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Progress on Pu-238 Production at Idaho National Laboratory From March 2021 to February 2022

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Changing the World's Energy Future

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Idaho National Laboratory has completed irradiation of Np-237 targets in the Advanced Test Reactor's (ATR) South Flux Trap (SFT) and I-7 positions. Idaho National Laboratory (INL) also progressed qualification of new ATR Gen I Np-237 targets for the North East Flux Trap (NEFT), inner A, and H positions. This paper gives an overview of operational and technical activities from March 2021 to February 2022.

I. PROGRAM OVERVIEW

The goal of this program is to contribute to constant rate production of Pu-238 in the United States, which is used as fuel in Radioisotope Power Systems that enable certain NASA missions. To achieve this objective, the INL team has implemented the strategy to qualify as many positions as feasible to maximize the number of targets that can be inserted in ATR and Oak Ridge National Laboratory (ORNL) has designed a new target for ATR called the ATR Gen I target. Figure 1 shows the various irradiation positions in ATR.

The ATR Gen I target was designed to allow the full use of the height of the ATR core. This is accomplished by stacking the targets, reflected around the ATR core centerline, in each position which allows 2 targets to be used per position. Utilizing the full height of the core will increase production by forty to fifty percent. It also allows the target to be processed at ORNL whereas a target the full height of the ATR core would be too tall to fit in their hot cells.

Currently the Plutonium Fuel Supply (PFS) program has qualified the I-7 and SFT position for the High Flux Isotope Reactor (HFIR) Gen II target Design, described elsewhere (Ref. 1). With the new ATR Gen I targets, the PFS program has completed qualification of the NEFT. The NEFT has 23 positions which will accommodate 46 ATR Gen I targets. Qualification has also started on the A and H positions, and they are in the final stages of completion. To help increase the efficiency of qualifying positions, bounding analyses have been created and will be used for future qualifications. This simplifies new analysis by allowing a comparison to be performed rather than performing new analyses for each position.

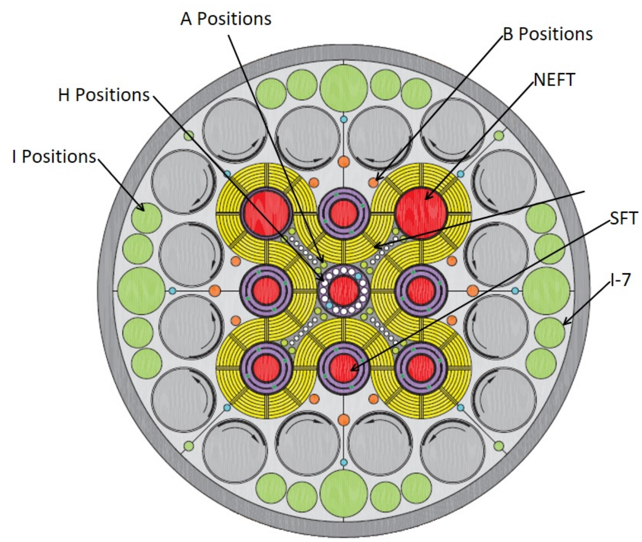

Fig 1. Section view of ATR showing various irradiation positions

II. PFS PRODUCTION FOR UPCOMING CYCLES

II.A CYCLE 169A IRRADIATION

With the completion of ATR cycle 169A in April 2021, the INL team has successfully irradiated 14 targets that now contain Pu-238. Seven targets were irradiated in the SFT and seven in the I-7 position. The targets in the I-7 position were irradiated for five sixty-day cycles and one seven-day palm cycle. The seven targets in the SFT were irradiated for one sixty-day cycle. The irradiated targets are currently in the ATR canal cooling for 6 months before they will be shipped to ORNL for processing.

III.B ATR CORE INTERNAL CHANGEOUT

The ATR Core Internal Changeout (CIC) is the periodic overhaul of key internal components in the ATR and was scheduled for about 9 months from Spring 2021 to mid-January 2022. This effort will help to restore performance, reliability, and efficiency. One change that will reduce manpower in moving experiments is the center flux trap collar was redesigned to eliminate unnecessary handling of targets in the H positions when changing the core loading in the inner A positions reducing manpower and project costs.

II.C UPCOMING ATR CYCLE 171A IRRADIATION

The 171A cycle is a 60-day cycle estimated to start in May 2022. This will be the first cycle where the newly design ATR GEN I target will be inserted into the ATR. Up to 95 targets may be irradiated in ATR, depending on target availability and position qualification status.

II.D UPCOMING CYCLE 172A

Cycle 171A is only a 7-day operating cycle instead of the typical 60-day cycle. Therefore, targets will not be inserted in the cycle. This is because the cycle is not long enough to produce enough heat source material and the ATR Gen I targets have not been qualified to be inserted during a shortened cycle.

II.E UPCOMING CYCLE 173A IRRADIATION

The 173A cycle is a 60-day cycle estimated to start late September 2022. Currently planned analysis should be completed on qualification of targets for the I positions. However, if other positions with higher flux become available for the cycle, then analysis may be performed for those positions instead of the I position. Up to 116 targets may be irradiated in cycle 173A, depending on position availability, target availability, and position qualification.

III. QUALIFICATION OF ATR GEN I TARGET IN ATR

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 have ridge features that help keep the basket vertically centered with in 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

Multiple neutronic analyses were completed to satisfy the safety requirements for irradiating the PFS ATR Gen I target in the NEFT of ATR. Nominal values of ATR operating conditions were used to model the ATR Gen I targets in MCNP. To properly capture the axially dependent behavior of the neptunium pellet material in each target, each pellet stack was divided into forty axial segments.

MCNP5, a general-purpose Monte Carlo N-Particle transport code, was used to calculate the pertinent neutron and photon heat generation rates within all experiment materials. MCNP was also used to calculate the neutron fluxes and reaction rates for pertinent reactions on the neptunium pellet material and this information was then passed into ORIGEN2 to deplete the neptunium pellet material. The ENDF/B-VII.0 cross section library that comes with MCNP was used along with the neptunium-236m cross section library obtained from TENDL-2017. The standard ATR cross section library was used for ORIGEN2 along with MCNP-calculated replacement cross sections. A python-based code, MCNP to ORIGEN2 in Python (MOPY), was used to extract the fluxes and reaction rates calculated from MCNP

In addition to the calculations listed above and to further demonstrate reactivity safety compliance the reactivity worth of the unirradiated PFS experiment and the end of cycle PFS experiment were calculated using the 19-plate MCNP model of ATR.

After 60 days of irradiation the Pu-238 average assay was calculated for each target located in the NEFT. The peak average Pu-238 assay is 88.53%. Using MOPY it is estimated that approximatly144 grams of Pu-238 will be produced in the NEFT of ATR during cycle 171A.

III.C. Thermal Analysis

Several thermal/hydraulic analyses were performed using the finite element software package ABAQUS (v.2018hf3) in conjunction with the system code RELAP5 to ensure that the safety requirements for irradiation of the PFS ATR Gen I target in the NEFT were met. Structural and fuel heating rates for each model were provided from the neutronics analysis. A safety factor multiplier of 1.26 was applied to all heat rates to account for instrumentation and lobe power uncertainties. The thermal results show that there is adequate coolant flow to avoid flow instabilities and local boiling under all safety scenarios. In all cases, the maximum temperatures (located in the aluminum finned extrusion) were well below the respective component melting point, while the maximum pressures (including the effects of fission gas release) were within the failure limits of the target.

A generalized safety analysis of the target was also performed to support thermal/hydraulic qualifications for other positions in ATR. The inputs for the heating rate and flow rate were parameterized using the software package HEEDS (v.2021.1) which automated the writing and submission of the ABAQUS input files. The inputs were varied over a wide enough range to encompass as many thermal/hydraulic conditions as possible. The minimum departure from nucleate boiling ratio (DNBR), flow instability ratio (FIR) and maximum temperature and subsequent internal pressure were extracted from each simulation as quantities of interest. These quantities were used to generate a system response surface (See 2) which provides a simple interactive method for determining safety parameters for a wide range of thermal/hydraulic conditions. The response surface was used to create a lookup table of minimum required flow rates for a given total experiment heat rate to facilitate qualification of the target in different positions in ATR.

The RELAP5 analysis utilized scaling to represent coolant flow and heat transfer areas to accommodate the expanded target loading, and to provide flexibility for future loading patterns. This approach significantly reduced the time and effort needed to develop and qualify a functional model. Further, it maintains similitude with respect to key thermal hydraulic parameters such as pressure drop, thermal energy transfer, and bulk mass continuity. Figure 3 presents the schematic used to represent the RELAP5 nominal and loss of coolant accident (LOCA) analyses.

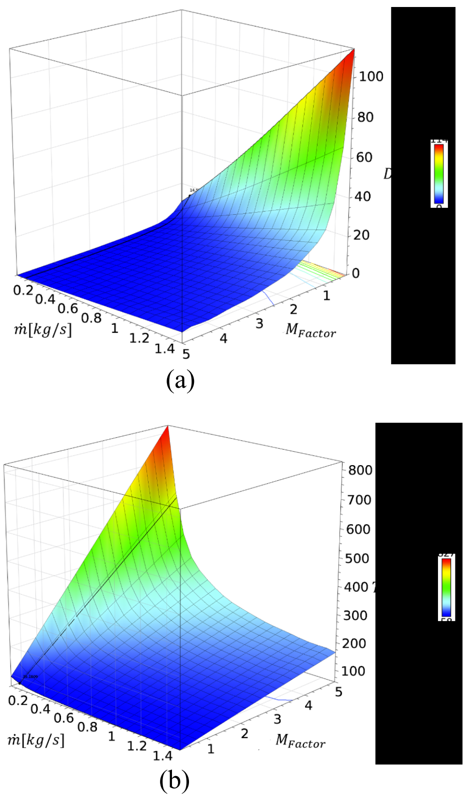

Fig. 2 Response surfaces for the (a) Minimum DNBR and (b) maximum component temperature based on the nominal/flow coast down (FCD) steady-state parametric analysis.

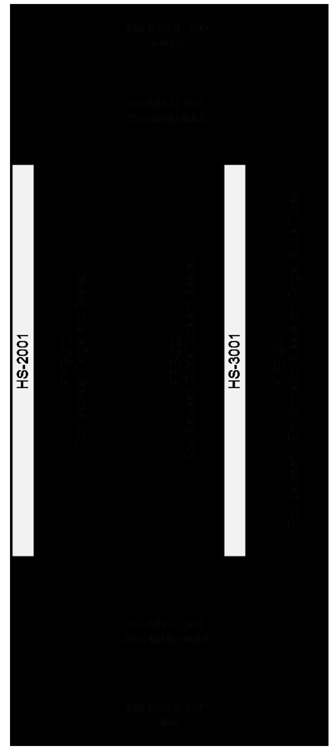

Fig 3. Model schematic used in the RELAP5 analyses.

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, overpressure, and a loss of coolant accident. Events with low probability of occurrence and when 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, etc.) under these limiting conditions was calculated using, where simplifications could be made, hand calculations or, where simplifications could not be made, using 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 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 using these acceptance criteria were compared to those calculated in the thermal analysis and from the design specification of the ATR. For the NEFT, these values were within these calculated limits and each structural component was considered to meet the safety requirements and 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 is being developed for the ATR Gen I target irradiations in the NEFT, inner A, and H positions. This ESA will utilize analysis performed, as previously discussed (neutronics, thermal, and structural), to demonstrate these new PFS Gen I targets can be irradiated in the ATR in compliance with the requirements of technical safety requirements and the approved authorization basis established by ATR′s Safety Analysis Report. The Gen I ESA will also be 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 Gen I ESA must demonstrate 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. TRANSPORTATION OF IRRADIATED TARGETS IN THE BATELLE RESEARCH REACTOR (BRR) CASK

IV.A. Comparison of Loading to SARP

The BRR dry storage cask has been designed to hold a maximum of 96 PFS targets. The shielding analysis performed for the BRR dry storage cask is documented in the SARP and assumed that all 96 positions were filled with a ″generic″ PFS target design. This generic PFS target design was intended to cover both the HFIR Gen II and ATR Gen I PFS target designs.

The activation analysis performed for the NpO_2-Al cermet material in the generic PFS target design assumed an irradiation period of 65 days in the inner ″A″ positions of the ATR followed by 180 days of decay. The activation calculations were performed with the ORIGIN 2 computer code which simulated the mass of NpO_2-Al in 1 inch of stack height. The activity calculated by the ORIGIN 2 code was performed on a per inch of NpO_2-Al stack height basis.

The HFIR Gen II and the ATR Gen I target designs both have a nominal NpO_2-Al stack length equal to 19.5 inches. The shielding analysis perform for the BRR dry storage cask assumed that each generic PFS target contains 23 inches of NpO_2-Al. The total activity assumed for the BRR dry storage cask shielding analysis was based upon 2208 (=96*23) inches of activated NpO_2-Al. The dose rate results for the BRR dry storage cask loaded with the generic target design are provide in Table 5.7-1 of the SARP. The dose rate results in Table 5.7-1 are provided for both the Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC). Table 5.7-1 also provides the 10 CFR 71.47 (b) radiation dose rate limits. The calculate dose rates for the BRR dry storage cask loaded with the generic PFS target design are significantly below the 10CFR 71.47 (b) limits for all the NCT and HAC requirements.

The BRR shielding analysis was recently updated at INL based upon the ATR Gen I specific target design. The recent INL BRR shielding analysis was performed with the MCNP 6.2 computer code and employed the ENDF/B-VII cross section library for neutron transport and the ENDF/B-VI cross section library for photon transport. This is the exact same methodology which was employed for the shielding analysis in the Safety Analysis Report for the BRR dry storage cask. The ATR Gen I specific target design was simulated to be irradiated in the NEFT of the ATR for 65 days and allowed to decay for 180 days. The INL calculated dose rates for the BRR dry storage cask fully loaded with the ATR Gen I target design are also significantly less than the 10CFR 71.47 (b) limits.

IV.B Safety Considerations

The BRR cask is designed, fabricated, and tested to regulations for safe transportation of radioactive material as documented in the various safety analyses supporting BRR shipments. The safety analyses include the safety requirements for shipping the BRR, with these types of targets from the ATR canal area at INL. Examples of these safety requirements include decay/cooling times prior to loading the cask, strict adherence to operating procedures during handling and loading of the targets, and adherence to transport plans once the cask has left the INL.

Specific to the operating procedures during handling and loading of the targets, examples of existing safety requirements include lifting height restrictions and specific hardware associated with the closure lid, shield plug, and retaining bar. These portions of the cask, including the specific hardware used, have been analyzed in various drop analyses, therefore strict adherence is required to these analyzed conditions.

V. CONCLUSIONS

INL has qualified the ATR Gen I target for irradiation in the NEFT, inner A, and H positions in the ATR. The first irradiations of the ATR Gen I target design are expected to begin in Spring 2022, corresponding to ATR Cycle 171A, upon completion of ATR CIC. Predicted decay heat and source terms from irradiated targets were compared against the BRR SAR, and no issues were identified for shipping irradiated targets to ORNL for processing if the targets are allowed to cool for 6 months after reactor discharge. INL is on track to meet or exceed heat source production goals by 2025.

ACKNOWLEDGMENTS

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