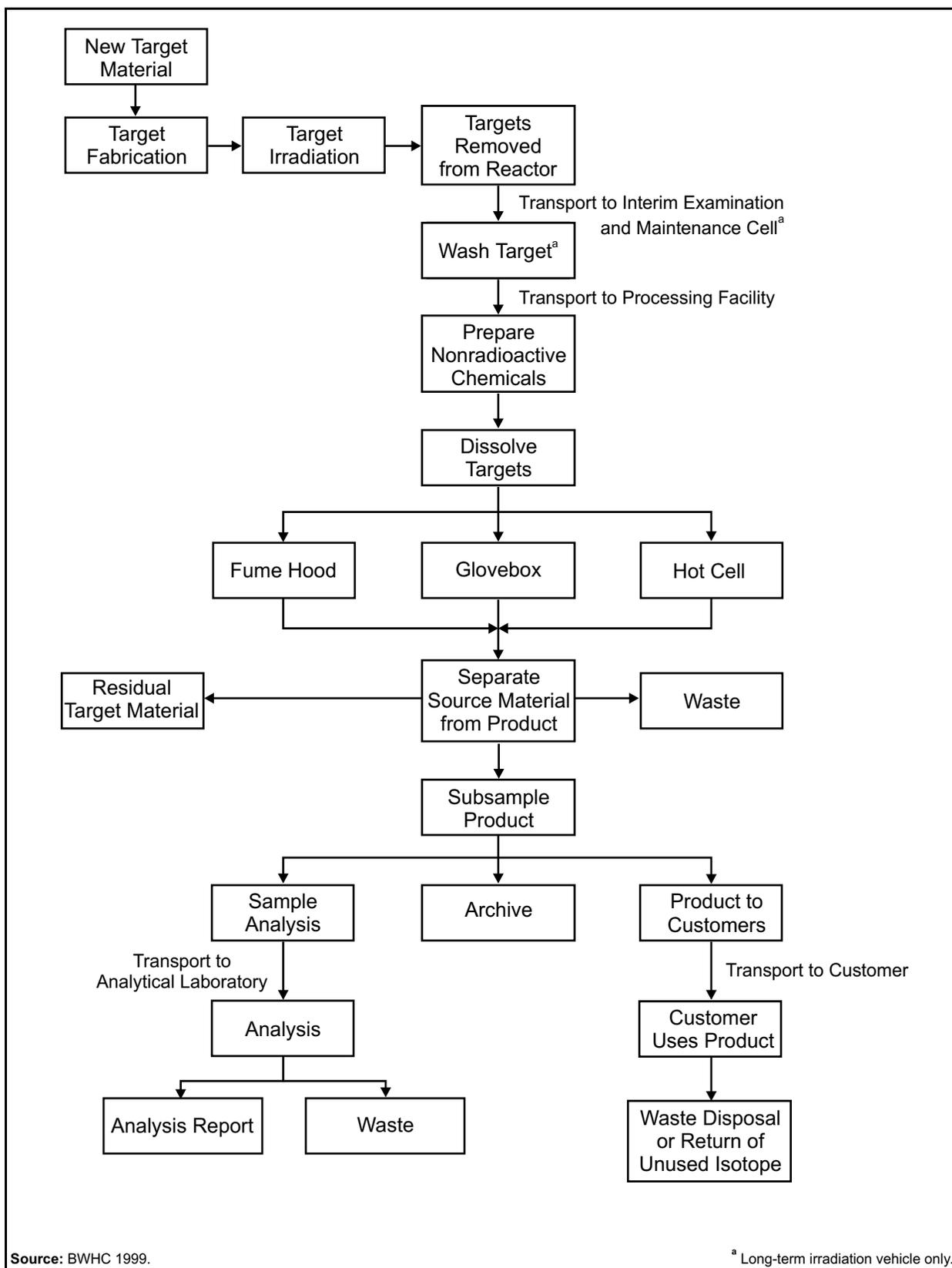
There is particular interest in materials testing associated with commercial nuclear power plant license renewals, cooperative international fusion energy research, space power technology, and transmutation of waste as a means to destroy long-lived isotopes from commercial spent nuclear fuel. Another area of interest is developing nuclear technologies that advance global nonproliferation. Target assemblies to be irradiated in support of these mission areas could be fuel or other materials configured similar to a standard driver fuel assembly (i.e., target material encased in sealed pins, and the pins placed in a ducted assembly). Material specimens also could be installed in an open test assembly position within a Material Open Test Assembly. Test target assemblies or materials (e.g., for a Material Open Test Assembly) could be fabricated on site or at the customer's facility and then transported to an accelerator or reactor for irradiation.

One of these proposed testing activities consists of fuel testing for the Accelerator Transmutation of Waste (ATW) program. A fuel development activity for this program could use a reactor for irradiation testing. The tests could be fabricated by the ATW program and transported to the reactor site for irradiation. Specific test compositions and irradiation parameters for these test fuel assemblies have not been defined. It is anticipated that initial tests would involve a few pins in an assembly, while later tests could involve entire assemblies. The target pins are described as containing a matrix of zirconium and transuranic waste with a composition of 25 percent transuranic waste and 75 percent zirconium by weight. The transuranic waste likely would be light water reactor discharge fuel at a typical burnup of 33,000 megawatt days per metric ton of uranium that is stripped of essentially all uranium and fission products. Another potential testing activity could be performed at FFTF for the ATW program is irradiation testing using lead-bismuth alloy as a coolant. Lead-bismuth alloy has been proposed as the target material for production of spallation neutrons in the ATW accelerator and as the coolant for the transuranic target assemblies undergoing irradiation at the ATW facility. FFTF has the capabilities for high-temperature irradiation closed-loop testing involving liquid metal coolant isolated from the reactor coolant system and test positions in the reactor that could be adapted for this test activity (Nielsen 1999).

C.3 DESCRIPTION OF THE POSTIRRADIATION TARGET PROCESSING

Processing of irradiated targets to recover medical- and industrial-grade isotopes can be broken down into distinct steps: receipt of irradiated targets into a chemical separation facility; chemical processing of the targets (using hot cells, shielded gloveboxes, and appropriate open-faced hoods); waste handling; analysis of the products; recycling of some of the target materials; and shipment of the isotope products to customers. A representation of this process for preparation of isotopes is shown in **Figure C-4**.

Each of the 30 representative medical isotope products evaluated for production is unique. Thirteen of the targets would produce an isotope of the same element and would not require separation. Five of the targets would produce the same element but would require some processing to remove impurities. Twelve target materials would produce different elements and would require chemical separation, both for separation of the target material and unwanted impurities (BWHC 1999).



Source: BWHC 1999.

^a Long-term irradiation vehicle only.

Figure C-4 Production and Radiochemical Processing Steps in the Preparation of Medical and Industrial Isotope Products

C.3.1 Same Target/Products Requiring No Separation

Thirteen of the same target element/products would not require extensive chemical separation (**Table C-2**). However, with experience, target purity and customer requirements might change, requiring additional separation and purification.

Table C-2 Products Requiring No Separation

Product	Curies per Target Campaign	Irradiation		Campaigns per Year
		Material	Time (days)	
Molybdenum-99	21,000	Molybdenum-98	10	25
Palladium-103	14,000	Palladium-102	25	10
Platinum-195m	200	Platinum-195	25	25
Rhenium-186	5,200	Rhenium-185	25	25
Selenium-75	18	Selenium-74	100	3
Samarium-153	100	Samarium-152	10	25
Tin-117m	51	Tin-116	25	10
Tungsten-188	59,000	Tungsten-186	100	3
Holmium-166	110	Holmium-165	10	25
Iridium-192	37,000	Iridium-193	100	3
Gold-198	170	Gold-197	10	25
Samarium-145	110	Samarium-144	100	3
Strontium-85	2,200	Strontium-84	100	3

Source: BWHC 1999.

A typical example of a product that would require no postirradiation separation is holmium-165/holmium-166 (**Figure C-5**). Medical isotope target carriers would be unloaded from the cask into A-Cell, where they would be separated and prepared for transport from A-Cell to isotope processing stations. The target material would be removed from the carrier and dissolved using nitric acid, hydrochloric acid, or a combination of acids. The dissolved material would be evaporated to near dryness to remove the acid. The resulting salt would be redissolved in dilute acid. The product solution would be analyzed for chemical and radionuclide purity, and aliquots of the analyzed product would be placed in appropriate containers and shipped to customers. Analytical techniques that might be used include (1) inductively coupled plasma/atomic emission spectroscopy, (2) gamma energy analysis, (3) alpha energy analysis, and (4) counting equipment.

All liquid waste would be neutralized and captured on a solid absorbent as solid waste. Solid compacted waste for each product (excluding cladding hardware) would range from 0.28 to 0.14 cubic meters (1 to 5 cubic feet) per year per product (BWHC 1999).

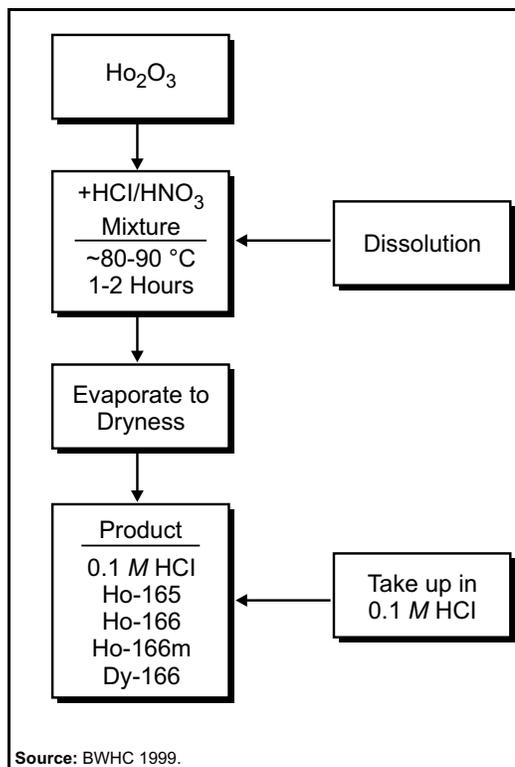


Figure C-5 Processing of Target Material Where the Isotope Product (Holmium-166) is Chemically Identical to the Target Isotope (Holmium-165)

C.3.2 Same Target/Product Requiring Separation

Approximately five target materials producing other isotopes of the same element would require further processing: cadmium-109, lutetium-177, osmium-194, strontium-89, and xenon-127. A typical example of a product requiring separation would be cadmium-109 from cadmium-108. The cadmium-109 produced from enriched cadmium-108 would contain small amounts of cobalt-60, zinc-65, and silver-110m, along with cadmium-109 and cadmium-115m. Medical isotope target carriers would be unloaded from the cask into a hot cell, where they would be prepared for transport from the hot cell to the appropriate isotope processing stations. The cadmium oxide would be removed from the target carrier, dissolved in hydrochloric acid, and diluted with water to 0.1 molar (*M*) hydrochloric acid. The impurities and cadmium would be adsorbed on a Dowex 50-X8 cation exchange column. The cadmium then would be selectively eluted with 0.2 *M* hydrochloric acid. The eluate would be analyzed using inductively coupled plasma/atomic emission spectroscopy and aliquots of the product solution placed in appropriate containers and shipped to the customers.

All liquid waste would be neutralized and captured on a solid absorbent as solid waste. Solid compacted waste (excluding cladding hardware for each same target/product) would be about 0.28 to 0.14 cubic meters (1 to 5 cubic feet) per year per product.

C.3.3 Processing Target Materials Containing New Elements

Twelve target materials would produce different product elements and each would require unique chemical separation. The processing steps would include separation and purification of the product isotope, retrieval of the original target material for possible reuse, and disposal of the residual waste. Four examples of process separation are discussed to show the diversity of the separation methods.

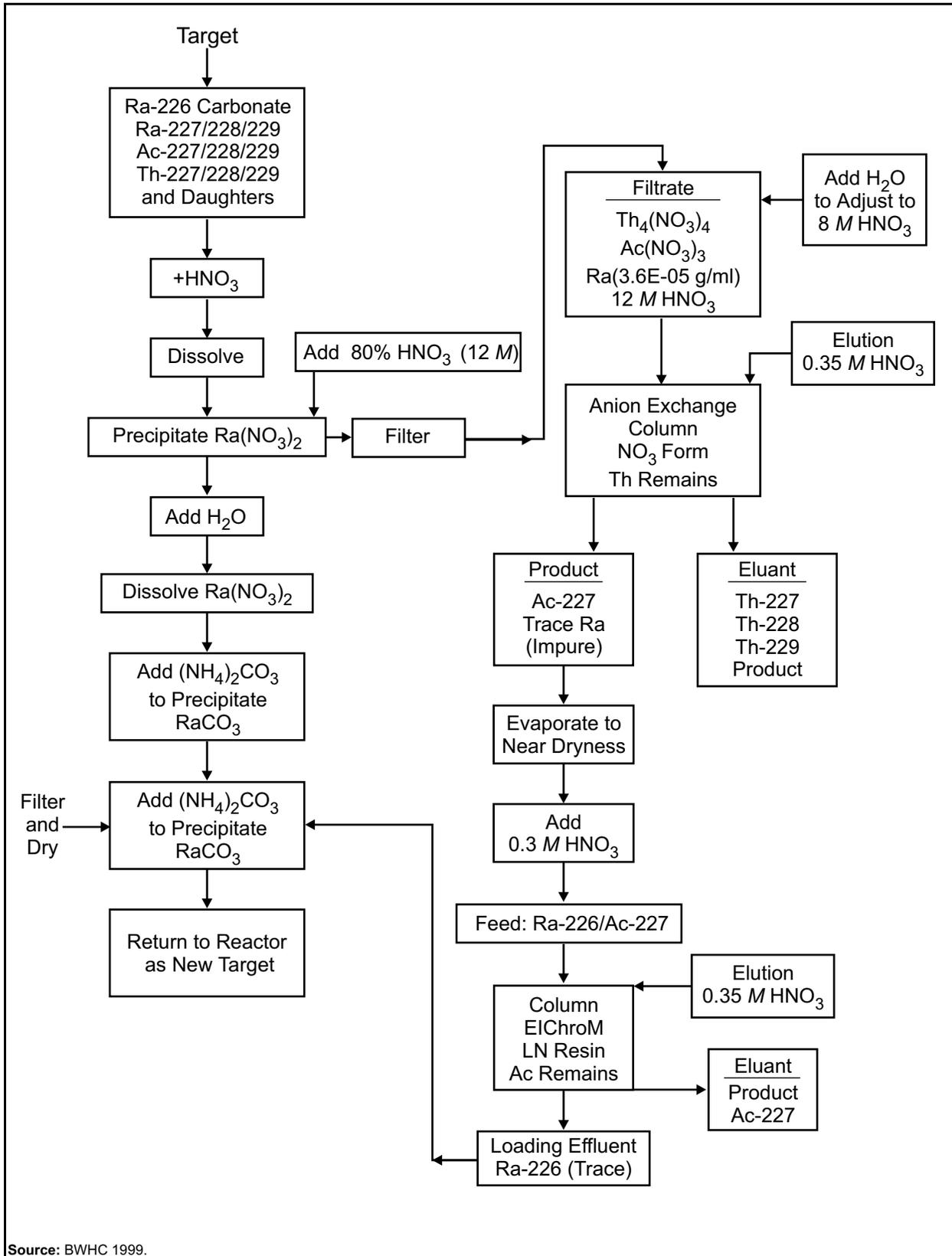
ACTINIUM-227 AND THORIUM-228/229 FROM RADIUM-226

Radium-226 would be the only target that would be radioactive before being irradiated. It would be used to produce three products: actinium-227, thorium-228, and thorium-229. The processing room would contain a high-level radiochemical cell, a leaded glovebox, and an open-face hood, all connected to a radon capture facility. The radon capture facility (temperature-controlled charcoal filter) is required to prevent release of radon gas from the facility while storing the original target material and processing the irradiated targets.

The capsule containing the irradiated target (radium carbonate) material would be transported to the hot cell in a shielded pig and opened for removal of the target material by cutting the cladding. The carbonate salt would be dissolved with acid (**Figure C-6**). The radium then would be separated by nitrate precipitation and filtration. The remaining thorium, actinium, and radium in the filtrate solution would be purified using ion exchange separation.

Detailed steps in processing the radium-226 would include removal of the radium carbonate solid from the metal capsule for dissolution in dilute nitric acid. An addition of 80 percent nitric acid to the mixture then would be performed to precipitate the radium as radium nitrate, followed by filtration. The precipitated radium would be retrieved by the addition of water to redissolve the radium nitrate precipitate. Ammonium carbonate then would be added to the dissolved radium nitrate solution to precipitate radium carbonate. The carbonate precipitate would be filtered, dried, and could be reused as target material.

The 12 *M* nitric acid filtrate from the radium nitrate precipitation, containing thorium, actinium, and traces of radium, would be adjusted with water to 8 *M* acid and loaded on to an anion exchange column. The thorium would be held on the column, allowing the actinium, radium, and other impurities to be captured in the



Source: BWHC 1999.

Figure C-6 Processing Procedures for Removal of Actinium-227, Thorium-228, and Thorium-229 Products from Irradiated Radium-226 Targets

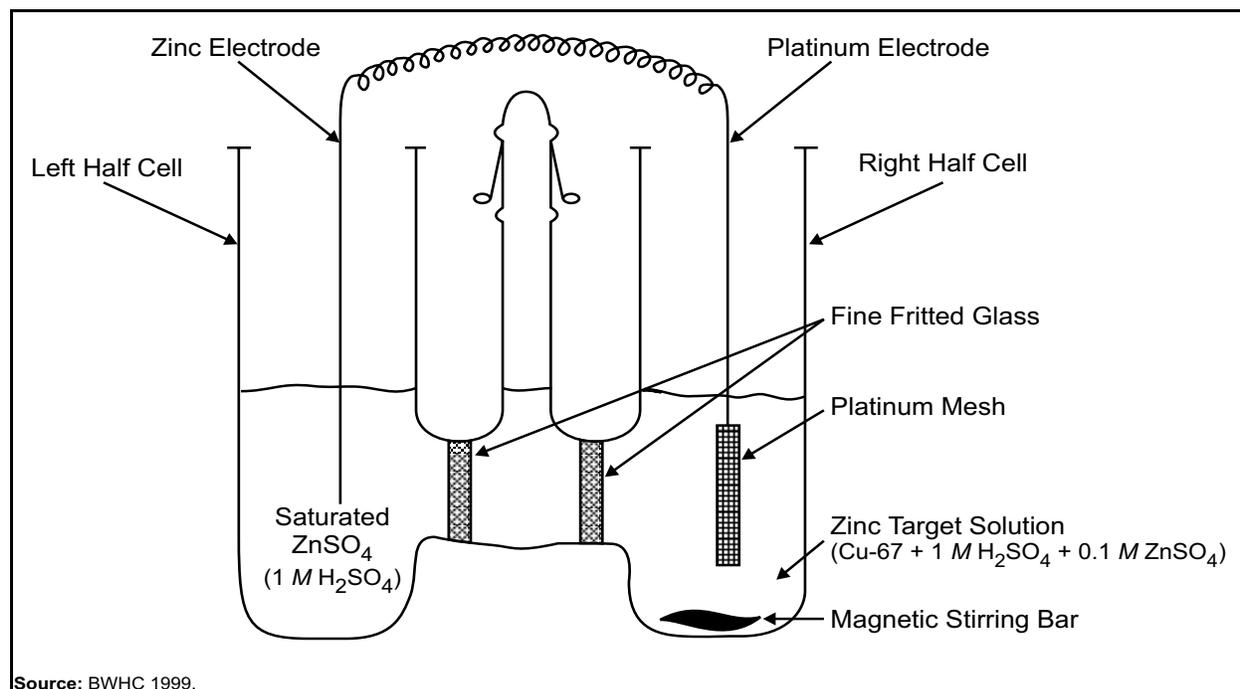
effluent. The purified thorium would be eluted from the anion exchange column with 0.35 M nitric acid. The thorium-228/229 product solution would be analyzed to ensure purity, and aliquots of the analyzed product would be placed in appropriate containers and shipped to customers.

The actinium-227 and traces of radium, found in the effluent solution from the anion exchange purification of thorium, would be evaporated to dryness and redissolved in 0.03 M nitric acid. This solution would be loaded onto an EiChroM resin column and the effluent containing the radium retrieved via carbonate precipitation. The resin column containing purified actinium-227 would be eluted with 0.35 M nitric acid. The actinium-227 product solution would be analyzed to ensure purity, and aliquots of the analyzed product would be placed in appropriate containers and shipped. The actinium-227 also can be used on site for recovery and shipment of radium-223.

All liquid waste would be neutralized and captured on a solid absorbent. Solid compacted waste (excluding cladding hardware and radon holdup charcoal) is estimated at 0.28 cubic meters (10 cubic feet) per year (BWHC 1999).

COPPER-64 AND COPPER-67 FROM ZINC-64 AND ZINC-67

The capsule containing the irradiated zinc oxide target would be transported to the receiving hot cell. Irradiated targets and carriers would be transported in shielded pigs from the hot cell to an isotope processing station containing a hot cell, a lead-shielded glovebox, a fume hood, and a laminar flow hood. The target material would be removed from the carrier by cutting the capsule containing the zinc oxide. The oxide would be dissolved with sulfuric acid. The sulfuric acid solution would be placed in an electrochemical cell and the copper deposited on a platinum electrode (Figure C-7). The platinum electrode would be removed from the electrochemical cell and copper-64 or copper-67 dissolved from the surface by immersing the platinum electrode in nitric acid. The zinc could be retrieved by converting it to the oxide form to be used as target material.



Source: BWHC 1999.

Figure C-7 Electrochemical Separation of Copper-67 Product from Zinc-67 Target Material

Detailed steps in processing the copper isotopes would include removal of the zinc oxide from the metal capsule for dissolution in 1 M sulfuric acid and transfer to the electrochemical cell. After 30 minutes, the copper would have been completely deposited on the platinum electrode. The target solution still containing the zinc would be removed and replaced with fresh acid and the copper deposition would be continued for an additional 30 minutes. This step would be repeated for a second time to assure purification. The deposited copper on the platinum electrode then would be removed from the cell and dissolved by immersing the electrode in concentrated nitric acid for 1 to 2 minutes. This solution would be evaporated to near dryness to remove the strong acid. The dried product then would be redissolved in an appropriate acid. The solution would be analyzed for chemical and radionuclide purity, and aliquots of the final product placed in containers and shipped to customers.

The zinc ion contained in the spent electrochemical solution could be retrieved, converted back to zinc oxide, and returned to the reactor for re-irradiation. If other unwanted metal ions are found in this solution, the zinc ion would be purified by ion exchange prior to oxalate precipitation and calcination to the oxide.

All liquid waste would be neutralized and captured on a solid absorbent as solid waste. Solid compacted waste (excluding cladding hardware and radon holdup charcoal) is estimated at 0.28 cubic meters (10 cubic feet) per year (BWHC 1999).

GADOLINIUM-153 FROM EUROPIUM AND SAMARIUM

The capsule containing the irradiated europium oxide target pellets would be transferred to a hot cell for dissolution and europium removal, followed by ion exchange band displacement in a heavily shielded glovebox. The pellets would be removed by cutting open the capsule containing the irradiated europium oxide. The europium oxide pellets would be dissolved with acid, greater than 99.9 percent of the europium isotopes would be removed by sulfate precipitation of europium (II), and ion exchange band displacement would be used to separate 0.1 percent of the remaining europium, along with the samarium and gadolinium, into fractional purified bands (**Figure C-8**). The final gadolinium-153 product would be precipitated, dried, and heated to a high temperature to form gadolinium oxide.

Detailed steps in processing the gadolinium-153 include removal of the oxide pellets from the metal capsule for dissolution in acetic acid. With argon sparging to prevent air oxidation, the solution would be contacted with amalgamated zinc (Jones reductor) to reduce the europium (III) to europium (II). A sulfate salt would be added to precipitate the europium (II), separating it from samarium (III) and gadolinium (III).

The resulting filtrate solution would contain gadolinium, samarium, and less than 0.1 percent of the europium. This solution would be transferred to a shielded glovebox to separate and purify the gadolinium-153 using band-displacement cation exchange chromatography. Ammonium-buffered chelating agents such as nitrilotriacetic acid or diethylenetriamine-pentaacetic acid would be used to fractionate gadolinium/samarium/europium using a zinc-loaded cation exchange column.

The purified gadolinium-153 product solution would be transferred into a second shielded glovebox for oxalate precipitation, filtration, and calcination to the oxide, and then pressed into pellets for shipment to customers.

Mixed liquid waste would contain zinc, acetic acid, 45 curies of europium per curie of gadolinium product separated, and ammonium diethylenetriamine-pentaacetic acid organic complexant (0.05 to 0.5 liters [0.013 to 0.13 gallons] of liquid waste per curie of gadolinium recovered). Solid compacted waste (excluding cladding and hardware) is estimated at 0.28 cubic meters (10 cubic feet) per year, including a Jones reductor. Ion exchange columns, sulfate precipitation equipment, and the pellet press would be reused and only become solid

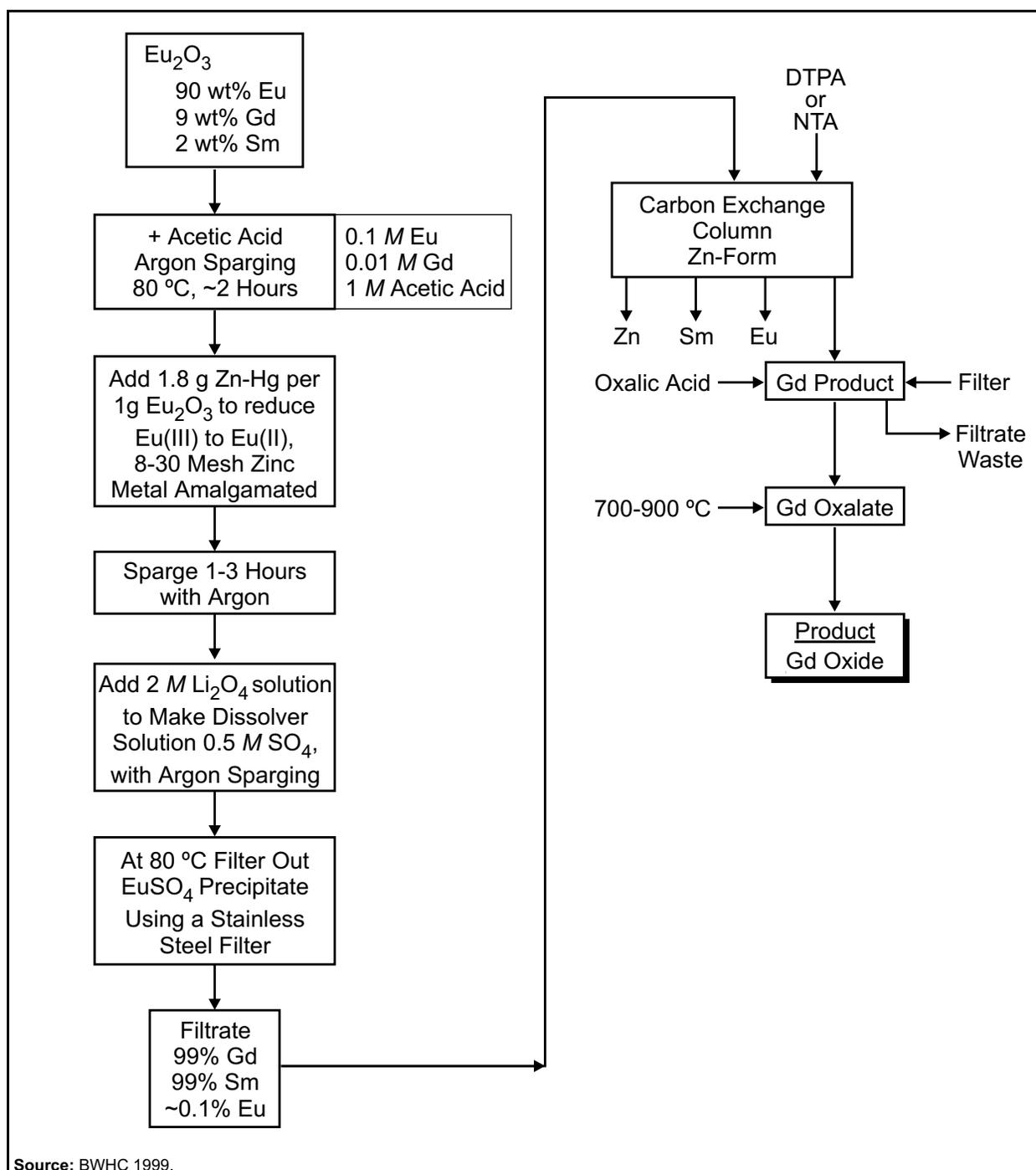


Figure C-8 Procedure for Separation of Gadolinium-153 Product from Europium Target Isotopes

waste at the termination of the project. This waste would be about 0.28 to 2.8 cubic meters (10 to 100 cubic feet) (BWHC 1999).

IODINE-125 FROM XENON-124 GAS TARGET

The method of production and separation would be based on the irradiation of 5 liters (1.3 gallons) of enriched xenon-124 for approximately 2.5 hours, trapping the irradiated gas, letting the xenon-125 gas decay to

iodine-125 for approximately 2 days, distilling off the inert gases, chemically reacting to remove iodine-125 from the wall of the cryotrap, followed by final processing, packaging, and shipping of the product. The conceptual system is shown in **Figure C-9**. The system would consist of several cold traps (cryotrap), flow restrictors, a gas canister, and a processing system (glovebox and cryopump).

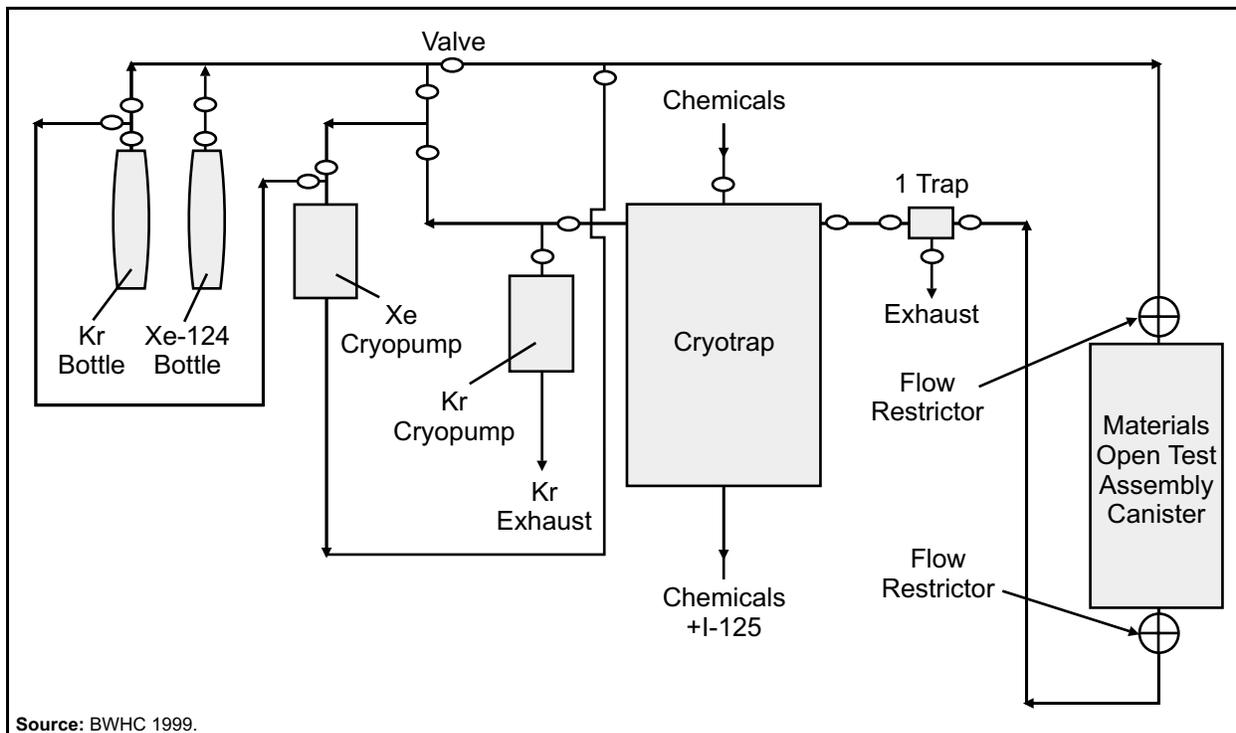


Figure C-9 Gas-Trapping Procedure for Separation of Iodine-125 Product from Xenon-124 Gas Target

Detailed procedures for producing iodine-125 are at the conceptual stage. Once the system is ready, the xenon-124 gas bottle valve would be opened. The rate of flow would be controlled by the flow restrictors located downstream and upstream of the canister. Once the xenon-124 bottle becomes empty, krypton gas would be valved in and used to push the xenon-124 first through the gas line and next through the canister. Once the gas has been pushed through the downstream flow restrictor, the gas would flow through the iodine trap to the cryotrap, which would be maintained at a low temperature and result in a low pressure (less than 10^{-6} torr). The xenon-124, transmuted xenon-125, and some krypton “pusher” gas would be absorbed on the cryotrap cold surface. The cryotrap inlet valve then would be closed and the krypton in the gas lines would be evacuated to the krypton cryopump.

After about 2 days with the xenon-125 (17-hour half-life) decaying to iodine-125 (60-day half-life), the cryotrap downstream valve would be opened and the cryotrap warmed to first distill the xenon-124 to the xenon cryopump, and then any krypton to the krypton cryopump. The product iodine-125 remaining in the cryotrap then would be chemically reacted (caustic) and the product analyzed, processed, packaged, and shipped.

C.4 REFERENCES

BWHC (B & W Hanford Company), 1999, *Hanford Data Request for FFTF Operational Support Facilities*, Richland, WA, October 6.

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