INL/CON-23-70850-Revision-0

Progress on Pu-238 Production at Idaho National Laboratory From February 2022 to December 2022

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Changing the World's Energy Future

William Spencer Green, Justin D Lower, Jill R Mitchell, Brittany Jean Grayson, Erik S Rosvall, Austen David Fradeneck, Joshua David Fishler, Ryan L Marlow, Mark A Hill, Andrew John Zillmer

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Idaho National Laboratory (INL) has continued to qualify irradiation positions in the Advanced Test Reactor (ATR) for Pu-238 production to support NASA deep space missions. Over the past year, INL qualified Np-237 targets for ATR's North East Flux Trap (NEFT), inner A, and H positions. Work has begun to requalify the South Flux Trap (SFT) and qualify the East Flux Trap (EFT) for the ATR GEN I target and is midway through the qualification process. This paper gives an overview of operational and technical activities from February 2022 to December 2022.

I. PROGRAM OVERVIEW

The purpose of this program is to contribute to the 1.5 kg per year constant rate production goal of Pu-238 in the United States by 2026, which is used as fuel in Radioisotope Power Systems that enable deep space NASA missions. To achieve this goal, the program has been working to qualify as many positions as possible for the insertion of the ATR GEN I. Currently the NEFT, inner-A and H positions have been qualified for the insertion of the ATR GEN I target, and work is in progress to qualify the SFT and EFT. The SFT and I-7 positions have been qualified for the High-Flux Isotope Reactor (HFIR) GEN II design, described in Reference 1. The SFT is currently being requalified for the newer ATR GEN I targets.

The ATR Gen I target (Fig 1) design was implemented to utilize the full height of the ATR core while maintaining a common target to be used in both the ATR and HFIR. The design consists of stacking the targets, reflected around the ATR core centerline, in each position which allows two targets to be used per position. A samarium pellet was also included at the nose of each target to reduce end effects. Utilizing the full height of the core will increase production by 40% to 50% as compared to a single target designed for the height of the HFIR reactor. This also allows the target to be processed at Oak Ridge National Laboratory (ORNL), whereas a target with the full height of the ATR core would not fit in the hot cells at ORNL.

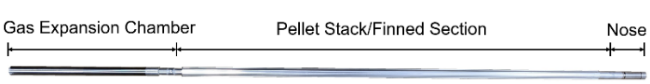

Fig 1. An ATR GEN I target.

II. Pu-238 PRODUCTION FOR UPCOMING CYCLES

II.A ATR CORE INTERNAL CHANGEOUT

ATR's Core Internals Changeout (CIC) refers to the changeout of reactor core components that are degraded by the high radiation environment. This degradation is caused by very high neutron radiation, beryllium/neutron reaction, thermal stresses, and normal wear and corrosion. For example, the beryllium/neutron reaction creates helium build-up leading to internal pressure that over time causes swelling and cracking in the beryllium reflector blocks. The degraded beryllium blocks can impede reactor operation due to stuck control rods, difficulty inserting/removing driver fuel, and possible fuel damage.

The CIC VI outage to replace ATR core internal components began on April 26, 2021, required approximately 11 months to complete or 332 days, and successfully ended on March 28, 2022. After completion of the CIC, nuclear testing was performed to ensure correct core operating parameters were initiated. Nuclear testing was successfully completed in November of 2023. It is expected that this CIC evolution will provide continued reactor operations for at least 10 years.

II.B UPCOMING ATR CYCLE 171A IRRADIATION

Cycle 171A is a 60-day cycle anticipated to start in early 2023. This will be the first cycle where the newly designed ATR GEN I target will be inserted into the ATR. Due to target availability, INL will only insert 57 targets in the ATR. Of the 57 targets, 46 ATR GEN I targets will be inserted in the NEFT, two ATR GEN I targets in the inner A position, two ATR GEN I targets in the H position, and seven HFIR GEN II targets in the SFT.

II.C UPCOMING CYCLE 171B

Cycle 171B is a 60-day cycle that was recently added to the ATR schedule and is estimated to begin in Spring 2023. With the addition of this cycle, there are approximately 50 positions available which would accommodate 100 targets. However, due to target availability, only 46 targets will be supplied to fill the NEFT. This will produce an estimated 200 g of heat source material. INL expects to have the qualification for insertion of the ATR GEN I targets into the SFT completed in time for this cycle.

II.D UPCOMING CYCLE 172A

Cycle 172A is a 7-day operating cycle instead of the typical 60-day cycle. Therefore, targets will not be inserted in the cycle because the short cycle length results in a low production amount of Pu-238.

II.E UPCOMING CYCLE 173A IRRADIATION

The 173A cycle is a 60-day cycle estimated to start in Fall 2023. Analysis to increase the percentage of Np-237 in the target from 20% to 30% will be completed. There are approximately 40 positions available which would accommodate 80 targets. This could produce approximately 300 g of heat source material depending on position availability, target availability, and position qualification.

II.F UPCOMING CYCLE 173B IRRADIATION

The 173B cycle is a 60-day cycle estimated to start in early 2024. There are approximately 17 inner core positions available which would accommodate 34 targets and could produce approximately 120 g of heat source material. Analysis for the I positions will be completed to support this cycle. This may include the medium and large I positions depending upon the difficulty of completing a bounding analysis. There are four large I positions available that could include approximately 23 irradiation locations each and nine medium I positions that would include seven irradiation locations each . This would allow for a maximum of 155 positions that would accommodate 310 targets. Ultimately, this could produce 700 g of heat source material. Also note that these positions have an assay of approximately 94% which is higher than the inner core positions. Producing this amount of material would require approximately six cycles which span approximately 2 years.

III. QUALIFICATION OF ATR GEN I TARGET IN NEFT, A, AND H POSITIONS (Maybe SFT)

III.A. Mechanical Design

The basket design for the NEFT, Inner-A, and H positions utilizes features from existing basket designs. The main basket body is made from extruding thin-walled aluminum tube to create ridge features along the longitudinal axis that help keep the basket vertically centered within the flux trap. The head of each basket is designed to be used with hand tools to remove and manipulate each basket. The nose of the basket has been redesigned to allow for a stronger fillet weld while still allowing for the optimal flow through the basket.

Each basket allows for two targets to be stacked ‘nose to nose’. This allows up to 46 targets to be irradiated in the NEFT.

III.B. Neutronics Analysis

The primary neutronics code used in qualifying the PFS ATR Gen I target design for the inner-A and H positions was MC21 (Ref. 3). MC21 and its associated API, PUMA are part of the common Monte Carlo design tool, CMCDT, provided for use in the ATR. MC21 was used to calculate neutron and photon heat generation rates during irradiation, fission gas production, Pu-238 production, fission density, and experiment reactivity. Due to the lack of development of the ATR model in MC21 for decay heat and dose consequence at the time calculations were performed, MCNP5 coupled with ORIGEN2 (MOPY) was used to calculate the decay heat and the dose consequence instead. MOPY works by using reaction rates and fluxes calculated by MCNP in an ORIGEN2 depletion on the pertinent materials. MC21 used ENDF-VIII.0 cross sections for all pertinent materials, while MCNP5 primarily used ENDF-VII.0 cross sections with one exception being the use of TENDL-2017 for the Np-236m isotope.

The MC21 model of the neptunium target included 52 axial and five radial discretization’s per target. An abbreviated PUMA model of a PFS target is shown in Fig 2. The PFS Gen I stack up includes two targets nose-to-nose at 5.25 in. and 0.125 in. above core centerline for the inner-A and H positions, respectively. Existing hardware in the inner-A positions is what causes the targets in the inner-A positions to be raised 5.25 in. above core centerline. The MCNP model utilized 40 axial and one radial section per target.

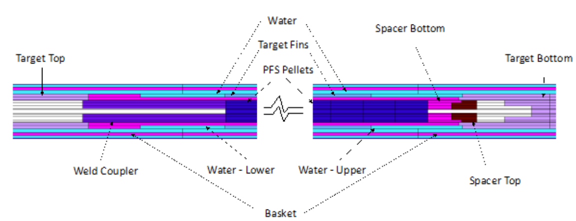

Fig 2. An abbreviated PUMA model of a PFS target.

Neutronics analysis for qualification of the PFS ATR Gen I target in the inner-A and H positions of the ATR was completed up to a maximum central lobe power of 25 MW for a 65-day irradiation period. The nominal cycle length that the ATR targets will be irradiated for is 60 days. Qualifying the targets to a total of 65 days and to a maximum power of 25 MW allows for the targets to be qualified for multiple cycles under one analysis. At the nominal 60 EFPD, the peak production positions are in the lower targets of A6 and H12. Pu-238 production rate estimates for the peak positions were calculated to be 2.43E-03 g/MWd and 1.64E-03 g/MWd for the lower and upper target in A6, respectively, and 2.26E-03 g/MWd and 2.14E-03 g/MWd for the lower and upper target in H12, respectively. The inner-A and H positions were able to achieve a higher assay than the NEFT targets with the average assay being approximately 92% for the inner-A and H positions compared to the overall peak assay in the NEFT of approximately 88%. For reference, NASA RPS missions typically use heat source materials with a minimum assay of 82.5%.

Work is currently underway to qualify the SFT position for the ATR Gen I targets. The qualification is performed using the newly developed automation tool, MCNP ORIGEN-S Activation Automation (MOAA). MOAA is a python-based automation tool that passes data between MCNP and the SCALE modules COUPLE, ORIGEN-S, and OPUS for depletion and activation. MOAA also automates the calculation of several results of interest. The SFT Gen-I analysis is being completed using ENDF-VIII.0 cross sections for all experimental materials.

The MCNP model used for the SFT qualification utilized 40 axial regions and one radial section for each neptunium target. Like the NEFT and A and H positions, the SFT Gen I stack up will have two targets stacked nose-to-nose and elevated slightly above the core centerline to accommodate existing hardware.

Like the previous neutronic analysis, the SFT qualification assumes a 65 EFPD cycle as well as the projected peak power for the south lobe of ATR. Preliminary analysis of the SFT is anticipated to produce 61.36 g of Pu-238, with an average peak Pu-238 Assay of 86.8% after 65 EFPDs of irradiation.

Qualification is also being done on the EFT position of ATR simultaneously. The SFT analysis will envelop the results for the EFT as the EFT and SFT have identical geometric configurations and the South Quadrant of ATR is typically operated at a higher power than the East Quadrant, which results in a higher flux and higher heating rate in the SFT as compared to the EFT.

III.C. Thermal Analysis

Thermal-hydraulic analyses were performed to support irradiation in the A and H positions, as well as the NEFT. The qualification of the ATR Gen-1 targets in the NEFT included a parametric analysis of requisite safety scenarios under a wide range of thermal-hydraulic conditions. The results of the analysis provided system response surfaces for critical safety quantities such as DNBR, FIR and peak component temperature (See Fig 3)

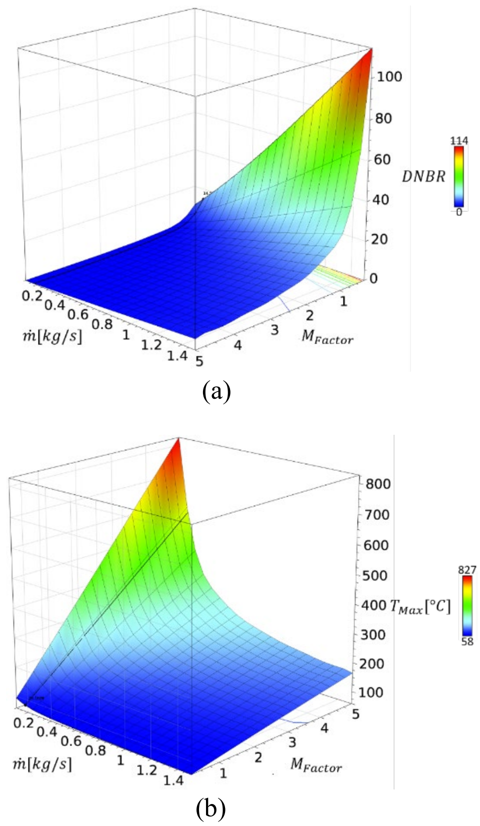

Fig 3. Response surfaces for the (a) Minimum DNBR and (b) maximum component temperature of the ATR Gen-1 target under various thermal-hydraulic conditions.

The response surfaces were used to facilitate the qualification of the A and H positions, by verifying that the mass flow rate and total heat rate for the most limiting positions were within the analyzed bounds. The heat generation rates for the pellet stacks in the most thermally limiting positions (A5 and H10 positions) are shown in Fig. 4.

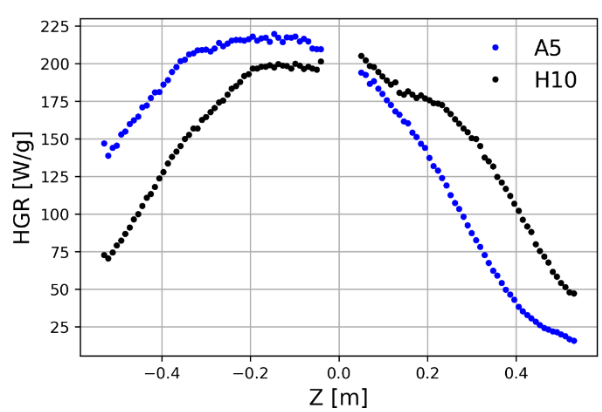

Fig 4. Axial heat generation rates of the NpO2-Al cermet pellets at the end of a 65-day cycle for the A5 and H10 positions.

Flow rates for all analyses were provided using RELAP5-3D (v.4.4.2) while the subsequent thermal analyses were performed using ABAQUS (v.2018hf3) and STAR-CCM+ (v.16.06.010-R8).

While the NEFT featured similar analyses as the other irradiation locations, to support discharge of the experiment from the NEFT, a CFD analysis was performed to determine the number and configuration of targets to be removed in order to avoid disruption to the operational cadence. The analytical approach used features a simplified geometry, specifically a 2-D symmetric cross section at the axial center, as shown in Fig 5.

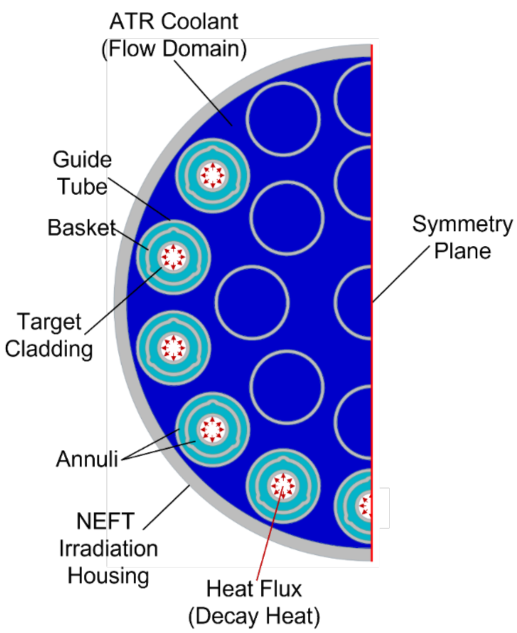

Fig 5. Schematic of the CFD model used for horizontal analyses of the PFS experiment in the NEFT housing.

The coolant was modeled as a solid between the cladding and baskets, as well as in between the baskets and guide tubes, thereby reducing analytical complexity exclusively to conduction in these regions. An effective thermal conductivity, accounting for heat transfer enhancement via free convection in these regions was implemented, as shown in Eqs.1-3. These were implemented via look-up table in STAR-CCM+.

$$\frac{k_{Eff}}{k} = 0.386\left(\frac{Pr}{0.861 + Pr}\right)^{1/4} Ra_c^{1/4} \tag{1}$$

$$L_C = \frac{2\left[ln\left({}^{r_o}/_{r_i}\right)\right]^{3/4}}{\left(r_i^{-3/5} + r_o^{-3/5}\right)^{5/3}} \tag{2}$$

$$Ra_C = \frac{g\beta(T_w - T_\infty)L_C^3\rho^2 C_p}{k\mu} \tag{3}$$

Equation 4 presents the Nusselt number correlation used to estimate the heat transfer from the guide tube surface to the flow domain.

$$Nu_{Ra} = \left[0.6 + \frac{0.387Ra^{1/6}}{\left[1 + \left({}^{0.559}/_{Pr}\right)^{9/16}\right]^{8/27}}\right]^2 \tag{4}$$

The results of this analysis predict a maximal coolant temperature of 85 °C (185 °F), with the general trend that the maximal coolant temperatures are inversely related to the gravitational field; this aligns with expectation.

A supporting calculation was performed using RELAP5-3D. Certain conservatisms were not implemented in the RELAP5-3D simulation, as exploratory calculations indicated a mixed convective-conductive heat transfer modality between the basket and guide tube. Fig 6 presents the nodalization implemented in RELAP5-3D.

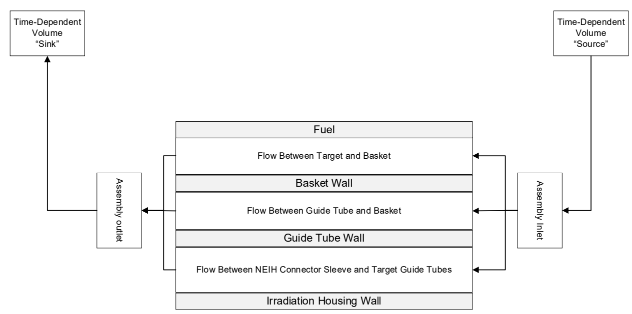

Fig 6. Schematic showing RELAP5-3D component and heat structure orientation.

Calculating a decay heat curve appropriate for the time scale and implementing the Nusselt number correlation presented in Equation (4) via ‘htc-temp’ tables in RELAP5-3D, resulted in the temperature traces shown in Fig 7. The temperature curves depict a maximum temperature within the region between the target and basket of 76 °C (168 °F), well within expectation, given the increased heat transfer implemented in the RELAP5-3D model.

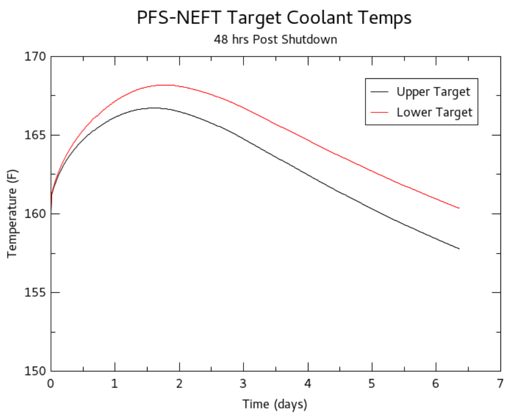

Fig 7. Upper and lower target coolant temperatures calculated in RELAP5-3D.

III.D. Structural Analysis

The purpose of the structural safety analysis was to evaluate the target and its associated hardware under various potential loading scenarios to ensure the safety of operational personnel and the public. The loadings considered in this evaluation, while within the ATR, included the following: internal pressure within the target due to the release of fission gas, external pressure, external pressure differential acting on the length of the assembly, pressure and skin friction drag forces due to coolant flow velocities, flow induced vibrations, thermal loads, and cyclical loads. The decision for which loading scenarios were to be evaluated in the structural analysis was based upon the probability of the event occurring and the desired state of the structural components after each event. These events include normal reactor operation, a flow coastdown event due to loss of commercial power, a reactivity insertion accident for in-pile tube voiding, an over-pressurization incident, and a loss of coolant accident. Events with a low probability of occurrence and situations where the consequence of a pressure boundary losing its integrity meets the safety limits defined by INL's Safety Analysis Report (SAR) were excluded from the structural evaluation. Other loadings, such as handling loads from transferring components to and from the reactor, were also considered. These include an accidental drop of the target through water from a height of 45 ft. which could occur at the deepest portion of the ATR canal.

For the analysis to be useful for multiple positions within the ATR, limits for temperature (peak and gradient), pressure (internal and external), and coolant flow velocities were established. The response of each structural component (i.e., stress, strain, deformation) under these limiting conditions was determined using, where simplifications could be made, hand calculations, or, where simplifications could not be made, the finite element software Abaqus. These responses were compared to acceptance criteria. For the non-pressure retaining components, this criterion was typically the yield strength of the material at temperature. Due to the potential of fission gas release, the target was treated as a pressure vessel. Acceptance criteria limits defined in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code were used. Though other acceptance criteria could be used, this specific code was used because it provides a nationally accepted design/analysis approach which INL has used and adapted to various nuclear experiments. Based on the low internal pressure of the target, the requirements of ASME Section III, Class 3 vessels were used as a guide. The limits of temperature, pressure, and coolant velocities, based on the acceptance criteria, were compared to those calculated in the thermal analysis and from the design specification of the ATR. For the NEFT, inner-A and H positions, these values were within the calculated limits and each structural component was considered to meet the safety requirements and thus allowed into the ATR.

III.E ATR Safety Considerations

Along with existing Experiment Safety Analysis (ESA) already developed for both the PFS experiment in the I-7 and SFT positions, an additional ESA was developed for the ATR Gen I target irradiations in the NEFT, inner-A, and H positions. This ESA utilizes the analyses performed, as previously discussed (neutronics, thermal, and structural), to demonstrate that the new PFS Gen I targets can be irradiated in the ATR in compliance with the technical safety requirements and the approved authorization basis, established by ATR's Safety Analysis Report. The Gen I ESA was also developed and authorized under an ATR Complex procedure that addresses experiment receipt, reactor loading, irradiation, discharge, storage, preparing for shipping from ATR, and waste disposal. The PFS ATR Gen I ESA demonstrates that operation of the PFS experiments are in accordance with the restrictions identified in the ESA and within the authorization basis of the ATR.

IV. INCREASING NP CONCENTRATION IN TARGETS

IV.A. Preliminary Neutronic Analysis

Work has begun on utilizing MC21 to analyze a potential increase in the Np-237 concentration from 20 vol% to 30 vol%. Preliminary findings indicate increasing the Np concentration to 30 volume percent would result in a potential Pu-238 production increase in the range of 20-30%.

V. SHIPMENTS

INL has received four shipments of targets from ORNL, consisting of 85 gal drums. Originally shipments have been made by inserting one target per drum. This allowed for a maximum of eight targets to be shipped to INL at a time. An updated design to the drums has allowed five targets to be shipped per drum with a maximum of 40 targets per shipment.

The updated design was completed to support the need to ship and receive approximately 200 targets per year. This design uses five 4 in. stainless steel pipes supported in the 85 gal drums, as shown in Fig 8.

Fig 8. An ATR GEN I target being loaded into the shipping container.

VI. CONCLUSIONS

INL has successfully completed qualification for the ATR GEN I targets in the NEFT, inner A, and H positions. This included using a CFD and Relap analysis to support unloading the NEFT within operational time limits. Efforts on requalifying the SFT and qualifying the EFT for the ATR GEN I target are in progress and should be complete in March 2023. With the qualification of more positions and increased efficiency in shipments, INL is on track to meet production goals by 2025.

ACKNOWLEDGMENTS

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