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Economical Production of Pu-238

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Abstract. All space exploration missions traveling beyond Jupiter must use radioisotopic power sources for electrical power. The best isotope to power these sources is plutonium-238. The US supply of Pu-238 is almost exhausted and will be gone within the next decade. The Department of Energy has initiated a production program with a $10M allocation from NASA but the cost is estimated at over $100 M to get to production levels. The Center for Space Nuclear Research has conceived of a potentially better process to produce Pu-238 earlier and for significantly less cost. The new process will also produce dramatically less waste. Potentially, the front end costs could be provided by private industry such that the government only had to pay for the product produced. Under a NASA Phase I NIAC grant, the CSNR has evaluated the feasibility of using a low power, commercially available nuclear reactor to produce at least 1.5 kg of Pu-238 per year. The impact on the neutronics of the reactor have been assessed, the amount of Neptunium target material estimated, and the production rates calculated. In addition, the size of the post-irradiation processing facility has been established. In addition, a new method for fabricating the Pu-238 product into the form used for power sources has been identified to reduce the cost of the final product. In short, the concept appears to be viable, can produce the amount of Pu-238 needed to support the NASA missions, can be available within a few years, and will cost significantly less than the current DOE program.

Keywords: Pu-238, isotope production

INTRODUCTION

Every mission launched by NASA to the outer planets has produced unexpected results. The Voyagers I and II, Galileo, and Cassini missions produced images and collected scientific data that totally revolutionized our understanding of the solar system and the formation of the planetary systems. These missions have been enabled by the use of Plutonium-238 (Pu-238). The conversion of the radioactive decay heat of the Pu-238 to electricity provides a long lived source of power for instruments. Unfortunately, the supply of Pu-238 is about to run out. Developing a reliable supply of Pu-238 is crucial to almost all future space missions.

Radioisotopic Thermoelectric Generators (RTGs) have been used in the past for all missions past Mars to provide electrical power to the platform. The recently deployed Mars Science Laboratory, Curiosity, however, utilizes Multi-Mission RTGs (MMRTGs) which can operate in the vacuum of space or in a planetary atmosphere. Because of the desire for no moving parts, reliability, and long life, these systems rely on thermocouples to convert heat to electricity and are inherently inefficient. Only about 6% of the thermal energy is converted into electricity. Consequently, RTGs and MMRTGs require large masses of Pu-238 to provide the heat to make the electricity. In addition, the specific masses of the RTG and MMRTG are 200 kg/kWe and 357 kg/kWe respectively [1]. Thus, the power supplies can be a significant fraction of the platform mass.

Recent advances in Stirling engines at the NASA Glenn Research Center indicate that Advanced Stirling Radioisotope Generators (ASRGs) may provide 25% conversion efficiency [2]. ASRGs will reduce the amount of Plutonium-238 (Pu-238) required for a given power level by a factor of 4. However, ASRGs contain moving parts and may suffer from vibration issues along with shorter life-spans than MMRTGs. In addition, the specific mass of the ASRG is 141 kg/kWe.

With current NASA mission plans, all of the Pu-238 remaining on Earth will be consumed within the next decade. After this mission, no spacecraft will travel beyond Jupiter or within the orbit of Mercury. The DOE has proposed

to produce Pu-238 using the Advanced Test Reactor (ATR) and the High Flux Isotope Reactor (HFIR) reactors at the Idaho National Laboratory (INL) and Oak Ridge National laboratory (ORNL) respectively. These reactors produce high fluxes of thermal neutrons and are very appropriate for Pu production. Recent estimates of actual production of Pu-238 indicate a rate of 1.5 kg/yr. However, the DOE has estimated that the cost to reach full production will be around $100M. With the increased pressure on the US Congress to cut spending, the Pu-238 production program may slow or be halted entirely. In 2012, NASA allocated $10M to DOE to start the production effort.

Current production methods rely on neutron irradiation of large samples of a few kgs of Neptunium-237 (Np-237) for a period of up to one year. The Np-237 will capture a neutron to make Np-238 which decays in 2.117 days to Pu-238. Unfortunately, the Np-238 has a very large fission probability so that around 85% of the Np-238 that is produced is destroyed before it can decay. In addition, after the irradiation, the large sample must be processed for the Pu-238 to be removed and accumulated. The facility needed to handle large quantities of highly radioactive material is large, complex and costly. In addition, large quantities of radioactive, acidic waste is generated that must be processed or stored for long periods.

An alternative method is to continuously flow an encapsulated aqueous solution containing a high concentration of dissolved Np-237 in a water carrier stream. The use of discrete capsules will make the separation process safer, cleaner and the sampling process more efficient. The capsule (made of one of several viable polymers) also provides another layer of thermal moderation to take advantage of the high thermal absorption cross section of ^{237}Np. In addition, if there is a pipe break in the water stream carrier, the capsules are easily retrieved and the reactor is not contaminated by the water stream which means there will be no change in reactivity in the nuclear reactor.

Once the irradiation period is completed, the encapsulated target slowly moves through the reactor's water tank and allows time for the decay of Np-238. After irradiation, the capsule contents are individually run through a standard ion exchange column to remove the Pu-238 specifically. This is the same process currently used by the DOE but at a smaller scale allowing for smaller and a less costly separation facility. This process allows small quantities of Pu-238 to be processed on a weekly basis so that a much smaller, and less costly, facility is needed to accumulate the Pu-238.

During the Phase I effort, the CSNR recognized that the new process will produce a substantially smaller waste stream of radioactive acidic solution, i.e. mixed waste, which has to be stored or processed. The past method produces 1000s of gallons of liquid waste per year. The new method will produce a few kgs of resin beads covered with grams of fission products. Thus, the method will produce substantially reduced costs.

During the Phase I study, we also recognized another possible cost reduction to the overall process of making sintered pellets for the MMRTGs, i.e. the "back end" of the process. The current process uses solid Pu-238 and ball-mills the material to make a distribution of powder. Some of the particles are sub-micron in size. These small particles migrate through seals in glove boxes and are responsible for all of the worker exposures over the past few decades. The CSNR has identified a potential process that will accept the aqueous solution resulting from the production process and produce large diameter spheres of Pu-238. Compaction of the spheres into a standard "pellet" geometry with the correct physical properties would enable less handling by human workers, a reduced facility footprint, reduced cost, and a smaller waste stream.

The CSNR has evaluated the potential performance of a continuous flow system by: 1) determining the maximum concentration of Np-237 that can be present in the capsules without suppressing the neutron flux too far, 2) determining the optimum residence time in the reactor to maximize Pu-238 production, 3) predicting the impact of cooling the flow stream to increase the neutron absorption, 4) designing post-irradiation separation chemical process of Pu-238 and 5) evaluating any chemical reactions, erosion, or out-plating of the Np in the system.

CONCEPT DESCRIPTION

The process proposed by the DOE for Pu production is a proven method which has been used successfully in the past. In essence, long aluminum tubes are filled with solid neptunium dioxide powder, 20 volume %, and placed

into the HFIR or ATR reactors for over 6 months. The problem with this process is that the Np-238 has a drastically large probability of fissioning before it decays. Thus, for long irradiations, around 80-90% of the Np-238 is destroyed before it can decay. This means that the targets have a large inventory of fission products making them highly radioactive and very hard to handle and process. Consequently, the facility necessary to handle several kgs of highly radioactive Np is large and expensive. In addition, a large waste stream of highly radioactive solution is produced that must be stored or processed. According to R. M Wham of the Oak Ridge National Laboratory in a paper presented at the Lunar and Planetary Science Conference, March 2012, "thousands of gallons of radioactive, acidic solution and several 5 gallon drums of trans-actinide waste are produced and need to be stored."

An alternative method that would enable higher production efficiencies of the Pu-238, is to flow the Np in a solution through the core of the reactor and extract the Pu-238 continuously at lower mass rates. This process produces Np-238 and then removes it from the reactor before substantial fraction can burn up and be lost. It also allows for much smaller processing facility because smaller amounts are processed continuously and the material is not full of fission products. The waste stream is, thus, significantly reduced.

The flowing target scenario is not possible to implement in the ATR or HFIR without major interruption of service and extensive cost. However, the flowing target can be implemented in a new, commercially available reactor. While several vendors can supply such a reactor, this study has used the TRIGA reactor available for General Atomics as a test case.

The primary innovation of this proposal is the continuous flow feedstream and the use of capsules to "quantize" the feedstream into small units to be treated individually. The capsules allow a solution with a high concentration of Np-237 to be irradiated without the risk of the Np plating out along the flow channel. Use of the capsules also allows small quantities of irradiated material to be processed at a time which reduces the processing facility's size even more. Finally, the use of capsules mitigates the risk of having the pipe break and spill the target solution into the reactor or facility. What is required, then, is a reactor with a high thermal flux of neutrons and the ability to alter its design to accommodate the flow channel. The residence time in the core, the decay time out of the core, and the processing of the irradiated solution must then be determined.

Production fundamentals

A continuous flow process to produce Pu-238 will requires a thermal neutron flux to take advantage of the very high thermal absorption cross section ($\sigma_{absorption}$) in the nuclear reaction

$$^{237}_{93}\mathrm{Np} + {}^{1}_{0}\mathrm{n} \xrightarrow[\sigma_{absorption}=150\,barns]{} {}^{238}_{93}\mathrm{Np} \xrightarrow[2.117\ days]{\beta-decay} {}^{238}_{94}\mathrm{Pu}$$

compared to a competing nuclear reactions

$$^{237}_{93}\mathrm{Np} + {}^{1}_{0}\mathrm{n} \xrightarrow[\sigma_f=0.02barns]{} \text{Fission poducts} + {}^{2}_{0}\mathrm{n}$$

$$^{237}_{93}\mathrm{Np} + {}^{1}_{0}\mathrm{n} \xrightarrow[\sigma_{n,2n}=0.2barns\ in\ fast\ region]{} {}^{236}_{93}\mathrm{Np} + {}^{2}_{0}\mathrm{n}$$

The third reaction leads to the production of Pu-236. Pu-236 is a contaminant that must be kept to below a 3 ppm level.

Figure 1 shows the difference in the three cross sections [3] with the plot in red being the absorption cross section for ^{237}Np, green plot is the fission cross section and the blue plot is the n,2n nuclear reaction producing Pu-236. Figure 1 clearly shows the need to have a very low energy, i.e. thermal, flux expose the target material. A heavily thermal flux of neutrons will produce Np-238 but not cause fissions in the Np-237 nor produce Np-236 leading to Pu-236.

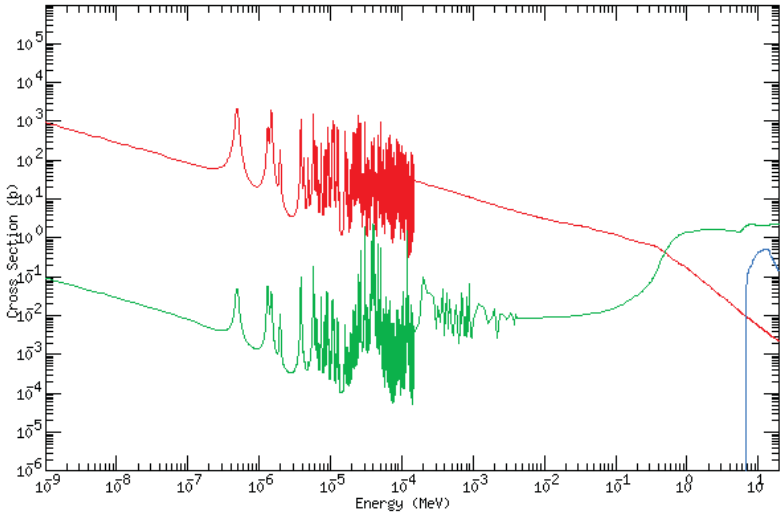

Figure 1. Plot of the energy dependent microscopic cross section, i.e. reaction probability, for Np-237 absorption in red, Np-237 fission in green and Np-237 to Np-236 in blue [3].

Typically, neutrons are born in the fission process at near 1 MeV. Letting the neutrons pass through water allows them to thermalize to energies near 0.025 eV. However, as the neutrons pass through the water they are absorbed so that the flux level is reduced. Thus, optimization of the Pu-238 production entailed an evaluation of where in the reactor core the target material is placed.

The neutron flux spectra for three different locations in a 1 MW TRIGA core at the Kansas State University (KSU) reactor are shown in Figure 2. The three locations in the KSU reactor are the central thimble (CT), RSR Well position and 2cm outside of the reactor's reflector. The central thimble is at the center of the reactor, the RSR Well position is on the outside of the reactor core but inside the reflector, and the last position is 2cm outside of the graphite reflector [4].

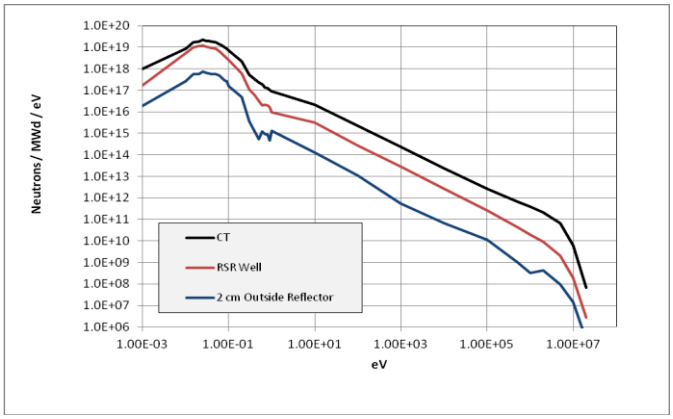

Figure 2. Spectra of neutrons at three locations in the 1 MW TRIGA reactor at Kansas State University.

The spectra show that the neutron fluence outside the core is more thermal, i.e. the ratio of low energy to high energy neutrons is bigger than at the other two locations. However, the magnitude of the flux is lower by two orders of magnitude. Placing the target material in the CT location will breed more Pu-238 by also more Pu-236. Based on these data, the optimum position in the TRIGA core appears to be at the RSR location.

PROJECT RESULTS

Simulations of reactor design were performed using the comprehensive modeling and simulation suite for nuclear safety analysis and design SCALE6.1 package. SCALE6.1 contains different sequences for reactor analysis. The reactor analysis and activation Np-237 calculations were performed using the TRITON sequence. The simulations were done using a 238 energy ENDFVII multigroup cross section library that was processed with BONAMI for the unresolved resonance self-shielding calculations, CENTRM for a 1-D point-energy flux calculation, and PMC to use the flux and point cross section data to generate a problem specific multigroup cross sections library.

The initial approach used in this study was to model a generic TRIGA configuration. TRIGA reactors were selected because their core configuration flexibility. In TRIGA reactors, fuel can be shuffled and placed in different positions around the core in order to achieve the desired flux profile at irradiation experiment locations. It was important to start with a generic known reactor in order to learn how the activation of the Np-237 target behaves and to make an initial estimation of flux requirements. Three different reactor configurations each with different modifications were modeled.

Traditional solid target processes and non-traditional liquid targets processes usually placed the Np-237 target inside experimental traps within nuclear reactor core. The first step of our study was to model a known hexagonal TRIGA reactor. The model is a 1 MW TRIGA core based on the University of Utah reactor [5]. The hexagonal core diameter is 53 cm with a 58.4 cm height. There are 77 twenty percent U-235 enriched zirconium hydride fuel pins with a 3.3 cm diameter and a 38.2 length. There are also some D_20 water and H_2O rods along with three control boron rods. Two different Np-237 target configurations were simulated using the 1 MW TRIGA model.

The first configuration was modeled as a tool to learn the limitations and parameters of a traditional process. The second model was the first attempt to place the target outside the fuel core boundaries. For the first 1MW TRIGA configuration the Np-237 targets were placed in different experimental locations inside the core fuel boundary. This configuration simulation of the 1 MW TRIGA reactor determined that the Np-237 encapsulated target when placed inside the reactor acts like a neutron absorber making it difficult to maintain criticality and limiting the amount of Np-237 that can be activated, therefore limiting Pu-238 production.

The second configuration for the 1 MW TRIGA reactor model has the Np-237 target outside the reactor fuel core. This second target configuration of the 1 MW TRIGA was examined because it more closely simulates the desire target placement of having a coil pipe target around the reactor. This second configuration established that placing the Np-237 target outside the reactor fuel boundaries will not decrease criticality; in fact it will act as a reflector. Having the target acting as a reflector signifies that the amount of Np-237 will not negatively affect the performance of the reactor. This discovery was unexpected because the initial thought was that no matter where the Np-237 target was placed it was going to negatively affect criticality, this meant that the amount of Np-237 mass had to be limit in order to avoid shutting down the reactor or having difficulties controlling it. However, by placing the target outside the reactor fuel boundaries the critical assembly is not disturbed by the introduction of negative reactivity. Having the target outside the reactor actually introduced some positive reactivity into the core, however, it can easily be accounted for in the final reactor design by changing the worth of the control rods.

The study established that it is possible to produce enough Pu-238 using a liquid target outside the reactor core if the average neutron flux required at the target is achieved. A total average neutron flux of 5.99E+13 n/cm^2*sec at the Np-237 target is needed produce 1.5 Kg of Pu-238. This is the first time that a study using this configuration with an encapsulated target have proven to be effective. The production can be further maximized by having a fuel blanket surrounding the Np-237 target however this configuration will increase design and production cost. The feasibility study shows that production can be achieve using a TRIGA reactor. However, any reactor that satisfies the target placement, core configuration and flux requirements could be used to produce Pu-238 in a encapsulated continuous manner.

Initial reaction rate modeling

A major focus of Phase I was on assessing whether a low power reactor had sufficient neutron flux to produce the amount of Pu-238 required without producing too much Pu-236. To produce Pu-238 in a flow loop around a nuclear reactor requires the coordination of several process components for the flowing target stream within a nuclear reactor.

Initial calculations assumed a spatially uniform flux with an order of magnitude of $4x10^{13}$ nts/cm^2-s which relates to roughly a 2.8 MW TRIGA style reactor. This estimate is based on information from the KSU measured flux at 1 MW and the McClellan Nuclear Research Center at the University of California Davis (UC Davis) [6] . The calculations assumed 10 kg of Np-237 dissolved into HNO_3 flowing around the reactor for the irradiation time frames. The 10 kg limit is based on the amount of Np-237 that will dissolve into solution which is 432 mg/cm^3 under standard conditions [7], and the surface area of the reactor face for irradiation. From the 432 mg/cm^3 limit a target tube diameter size can be estimated, which is 3 cm. The calculations show that the CT, RSR and outside locations in the TRIGA produced 15, 2, and 2 ppm of Pu-236 respectively.

Another major issue is the balance of the residence time Np-237 stays exposed to thermal neutrons to become Np-238. The removal of the Np-238/ Np-237 from the neutron flux at an optimum time maximizes the amount of Np-238 that can be extracted and minimizes the amount lost through the competing nuclear reactions. Np-237 is recycled with this process so a shorter irradiation time can be implemented since the unreacted Np-237 can be sent back through the reactor loop to produce more Pu-238. The optimum is based on production of Np-238 and this can be seen in Figure 3.

Figure 3 shows the number of atoms produced versus irradiation time in the reactor at the three different locations. The optimum amount of Np-238 made is with a 17days to 19 days irradiation, depending on which location the target is placed. The 1.5kg/yr production rate can be achieved with a shorter irradiation time frame, but the optimum time for producing the most Pu-238 is a 17 day irradiation for the RSR Well position or 2 cm outside the reflector.

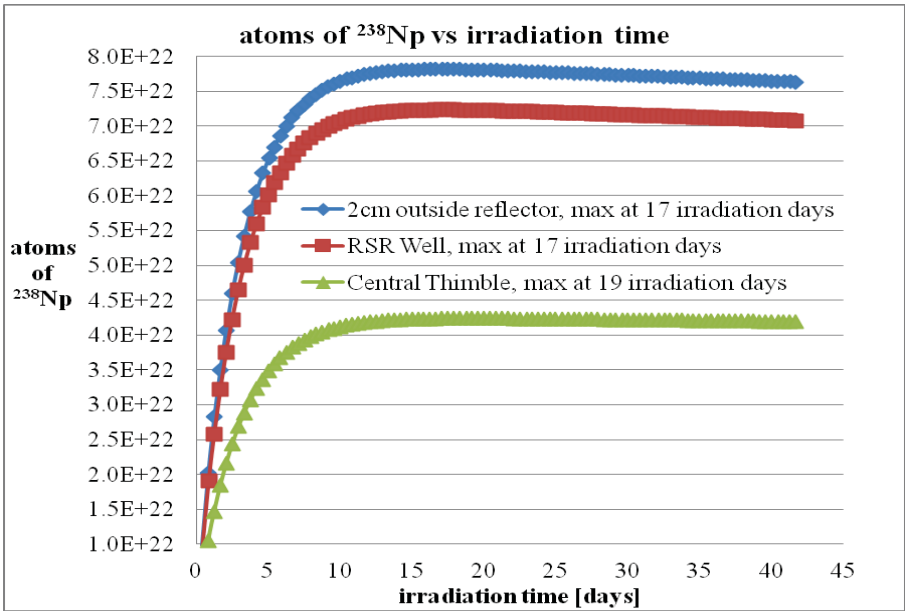

Figure 3. Optimum time for irradiation of flowing Np-237 target

System modeling

To establish various correlations with reactor power and irradiation time, a parametric trade study has been performed. In essence, the flow of capsules was modeled around a core whose diameter varied as the square root of the power level. The diameter dictated the length of the pipe surrounding the core which in turn dictated the capsule velocity in order to match the desired irradiation time. Given the capsule velocity, the length of pipe necessary for the decay time to elapse was determined. The pipe lengths dictated the mass of Np-237 residing in the reactor tank. Thus, by using the estimated production rate of Pu-238 per kg of Np-237 versus irradiation time and the maximum concentration of Np-237 per capsule, we can parametrically estimate the annual Pu-238 production as a function of neutron flux level.

The trade study used a TRIGA as a default reactor design. Possibly, other vendors may have reactors available in the 1 to 10 MW power class but the TRIGA is a well known design and is a good example of the class. In addition, the measured neutron spectra for the 1 MW TRIGA system was readily available. The actual correlation between neutron flux level and power level may not be well established for all different reactor types. The calculations assumed that the flux in the same position in a TRIGA reactor is linear with power level. Thus, a flux of 1.4×10^{13} n/cm2-s per MW was assumed for the trade study.

Results for a single point design for a 5 MW reactor are summarized in Table 1.

Table 1. Point design of the Pu-238 production system

Annual Production Pu-238(kg/yr)	2.25
Capsule diameter (cm)	4.0
Capsule length (cm)	4.0
Neptunium concentration (gm/cm3)	0.432
Feedline diameter (cm)	6.0
Reactor height (cm)	38.0
Reactor diameter (cm)	70.0
Irradiation time (d)	18.
Decay time (d)	12.
Processing time per capsule (s)	4392.
Capsule velocity (m/hr)	0.033
Number coils core	6.3
Number coils decay line	0.7
Length feedline core (m)	14.0
Length decay line (m)	9.4
Mass Np around core (kg)	7.65
Mass Np in decay line (kg)	5.1

The analysis allows for variable capsule sizes, variable reactor power and neutron flux levels, and determines the mass of Np-237 and the amount of Pu-238 produced. In short, at a given reactor power level, there is an optimum amount of Pu-238 that can be produced due to the requirement of a fixed capsule velocity around the core. Consequently, the trade study does not vary the cycle time of the capsules to just produce the 1.5 kg/yr but, instead,

determines the mass of Pu-238 produced by that power reactor for a cycle time commensurate with the capsule velocity through the reactor.

The results in Table 1 show that for roughly 13 kgs of Np-237 in the entire line, a 5 MW reactor can produce the 2.25 kg/yr of Pu-238. This value varies with reactor power. The amount of Np used is an important factor both from the cost standpoint, i.e. the latest quote is $200,000/kg-Np-237, and from a waste production standpoint, i.e. more fissions occurring in the Np.

The optimum amount of Pu-238 produced as a function of reactor power is shown in Figure 4. The Figure shows a slightly non-linear dependence with improved performance at higher powers. This dependence actually reduces the price per kg of the Pu-238 (see cost section below).

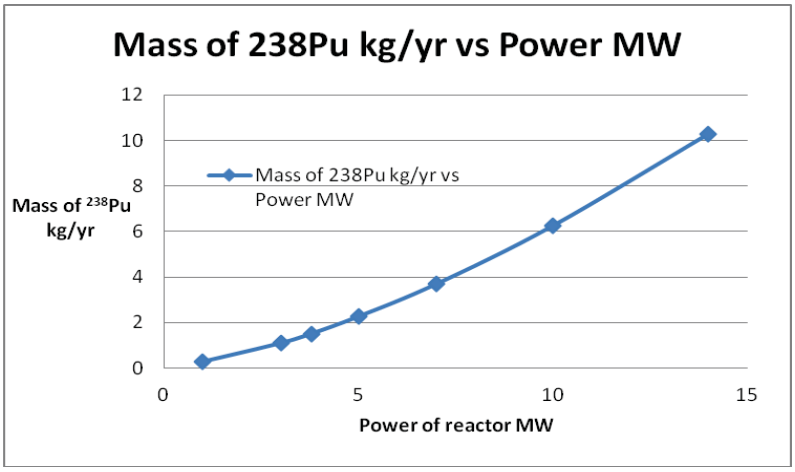

Figure 4. Optimum mass of Pu-238 produced as a function of reactor power. The neutron flux used in the optimization is assumed to be linear with power.

The trade study also allowed confirmation of the optimum irradiation time. One possible Figure of Merit (FOM) is the mass of Pu-238 produced per unit mass of Np-237 per time of irradiation. The desired goal is to have the most Pu produced for the least amount of Np in the shortest time. The results of the trade study were used to generate the FOM shown in Figure 5. The figure clearly shows that an optimum time is in the 12-15 day interval. By adding in the factor of mass Pu-238 versus mass fission products, which should be maximized also, the optimum time of 18 days was found for the irradiation time.

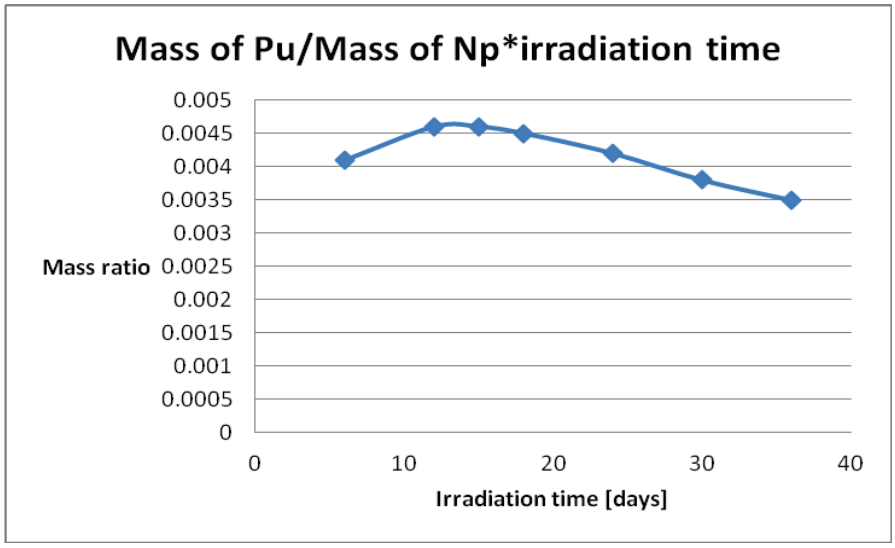

Figure 5. Plot of the Figure of Merit versus irradiation time indicating an optimum near 12 to 15 days.

Sphere fabrication

Early in the Phase I study, the issue what form the final product from the irradiation should take was raised. The issue hinges on the current process to make clad pellets for MMRTGs or ASRGs. The process involves ball milling of the Pu-238 powder and then sintering the ball milled product into shape. During the ball milling stage, sub-micron particles are produced which are very mobile and difficult to contain. All human exposures to Pu-238 at LANL over the past few decades are the result of sub-micron particles migrating through the seals on glove boxes.

The quantized, continuous flow process produces resin spheres covered in the Pu-238. By washing these spheres with a suitable solution, the Pu-238 is removed and collected as a solution. Conceptually, this solution can be fed directly into a process that creates large diameter, perhaps 100 micron, spheres. These spheres would not be mobile not be able to breach glove box seals. Consequently, the "back end" process was seen as an extension of the goal of the Phase I project to reduce costs of making radioisotope power units. An experimental effort was developed to demonstrate the production of spheres directly from an acidic solution containing a trans-uranic element.

Collaborators at the University of Michigan's Neutron Science Laboratory have established a sol-gel microsphere production rig suitable for work with radioactive materials. The sol-gel apparatus uses the internal gelation method to fabricate spheres with tunable diameters less than a millimeter without dust generation. During the development and construction of the sol-gel rig, a non-radioactive cerium surrogate for plutonium was used successfully. Currently, depleted uranium is being acquired for use in the system to act as a nearer surrogate for plutonium.
Work with surrogates has indicated that the internal gelation sol-gel fabrication technique will offer substantial benefits over current precipitation and powder processing methods for Pu-238 sources. Current methods involve an oxalate precipitation of plutonium that yields a powder morphology requiring ball milling, which results in respirable fines. A major advantage of the sol-gel method is that no powders are produced. Plutonium nitrate obtained directly from separation of neptunium following reactor production is used as the feed solution for the sol-gel process. Plutonium remains in solution until it is formed into microspheres of the prescribed size. These gelled microspheres can then be washed, sintered, and mixed with tungsten powder for spark plasma sintering into the final fuel form.

Eventually, depleted uranium spheres will be used as a surrogate for Pu spheres. The spheres will be sintered using the Spark Plasma Sintering (SPS) furnace at the Idaho National Laboratory. The SPS has been used by the CSNR to fabricate a variety of shapes and materials with variable porosity and density profiles. The goal is to produce a

sintered pellet that is identical to the currently produced pellet for the General Purpose Heat Sources (GPHS) used in the MMRTG and ASRG.

COST ESTIMATES

The current method of production by the DOE has major costs associated with handling large quantities of highly radioactive material at a given time. This necessitates large facilities, hot cells with remote manipulation, and transportation costs. The target pins have to be stored in a shielded facility for the required decay time to convert the Np-238 to Pu-238. They are then moved to the area where they have to be dissolved in the nitric acid solution prior to flowing into the extensive separation lines. In short, the facility footprints and the staff associated with the complex handling scheme are significant costs of the current process.

Qualitatively, the continuous feed method does not require the facility space to decay the irradiated product because the feedline is designed to allow the decay within the water tank of the reactor, i.e. the velocity of the capsules matches the required irradiation time in the core as well as the decay time through the tank. Because we are treating one capsule at a time, the separation lines are very modest. Early indications are that the entire separation system may be able to sit on the top of the reactor or immediately nearby so that a separate facility is not required.

Initially, the Phase I project started evaluating two possible scenarios for locating the facility: 1) a private entity builds a new facility on a "green field" site and transports the Pu-238 product to a DOE laboratory for fabrication, or 2) a private entity leases land from the government at a DOE Laboratory. The advantage of scenario 1 was that the land may be cheaper and it would be owned by the private entity which would be attractive to investors. The disadvantages were that all security costs and general infrastructure must be created fresh and would be part of the overhead costs. In addition, the transportation costs over public lands from the reactor facility to the DOE national laboratories may be significant. The advantages of scenario 2 were that the security and utilities infrastructure exists at the DOE sites as well as safety and handling procedures. Also, the transportation would only be on-site and never cross public land. The disadvantages are an increased cost of the land and operation on government controlled land.

Early discussions with personnel at the INL allowed the project to quickly determine that scenario 1 was not a competitive concept. In essence, the amount of Np-237 that would be present on site will make the facility Category II. This requires significant physical security to be in place as well as substantial human security.

Scenario II entails placing the facility at an existing DOE Laboratory site that already has Category II facilities. In this way, physical security and infrastructure are present and the human security can be shared with the other facilities. The DOE has several national laboratories around the country but many of them are now surrounded by residential housing or otherwise unsuitable to house a new reactor. Among the DOE labs that could support a reactor facility that produces Pu-238, the costs of human security can vary significantly.

The cost of the human security will be lowest at the INL. Consequently, the Phase I recommendation is for the production facility to be placed at the Materials and Fuels Complex (MFC) at the INL. This recommendation has the added benefit of reducing the issues of transporting any Pu-238 as the Space and Security Power Systems Facility (SSPSF) which assemblies the MMRTGs is in the MFC site. Thus, the production, processing, and assembly could be accomplished in one location.

Preliminary results of the cost estimate are very dependent upon the reactor power required. The initial capital cost depends almost linearly on the reactor power level and is the largest single component. In addition, the amount of Np-237 present in the reactor is dependent on the power level. Finally, due to the requirement to fix the capsule velocity for the appropriate irradiation time, the optimum amount of Pu-238 produced, thus the revenue stream, varies with reactor power. Figure 6 shows the mass of Pu-238 produced versus reactor power. (To be clear, the power level shown is a correlation between neutron flux and power level. The measured flux of 1.4×10^{13} for a 1 MW reactor is linearly scaled in the figure, i.e. a 5 MW reactor will have 7×10^{13} n/cm2/s.) Figure 6 also shows the price per kg of Pu-238 to be charged to the government in order to make a) a 20% ROI and b) a 0% ROI, i.e. if the government elected to use this process and not make a profit. The figure shows that a 5 MW reactor could produce 2.25 kg of Pu-238 per year and would need to sell material for $7.8 M/kg for a 20% ROI and $3.5 M per kg for a 0%

ROI. For a 10 MW reactor, the prices are $4.3 M and $ 1.6 M for a 20% and 0% ROIs respectively. This is because the 10 MW system will optimally make 6.25 kg/yr. Thus, even though the initial cost for the reactor goes up, the revenue goes up faster and the cost per kg is less.

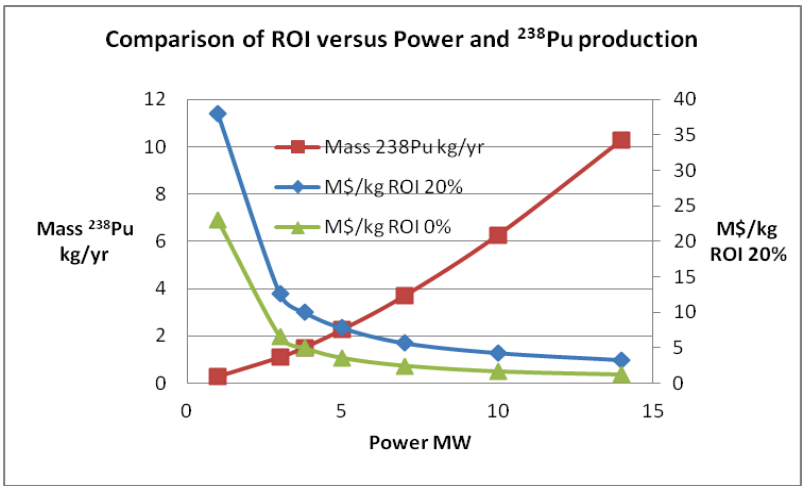

Figure 6. Plots of Pu-238 mass produced, price per kg Pu for 20% ROI and price per kg PU for a 0% ROI versus reactor power.

CONCLUSIONS

In the Phase I NIAC grant, the CSNR has evaluated the feasibility of using a low power, commercially available nuclear reactor to produce 1.5 kg of Pu-238 per year. The impact on the neutronics of the reactor have been assessed, the amount of Neptunium target material estimated, and the production rates calculated. In addition, the size of the post-irradiation processing facility has been established. Finally, as the study progressed, a new method for fabricating the Pu-238 product into the form used for power sources has been identified to reduce the cost of the final product. In short, the concept appears to be viable, can produce the amount of Pu-238 needed to support the NASA missions, can be available within a few years, and will cost significantly less than the current DOE program.

The alternative method to produce Pu-238 is to continuously flow an encapsulated aqueous solution containing a high concentration of dissolved Np-237 in a water carrier stream. Once the optimum irradiation period is completed, the encapsulated target slowly moves through the reactor's water tank which allows for time for the decay of Np-238 to Pu-238. This process allows small quantities of Pu-238 to be processed on a weekly basis so that a much smaller, and less costly, facility is needed to accumulate the Pu-238. One other aspect that has come out of the Phase I effort is the recognition that the new process will produce a substantially smaller waste stream of radioactive acidic solution, mixed waste, which has to be stored or processed. Thus, the method will produce substantially reduced costs.

In addition to the technical assessment of the production, the study sought to determine the answers to two major issues: 1) given a sufficient price per kg of Pu-238, could a sufficient return on investment (ROI) be possible so that private venture would pay the up-front capital costs saving the government this requirement in times of diminished budgets, and 2) is it more cost effective to install the new reactor on private land with private operations rather than locating the reactor at a DOE facility? The results of the study indicate that a 20% ROI is possible if the price per kg paid for the material is commensurate with the last known, circa 2007, asking price from Russia. The results also show that, due to lower security and transportation costs, the only responsible option is to locate the reactor at the Idaho National Laboratory site of the DOE.

The Phase I effort utilized experimentally measured neutron spectra from a 1 MW TRIGA reactor at the Kansas State University to estimate the Pu-238 production. Although several reactor types may be available for Pu-238

production, the TRIGA was used to model the production rates due to the large database regarding neutron flux and costs. By assuming a linear scaling of the neutron flux but keeping the neutron spectra the same, the production rate of the Pu-238, the concentration of Pu-236, and the amount of fission products could be calculated. The calculations show that an irradiation time of between 15 to 18 days with a 12 day decay time is optimum. Pu-236 contamination should be less than 2 ppm. The amount of fission products is estimated to be 150 gms/yr.

Using an 18 day irradiation time, the production rate versus neutron flux, i.e. reactor power, was determined. The trade studies indicate that a reactor at 3.8 or 10 MW can produce 1.5 or 6.25 kgs/yr of Pu-238 respectively. If a 20% return on investment is required, i.e. if the facility is privately funded, the price of the Pu-238 sold to the DOE would have to be 10 $M/kg and 4.3 $M/kg respectively. If no ROI is required, i.e. the US government funds the facility, then the price is 4.9 $M/kg and 1.6 $M/kg for the 1.5 kg and the 6.25 kg respectively. In either case, the results of this study indicate that the 1.5 kgs/yr of Pu-238 can be produced in a new facility within a 3-4 year timeframe for around $50M and return a 20 % ROI to an investor group.

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