INL/JOU-24-77213-Revision-0

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Prepared for the
U.S. Department of Energy
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

History of Pu-238 Production Restart Efforts at Idaho National Laboratory and Oak Ridge National Laboratory

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Abstract **In the early 2010s, efforts to restart production within the United States of plutonium-238 heat source material (HSPu) for NASA deep space missions were initiated. Processes, procedures, hardware, and chemical separations were developed and implemented to enable the production of heat source material at Oak Ridge National Laboratory (ORNL) and Idaho National Laboratory (INL). This review provides an overview of the timeline and efforts associated with the restart of production, as well as upcoming efforts to increase production.**

***Index Terms*— Plutonium, Isotopes, Nuclear Reactors.**

I. Introduction

Pu-238, a heat source material in the form of plutonium oxide (PuO_2), is used to power NASA deep space missions such as the Mars Perseverance Rover. It is preferred over other potential radioisotopes because of its long half-life of 87 years, high energy density of 0.57 W/g, and lower radiation field compared to other radioactive heat sources.

Future NASA missions need Pu-238 fueled Radioisotope Power Systems (RPS) to survive extreme conditions. Currently, the Multi-Mission Radioisotope Thermoelectric Generator (MMRTG) is in use for NASA missions like the Curiosity and Perseverance Mars rovers. It is powered by 4.83 kg of plutonium oxide, generating 110 We at the beginning of its lifespan. [1] The MMRTG is designed with flexibility in mind operating in a variety of harsh conditions such as planetary landers or deep space missions. [1] [2] Recent work on the Next Generation Radioisotope Thermoelectric Generator (Next Gen RTG) has focused on assuring that vacuum rated high power RTG's are available for future deep space missions. [3] Design studies were also performed on the Dynamic Radioisotope Power System (DRPS), which produce a 293 We DC power supply with thermal to electrical efficiencies of ~20%. [4] All of these RPS missions require a source of Pu-238 to provide thermal power to generate electricity. Given this need, a U.S. domestic source of Pu-238 has become an important part of continued U.S. space exploration.

A. Previous domestic production

The United States Department of Energy's (DOE) predecessor, The Atomic Energy Commission, initially assigned Savannah River Plant (SRP) in South Carolina the task of producing Pu-238 from Neptunium-237 (Np-237) feedstock in 1959. SRP was ideally positioned for this task because the feedstock harvested from Pu-239 production streams was located at the site. Targets were irradiated and source material was obtained through chemical processing. Annual production rates of this magnitude occurred until 1988, fueling numerous radioisotope thermoelectric generators (RTGs), including those used for the Voyager 1 and 2 missions, Pluto New Horizons, and others. During 28 years of operation, approximately 300 kg of heat source material was produced by SRP, subsequently named the Savannah River Site (SRS). Given the halt of domestic production, the DOE-owned inventory of Pu-238 decreased as continual production of RTGs consumed the heat source material. With the end of the cold war in 1991, the United States began sourcing Pu-238 from Russia in 1992 to augment the domestic heat source material inventory.

II. Restart Options

A. Options

A study was performed in the early 2010's evaluating options for restarting Pu-238 production to generate at least 1.5 kg of Pu-238 oxide per year by 2018 for a time period of at least 35 years. The driving requirements of this study were to lower risk, offer a near term solution for providing Pu-238 heat source material to NASA, and be compatible with existing production technology. Finally, the source material should not require development of additional reprocessing or separation technologies.[5]

All evaluated options involved irradiation at the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL) and Advanced Test Reactor (ATR) reactor at Idaho National Laboratory (INL). Np feedstock would be stored at INL for all considered options, while fabricated production targets at Babcock and Wilcox in Mt. Athos, VA, INL, and ORNL were evaluated. Target processing was evaluated at the H-Canyon at SRS, the Radiochemical Engineering Development Center (REDC) at ORNL, and the Remote Analytical Laboratory at INL. This study concluded that the lowest cost and lowest risk plan was for ORNL to fabricate and process production targets, target irradiation at ATR and HFIR, and for INL to store Np feedstock. This option also provided the shortest time to re-establish Pu-238 production in the U.S.[5]

B. Current Production Organization

The structure of the current Pu-238 production program mirrors the best option from the early 2010's study and is shown in Fig. 1. The lifecycle of this program starts with Np feedstock shipped from INL to ORNL for target fabrication. ORNL builds production targets, sends production targets to HFIR and ATR for irradiation, processes irradiated targets by removing Pu-238 and recycling Np, and stores Pu-238 for future shipment to Los Alamos National Laboratory (LANL).

Fig.1. Pu-238 production organization between INL, ORNL and LANL.

III. Early Work

Researchers at ORNL started work to design and qualify Pu-238 production targets for irradiation in HFIR. It was decided that the feedstock pellet would consist of aluminum powder and $^{237}NpO_2$ forming a cermet structure. A cermet structure was chosen because the requirements for developing a fully ceramic pellet was not fully understood at the project onset and aluminum cermet fabrication techniques were considered less risky. Given the lack of information on material and thermophysical properties for such a cermet pellet, the initial work performed supporting the restart of domestic Pu-238 production included characterizing the pellet irradiation dependent performance data and establish a HFIR safety basis. This research phase of the project consisted of three phases: 1) single pellet irradiations, 2) partially loaded target irradiations, and 3) fully loaded target irradiations. The graded approach allowed researchers at ORNL to systematically reduce risk while also learning and verifying safe operating conditions based on as-built and irradiated pellet properties such as shrinkage, thermal conductivity, thermal expansion, and fission gas release.

A. Material and Heat Transfer Characterization

1) Material and Thermophysical Property Characterization

Initial pellet and target irradiation vehicle designs were developed to study irradiation performance of the pellet and understand the safety margin relating to the HFIR safety basis. Much of the initial safety basis establishment was based on the californium – 252 (Cf-252) radioisotope production target that was produced at ORNL. That target had many design and safety similarities to early hypothesized Pu-238 production target concepts, such as the use of aluminum-based cermet pellets, accommodation of pellet shrinkage and fission gas release, high heat flux from fissionable/fissile transuranic isotopes, etc. The three phases of target evolution and qualification were designed to reduce risk and lessen overly conservative safety basis calculation assumptions; the knowledge gained from each phase was applied to the subsequent phase, as follows:

- Single pellet irradiation: understand pellet interaction with target structural materials (namely aluminum alloys), high-level verification of irradiation and thermal performance (i.e., pellets were not expected to melt or deconsolidate during irradiation, which could be verified during post-irradiation examination)
- Partially loaded targets: first inception of the finned tube target design contained eight prototypic pellets (20%-vol ceramic material, 10%-vol void, balance aluminum), continued verification of single and multiple HFIR cycle irradiation thermal performance and fission gas release, quantify cermet pellet shrinkage, etc.
- Fully loaded targets: similar outcomes as partially loaded targets but with a full complement of 50-52 feedstock pellets in a prototypic target to complete qualification.

Each target design was initially designed and analyzed based on "best available" data from thermophysical property data sets, pre-irradiation experiments, etc. Complex thermal-hydraulic models were produced to give a best estimate of thermal performance,[6] [7] and assumptions/data were refined from the outcomes of post-irradiation examination (PIE). Over the course of this early work, safety basis limits and margins were maximized allowing target designers to optimize the target design maximizing irradiation time and Pu-238 production within the pellet. More details on the target design effort are discussed later in this section.

2) Neutronic Characterization

Neutronic characterization efforts of the early targets faced similar challenges as the thermal property characterization work. Neutronics calculations are vital to support safety basis, irradiation behavior, transportation/storage, and programmatic efforts. Production of Pu-238 includes neutron capture in the Np-237 feedstock atoms to generate Np-238, which then decays with a half-life of about 2.12 days to Pu-238. Irradiation of Np-237 also produces other Pu isotopes and fission reactions resulting in high fission densities, heating rates, fission gases, and radionuclide inventories. The lack of confidence in relevant nuclear data (e.g., Np-238 fission cross section) forced researchers to impose overly conservative assumptions on target performance.

Also, the level of modeling sophistication required to calculate nuclear data accurately and efficiently (cross-sections, fluxes, reaction rates, heating rates, etc.) had to increase as the irradiation phases progressed because the target designs became more complex, higher resolution data in time and space became necessary, etc. Significant work was performed by ORNL researchers developing analysis methods and strategies supporting the target design evolution.[7] [8] [9] [10] Much of the PIE data, such as fission gas release, axially dependent gamma profiles, and pellet geometry changes, were also used to experimentally verify and refine the various computational models. Figs. 2 and 3 are representative PIE data scaled to neutronic fission density calculations and employed in thermal-structural-hydraulic calculations.

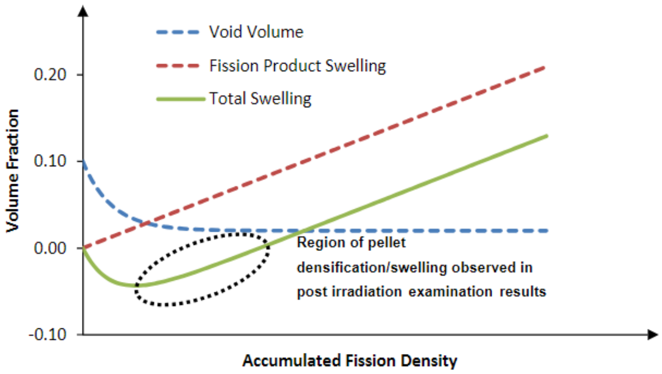

Fig. 2. Volume fraction of the cermet pellet vs. accumulated fission density.[11]

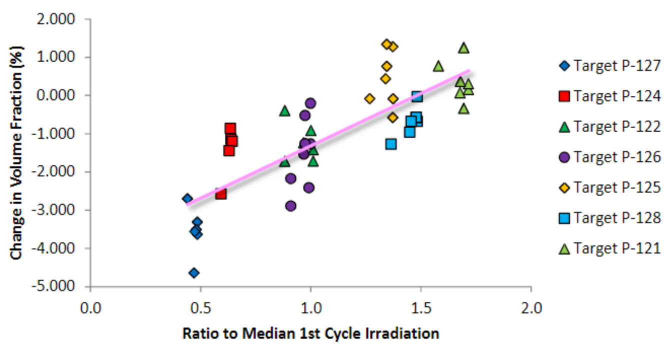

Fig. 3 Volume fraction change data of partially loaded targets.[11]

B. Target Design Development

For radioisotope production experiments, the primary design function (i.e., non-safety function) was to meet target production rates. This figure of merit generally simplified target design efforts because researchers did not need to be concerned with maintaining target internal conditions such as temperature, internal pressure, etc. to perform scientific studies on materials under those conditions. Therefore, radioisotope targets were generally not expected to breach the target containment or violate safety limits of the host reactor.

The single pellet irradiation target (Fig. 4) was designed to minimize heat flux from the pellet maximizing the thermal hydraulic margin in the reactor coolant, while also providing insight on the neutronic performance of the pellet. This initial data provided insight on the feasibility of irradiating a multi-pellet configuration. For example, the single pellet target included flux wires to independently verify flux spectra supporting other performance data that could inform neutronic performance (such as fission rates and fission gas release from the pellet). Primary design functions included maintenance of a hermetically sealed capsule containment that rejected pellet heat such that the containment did not fail (through melting) nor boil reactor coolant.

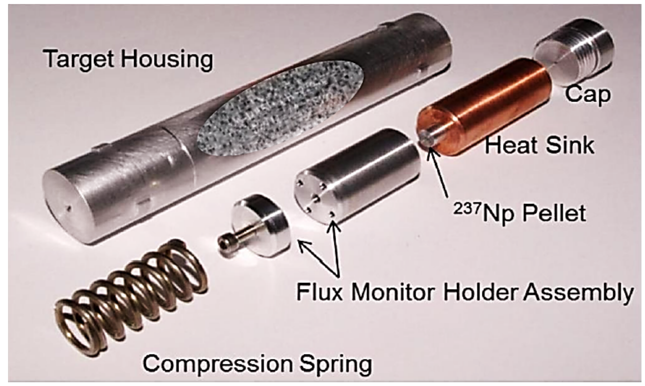

Fig. 4. Single pellet irradiation target and internal components.[12]

The partially loaded target housed eight cermet pellets, centered on the core horizontal midplane, between two tantalum pellets to reduce end power peaking effects during irradiation. Seven targets were loaded into a single holder for irradiation in the HFIR permanent beryllium inner small vertical experiment facilities (VXFs). An image showing the partially loaded target and holder is shown in Fig 5.

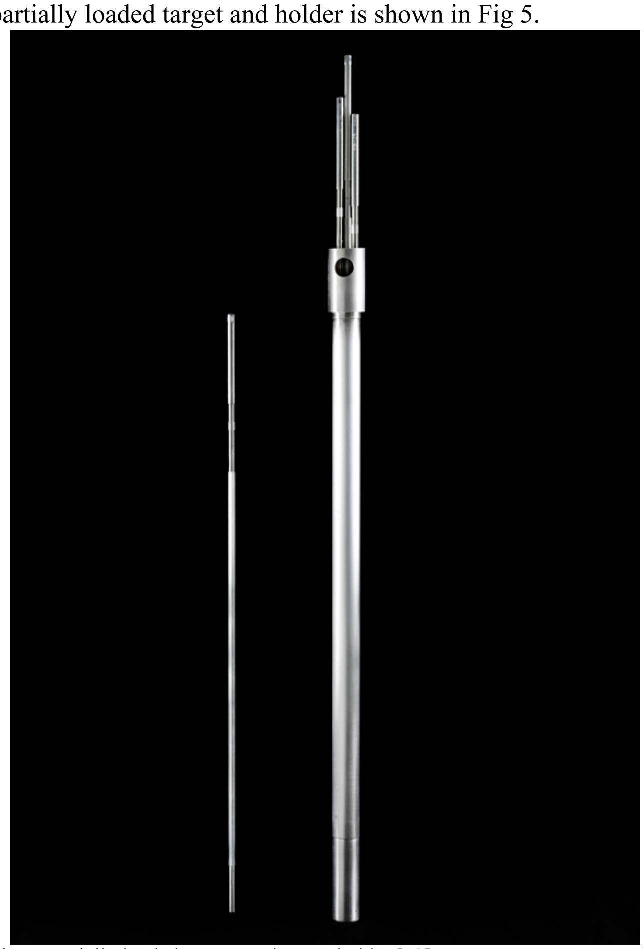

Fig.5. Partially loaded targets and target holder.[13]

This design was the first to use an Al-6061 (T-4 temper) finned extrusion tube with aluminum caps. Most of the target inner volume was void space allowing for an unknown amount of fission gas to be released without failing the containment. Fission gas release measurements were performed later, and the results helped optimize the target designs and reduce calculation conservatisms. Since the cermet pellets undergo shrinkage during irradiation, as indicated from early PIE, the target was hydrostatically compressed to eliminate the initial radial gaps between the pellet and target tube during irradiation, thus enhancing heat transfer from the pellet to reactor coolant. This design feature ensured a minimal gap, that was backfilled with helium gas, ensuring the best-case heat transfer over the irradiation period.

As with the single pellet design, the partially loaded target's containment was designed to prevent coolant from boiling while remaining intact during irradiation. To fully characterize the targets, significant weld development, destructive and nondestructive testing, and thermal hydraulic experimentation (e.g., out of pile flow testing) were performed. For example, temperature-dependent compression testing (Fig. 6) was performed on the pellets to verify that axial loading of the containment during irradiation would not fail the target.

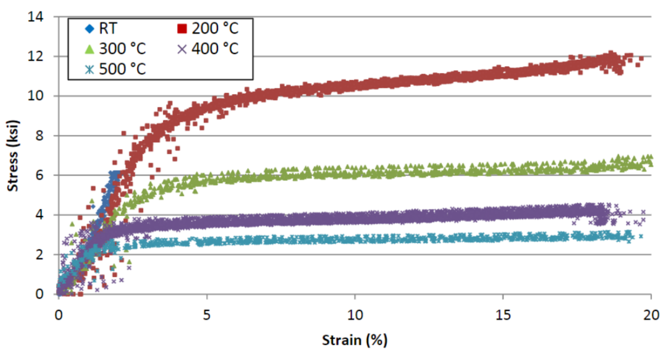

Fig. 6. Temperature dependent stress as a function of strain for cermet pellets [7]

The fully loaded target design facilitated irradiation of the maximum feedstock payload for a single target and enabled the project to start optimizing the chemical separations techniques needed to recover the Pu-238 product and the Np-237 feedstock. The target, referred to as the generation I "Gen I" target, housed between 50 and 52 pellets, as well as a number of design solutions (i.e., parts) to mitigate risks posed by a fully loaded target and other undetermined safety challenges. Fig. 7 shows the major design features of the target.

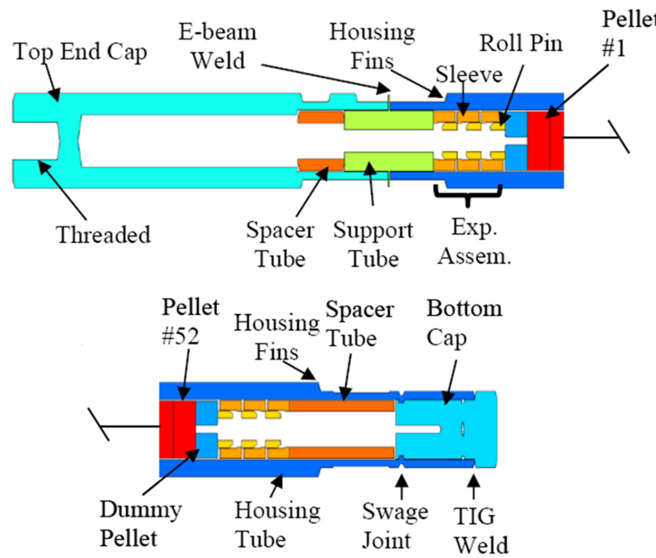

Fig 7. Schematic of the Gen I target (not to scale).[12]

Various attributes such as the dummy pellet, roll pin/sleeve assembly, and spacer tubes were used to ensure loading and in-reactor operations were successful. For example, there was inherent variability in the pellet stack length due to accumulated tolerance stack-up. Thus, spacer tubes were cut to length for each target ensuring the target body was internally supported and would not collapse during the external hydrostatic compression operation.

Seven targets were placed into a VXF target holder (seen in Fig. 5) designed to provide equal cooling for each of the target positions. Total coolant flow through the holder was controlled by an exit orifice, with flow evenly distributed for each of the seven targets. The target "assembly" comprising of the VXF holder, and the seven-target payload were shipped together for post-irradiation processing.

While the Gen I target design was effective at converting feedstock, and providing necessary performance data to relax stringent safety constraints, this design iteration was not optimal for fabricating targets at the rate required to meet the sponsor's heat source material production goals. Further, other aspects of the workflow, such as pellet pressing and inspection, were manually executed. Updates to these labor and radiation exposure intensive operations were needed to ensure continued success of the program.

C. Quality

Target campaigns, each of which contain multiple targets, had a fabrication file associated with them. These files included auditable data verifying all critical characteristics required for reactor insertion. This information included metrology data and composition of individual pellets, material pedigree for all components, weld inspections, and other information relevant to the fabrication of each target. The data reported in the file was also required for the pre-/post-irradiation shipping and processing of each target as it demonstrated that safety or regulatory limits were not exceeded. As for reactor qualification, the data was used to verify that as-built targets met safety basis specifications for the intended irradiation facility (either ATR or HFIR) and that there were no detrimental feeds of material that could be introduced into target processing. Fabrication files are not publicly available, but these records are maintained indefinitely by ORNL and, for ATR irradiation campaigns, by INL.

IV. Ramp Up at Oak Ridge

A. Initial Characterization

As stated earlier, the Gen I target was highly overdesigned, ensuring reactor safety constraints were met. Extensive testing and experience with successfully irradiated targets allowed for the quantification of the effectiveness of the primary containment and performance limits of the design. The project focused on a more streamlined version of the fully loaded target, eliminating extraneous parts, and improved fabricability without reducing target production limits or diminishing safety credit.

From the standpoint of yield quality, neutronic and thermal hydraulic predictions showed that irradiating Gen I targets for two HFIR cycles (roughly 50 days) produced the best quality heat source material. While more days of irradiation could yield more product, it was discovered that troublesome co-produced isotopes such as Pu-239 could not be chemically separated and the thermal safety margin was also reduced with longer irradiation durations due to conversion of fertile material during additional cycles. Furthermore, a third cycle of irradiation proved much less efficient than the first two because of the continuous consumption of feed material. Therefore, the initial systematic risk reduction and learning strategy initially employed for this project was successfully executed to inform a functional production target design.

B. Pellet Press and Metrology Automation

Another investment made by the program to improve efficiency and safety was to develop an automated pellet pressing and metrology workflow. The initial process was not only time consuming, but also exposed ORNL technical staff to high radiation doses during the pellet pressing work. The automated press and metrology technology ensured more uniformity in pellets, and the system collected measured data from the metrology instruments to compare geometric data to the set specification to automatically qualify the pellet for use. Automating the process allowed the program to significantly increase the production of pellets, more than triple the output. An image of the automated metrology system analyzing cermet pellets is shown in Fig. 8.

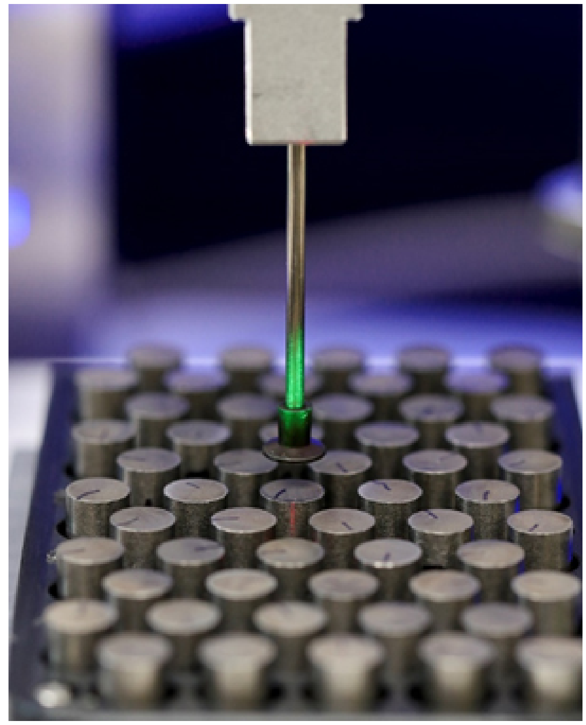

Fig.8. Automated pellet metrology system.[14]

V. Oak Ridge Regular Production

A. HFIR Gen II Targets

Second generation HFIR "Gen II" targets were modified to improve:

- Establishment of reliable weld processes,
- Optimization of internal component design features simplifying fabrication and assembly,
- Development of a quality assurance plan for receiving and accepting fabricated parts reducing nonconformances with no negative impacts on reactor safety protocols.[15] The strategy for designing and qualifying the Gen II target was to not make any functional changes violating the design and safety credit established by the Gen I design. In addition to ensuring no impacts to the integrity of target containment the coolant availability within the holder was conserved, not increasing heat produced by the target.

The redesign focused on extraneous parts and difficult welding operations. Earlier iterations included unreliable aluminum welding schedules. Further, the hydrostatic collapse could fail inferior welds, and recovering pellets from failed targets required chemically dissolving the entire target, which was expensive and resource intensive. Researchers invested in developing structural welds that used bi-metallic stainless steel to aluminum transition joints. This design change allowed researchers to perform a high volume of inexpensive electron beam aluminum welds connecting the transition joint to the aluminum finned tube, relying on more reliable stainless-steel welds for final closure. This order of operations eliminated closure weld failures. The Gen II target containment can be seen in Fig. 9.

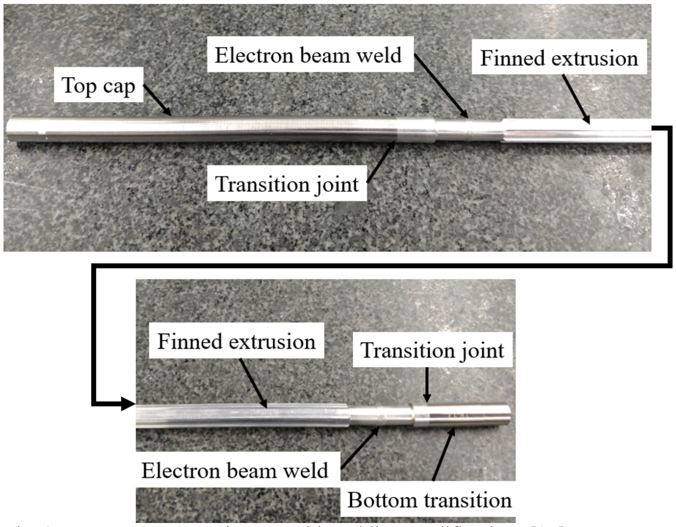

Fig. 9. Gen II target containment with welding modifications.[15]

The VXF target holder design was modified to comprise an inner holder containing the seven Gen II targets and a VXF "liner" that controlled coolant flow with an exit orifice. The inner holder was installed into the liner, and the whole assembly was installed into the VXF position within the HFIR. This new design was developed to allow the liner portion to be recycled, reducing radioactive waste. Detailed analyses and testing were performed to ensure adequate cooling was distributed to each of the seven target positions within the inner holder.

The Gen II design modifications enabled the ORNL target supply chain to pivot to a production level of throughput. The project established larger scale inventories of parts to maintain constant and reliable insertions of targets. This helped yield a portion of the heat source material that was included on the RTG powering the Mars Perseverance Rover.[16] A full list of the Gen II design modifications can be found in the literature.[15]

B. HFIR Gen III Targets

Given that the production targets for ^{238}Pu were ambitious, the project decided to pursue a common target design that could be irradiated in both of the DOE high-flux reactors, HFIR and ATR. Additionally, critical HFIR components necessitated shortening the existing target design by roughly 10 cm. The same approach to design, i.e. to not make any functional changes that violated the Gen I design and safety credit, was used to guide all design modifications employed in the third generation "Gen III" target design. The primary modifications for the Gen III design further optimized weld joints by combining the final seal and structural welds into a single operation, thus minimizing internal component lengths to shorten the overall target. Russell et al. described the full list of Gen III design changes in their work.[17]

Additional design modifications were developed to improve flow characteristics and reduce the overall length of the target holder. Primarily, the entrance region of the target holder was simplified, and the top and bottom cups were modified to improve flow performance. Figure 10 shows the Gen III holder, top, and bottom cups. Specific improvements include the simplified bottom cup coolant exit channels and coolant access channel which greatly reduced flow losses in this part of the design.

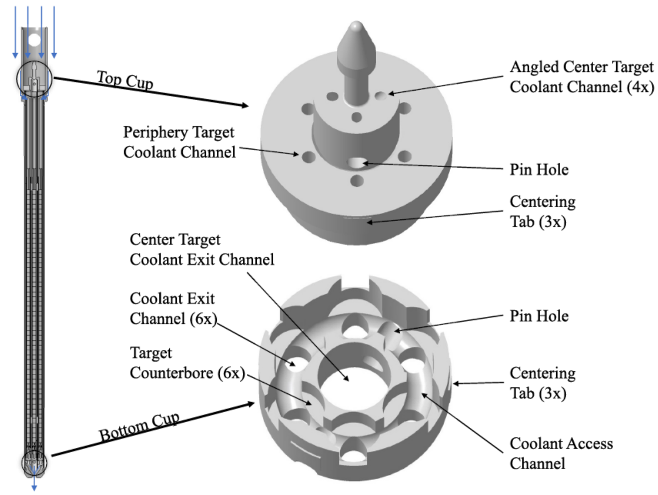

Fig. 10. Pictural representation of the Gen III holder modifications.[17]

VI. Idaho National Laboratory Initial Qualification and Irradiations

A. Flux Wire Measurements

The qualification process for Pu-238 production at INL began with irradiation of four NpO_2 flux wire dosimeters in the Advanced Test Reactor-Critical (ATR-C) facility. The ATR-C verified nuclear data and production target yields. This was important for estimating production as well as ensuring safe operation of the ATR.

In January 2018, flux wires were irradiated for 658 watts in ATR-C. A bare and Cadmium-wrapped dosimeter were placed in the I-7 position in ATR-C and irradiated for 20 minutes, with a two-hour cool down after irradiation. Predicted and measured results for production of Pu-238 were close enough to start early production in the ATR.[18]

B. I-7 Position Qualification

INL began irradiation qualification in the ATR medium I positions in 2018. Scoping studies were performed, and the neutron flux and energy spectrum in the I positions could yield Plutonium with a high Pu-238 assay, but would require approximately two years of irradiation to reach the needed assays. While this work was being planned, many I positions were available in the ATR due to low demand from other programs. An irradiation housing containing seven HFIR Gen II type targets was irradiated in the ATR-C to characterize the power distribution and neutronics in the ATR core. This testing demonstrated that Pu-238 production in the I-7 position would not impact the safe operation of the ATR.

Irradiation in the I-7 position began in ATR cycle 166A, continuing through cycle 169A. This irradiation campaign resulted in more than 300 irradiation days. Upon completion of this irradiation, the production targets were stored in the

ATR canal, awaiting processing at REDC at ORNL.[19]

C. South Flux Trap Qualification

The South Flux Trap became available for cycle 169A and a qualification effort began to utilize this position. The flux traps in ATR were ideal irradiation locations, as they had a large volume and high neutron flux.[20] Figure 11 shows irradiation of the Pu-238 production targets in the South Flux Trap.

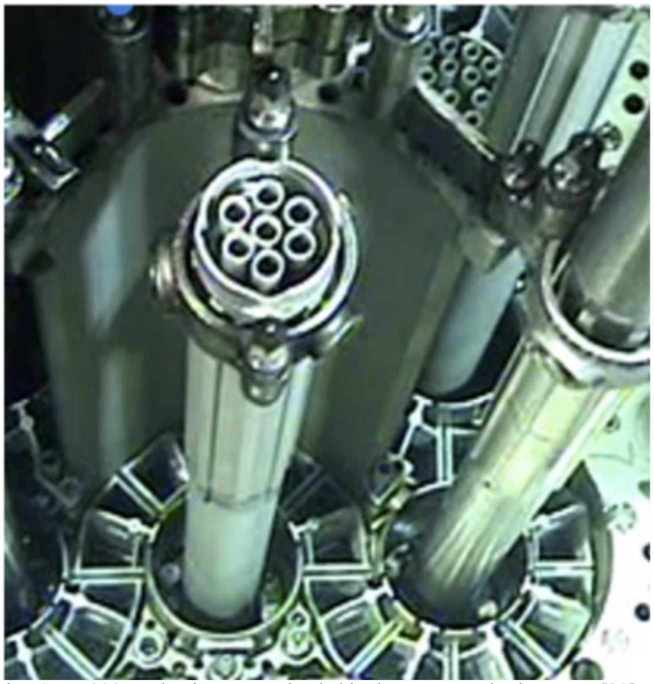

Fig.11. Pu-238 production targets loaded in the ATR South Flux Trap. [20]

D. Inner A qualifications

INL performed initial qualification work on the ATR inner A positions with HFIR Gen II targets. This effort initially ceased when staff were reassigned to the qualification of the South Flux Trap. Work was not restarted due to the ATR Core Internal Changeout (CIC) Outage, providing sufficient time to qualify the ATR Gen 1 target, increasing production, and yielding better utilization of the ATR.

E. Transition to ATR Gen 1 Targets

During the ATR CIC outage, INL worked to qualify a new Pu-238 production target design, designated HFIR Gen III/ATR Gen I. This target was an evolution of the HFIR Gen II targets, providing improvements to manufacturability while doubling the number of production targets that could be irradiated in each ATR position.

The HFIR Gen II targets sat at the centerline of the ATR core, leaving approximately a foot of unused core height at the top and bottom of the core. The ATR Gen I target stacked two targets similar to the HFIR Gen II targets on top of each other. This made full use of the height of the ATR core in each position and effectively doubled the amount of Pu-238 that was produced in each position. Additionally, having an irradiation target the entire length of the core minimized perturbations to the axial neutron flux profile, extending the life of ATR fuel and simplifying reactor operations.[21]

INL qualified all of the single inner core positions and flux traps for ATR Gen I targets. All future irradiation campaigns will use this style of target, simplifying logistics by tracking a single type of target.

VII. Transportation and Shipments

The transportation of targets between the ORNL and INL is an important part of the Pu-238 production program.

A. Target Shipments from ORNL to INL

Once targets were fabricated at ORNL, they needed to be shipped to INL for irradiation in the ATR. Initially, targets were shipped one per 85-gallon shipping drum. During production ramp up, this solution was adequate since only a few targets were needed every year with a maximum of eight targets shipped at a time. However, with the ramp up in irradiation at ATR, an alternative to this approach was needed. [21]

The solution to improving the target shipping process was to modify the shipping drums to hold five (5) production targets, rather than one (1). Drums holding 5 targets simplified and streamlined shipping between ORNL and INL by reducing the number of shipments between the two sites. It also simplified receipt at ATR by reducing the number of unloading that needed to occur. Figure 12 shows the top of the improved shipping drum.[21]

Fig. 12. A Pu-238 production target being loaded into a shipping container capable of holding 5 targets. Note the removed cap from the center position in the 5 o'clock position. [21]

B. BEA Research Reactor (BRR) Cask

The BRR cask was a key part of shipping irradiated Pu-238 targets from ATR to ORNL for processing. Figure 13 shows the major features of the BRR cask. The BRR is a type B shipping container that has been used since 2010 and was originally designed for the shipment of fuel elements from government and university research laboratories. The gross weight of the cask was 32,000 pounds (14,512 kg) and was certified for Department of Transportation (DOT) shipments.[22]

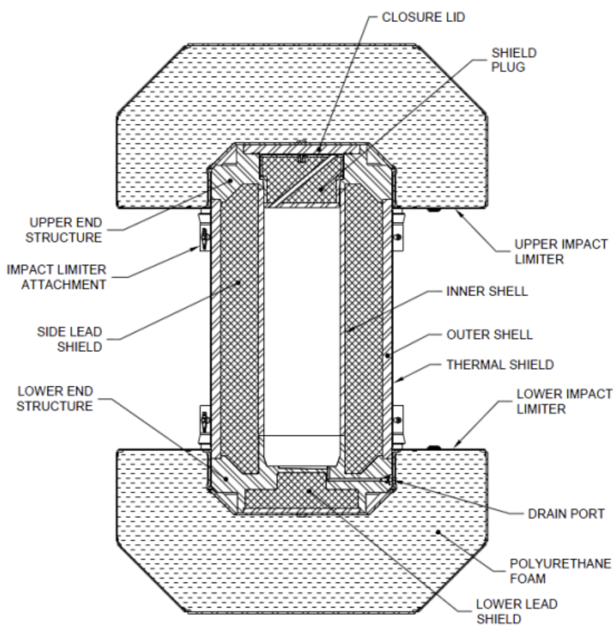

Fig. 13. BRR cask cross section. [23]

The BRR cask was certified as a type B shipping container through a variety of analytical and physical testing. Some of the analysis performed included thermal, structural, neutronic, radiation shielding, and materials compatibility for both nominal use and transportation, as well as accident scenarios such as drops, water immersion, criticality analysis, and crush analysis . Physical testing of the qualification included a half scale certification unit test (CTU) to establish expected physical damage to the container.

Figure 14 shows target racks that are used to hold Pu-238 production targets inside the BRR cask. Up to six of these target racks can be held in the BRR cask insert shown in Fig. 15, which is then loaded into the BRR cask. At the maximum six target racks filled and loaded in the BRR cask, a maximum of 96 irradiated production targets can be sent per cask shipment. Heat loading from each target was limited to a maximum of 5.32 W, which was defined in the Safety Analysis Review (SAR). The total plutonium mass was limited to 1000 g [22]

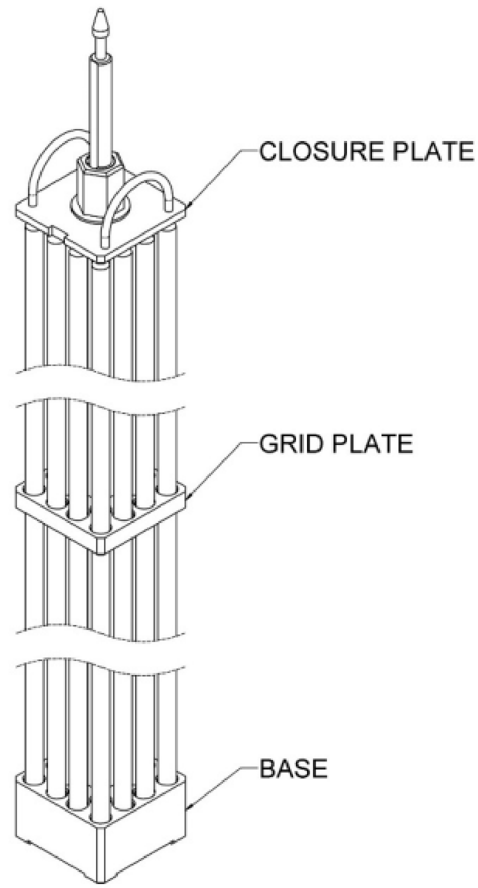

Fig. 14. BRR Pu-238 production target rack.[22]

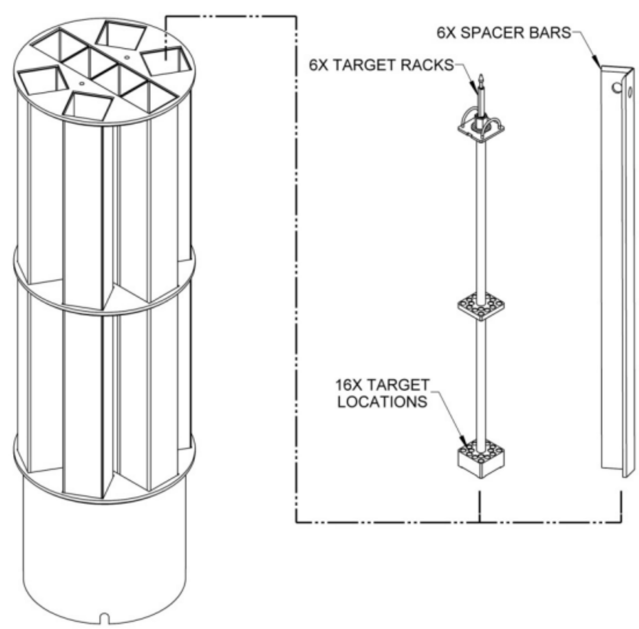

Fig. 15. BRR payload insert structure capable of holding six target racks.[22]

VIII. Future Activities

A. Increased Np Loadings

Efforts were initiated to determine the feasibility of increasing the NpO_2 loading beyond 20%-vol to increase the annual Pu-238 production rates. Loadings of 25-35%-vol could be possible based on scoping studies performed at ORNL (e.g., pellet pressing, neutronic, thermal-hydraulic). However, thermophysical and irradiation properties of increased loading pellets were not well understood at this time and therefore additional characterization activities would be required if these efforts were pursued. Figure 16 shows the production of Pu-238 in two irradiation cycles in 10 HFIR inner small VXFs (70 targets). As shown, an ~13-31% increase in production could be achieved by increasing the initial NpO_2 loading in the range previously discussed.

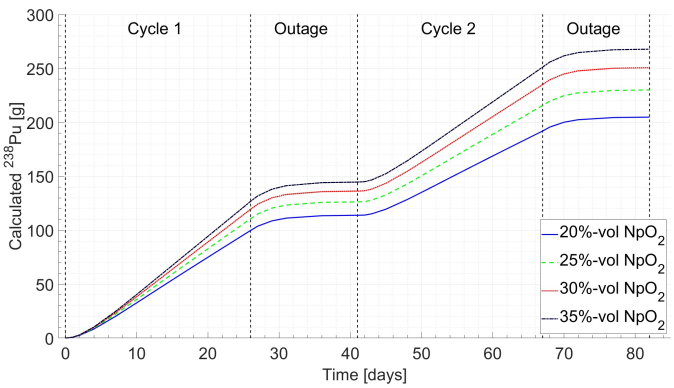

Fig. 16. Pu-238 production in 10 HFIR inner small VXFs over two.[24]

B. Alternative Target Designs

Once production has moved towards a stable process of target irradiation and separation, the ORNL and INL teams plan to evaluate additional improvements in target designs. These improvements may improve manufacturability of targets, reduce neutron absorption in target cladding, improve the assay of produced plutonium, and simplify the processing of irradiated targets. Alternative pellet materials may also be used to increase plutonium yields. These alternative designs would ensure that reactor operations and safety are maintained or improved, while increasing the throughput of the program.

C. Further Increase in Production Rates

Additional increases in production rates have been identified including reducing reactor downtime and decreasing reactor outage length, which will increase the total number of Megawatt-Days (MW-d) that the reactors operate. Further, modifications to equipment in the reactor cores could increase the number of irradiation positions.

IX. Conclusions

The plutonium-238 production program at ORNL and INL has successfully demonstrated the U.S. domestic production of plutonium-238 to be used in NASA deep space missions. The program is currently optimizing the production processes. Lessons learned and improvements in modeling, operations, manufacturing, and processing have helped the program move towards a production cadence. Future improvements in processes and irradiation techniques are expected to further increase Pu-238 to meet future NASA mission needs.

Acknowledgements

This work was funded by the U.S. National Aeronautics and Space Administration (NASA) Interagency Agreement # NNH19OB05A and U.S. Department of Energy (DOE) contract DE-AC05-00OR22725 and DE-AC07-05ID14517.

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