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DESIGN AND PROTOYPING OF A FISSILE-BEARING CHLORIDE SALT IRRADIATION EXPERIMENT

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Changing the World's Energy Future

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Limited data exists on the properties of irradiated fueled chloride-based salts for use in advanced nuclear reactors. The design and prototyping of a salt irradiation experiment is underway. Mechanical design will integrate the experiment into existing assemblies in TRIGA type cores. Neutronics modeling demonstrated how targeted power densities can be reached and provide insight on radionuclide source term generation within the experiment. Thermal-hydraulic and computational fluid dynamics (CFD) analysis indicated that temperature ranges in the salt can satisfy both safety and programmatic requirements. Initial prototyping tests provided confidence in the ability to extract the salt post-irradiation, and in the heater performance. Future planned prototyping will provide an integral assessment of the proposed design prior to insertion in the reactor.

I. INTRODUCTION

Molten Salt Reactors (MSR) can potentially upend the nuclear industry by, among other things, providing a promising path to a fuel cycle with near-zero nuclear waste. As a dozen MSR developers in the U.S. work toward an aggressive commercialization timeline, many of their fueled-salts—notably, chloride-based compositions—have never been irradiated. Licensing and operating these reactors requires an understanding of (1) the source term, radiation chemistry, and gas generation; (2) unanticipated irradiation-induced corrosion effects; and (3) the impact of burnup on thermophysical properties, all of which can be deduced through irradiation testing. An irradiation experiment to fill this gap is proposed at the Neutron Radiography Reactor (NRAD) at Idaho National Laboratory (INL).

The proposed Molten-salt Research Temperature-controlled Irradiation (MRTI) consists of a salt-containing capsule that is internally heated. The experiment must rely on resistive heating to melt the salt before irradiation (to avoid the impact of radiolysis) and once the reactor is turned on and fission reactions commence in the salt, the resistive power can then be reduced. Salt-immersed thermocouples coupled to a controller allow for the heater power to be adjusted as needed to meet experimental objectives. The bulk of the experimental results will be achieved through Post-Irradiation Examination (PIE). At which point a range of different measurements are anticipated to assess the salt/plenum/wall composition, the capsule corrosion rate, and the evolution of salt properties. An out-of-pile 'sister experiment' is planned to baseline these various PIE measurements.

The conceptual design analysis for MRTI was previously summarized in Ref. 1. Here, an overview of the final design (with an emphasis on mechanical, neutronic, and thermal characteristics) is provided alongside an overview of prototyping activities.

II. DESIGN OVERVIEW

II. A. Mechanical Design

To ensure integration into the NRAD reactor, the MRTI mechanical design leveraged the typical TRIGA fuel cluster design. The stainless-steel experiment pin will replace one of four fuel pins in a typical cluster fitting and top bail assembly, and this cluster is slated to be in the NRAD F1 outer position. Inside of the pin, an inner salt-containing capsule will hold the salt during irradiation and store fission gas products in a plenum region. The material of the capsule was chosen to be Inconel 625 (I625) for its high temperature strength, corrosion resistance, and ample supply chain. Only this material will be in contact with the salt in order to prevent any galvanic corrosion effects. The midplane of melted salt will be placed at NRAD fuel midplane via a graphite bottom spacer. Between the capsule and the bottom graphite moderator, and insulative wafer will minimize axial heat transfer. Machined standoffs on the outer wall of the capsule will create a 0.03" radial gas gap between the capsule and the outer pin wall. Experiment pin internals will be held in place axially by stainless steel compression springs.

An 800W immersion heater will be used to melt the powder salt in the reactor position prior to irradiation and to provide supplemental heat input during the experiment. An integrated thermocouple in the heater will serve as the over-temperature shut-off sensor. Because the heater sheath is Incoloy 800, the heater will be placed inside of a thin wall thermowell made of I625. Multiple other Type-N I625-sheathed thermocouples will be placed at varying axial positions to provide continuous data on the average temperature in the salt. Changes in plenum pressure will be monitored via an optical fiber pressure sensor assembly fabricated at INL's High Temperature Test Laboratory (HTTL). The heater thermowell, thermocouples, and pressure sensor will be induction brazed through the top of the inner capsule. All leadouts will be routed out of the pin by a Conax compression fitting with a Lava seal. Outside of

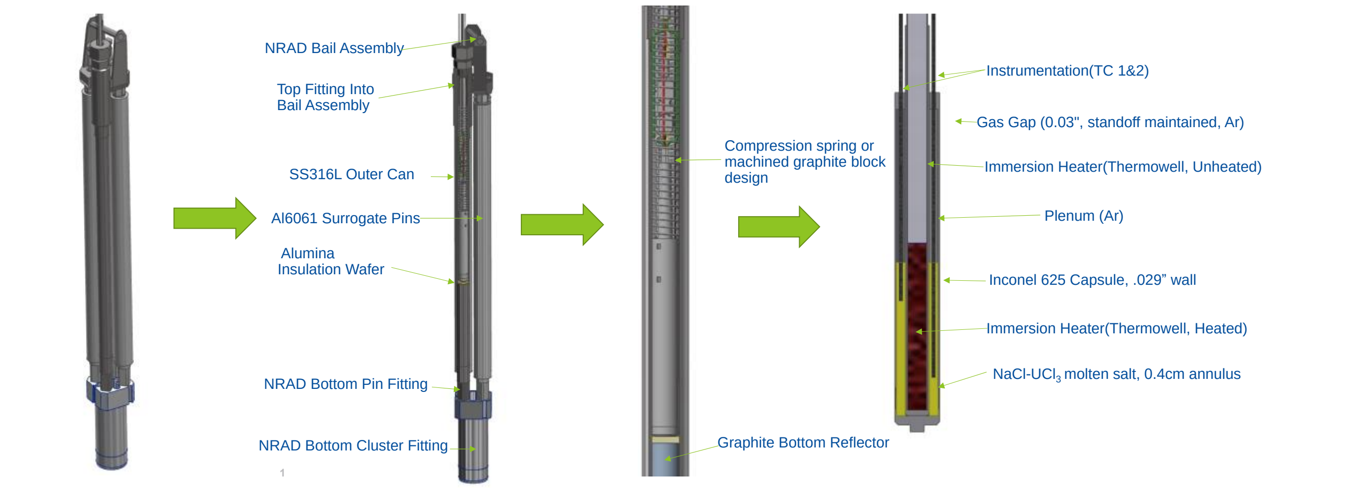

Fig. 1. Overview of the mechanical design.

the experiment pin, a flux wire package in the center of the cluster assembly will activate and provide knowledge of the axial flux distribution experienced by the salt. An overview of the experiment is provided in Fig. 1.

II.B. Neutronic Evaluation

The neutron transport code MCNP6.1 was used for the neutronic analysis [2]. Building on the analysis in Ref .1, the experiment was explicitly modeled inside the NRAD core geometry as shown in Fig. 2. While different dimensions and materials were considered in Ref. 1, the results for the final configuration outlined in Section II.A are shown here. Because limited data exist on the UCl_3-NaCl salt compound, temperature-dependent density correlations used in Ref. 3 were used.

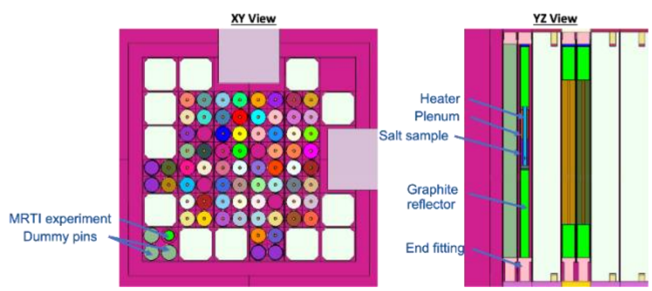

Fig. 2. Visualization of the MCNP model.

In light of the programmatic requirement to maximize burnup, highly enriched uranium (HEU) is used as the fuel. The isotopic vector is shown in Table I. Ref. 1, did not model the additional dummy pins which are now captured by the model as shown in Fig. 2. Lastly, some minor mismatches between the MCNP model and CAD were corrected here. Two variants of I625 were considered, each with different Ta/Nb contents. Ultimately the I625 with the highest Nb content was selected moving forward for conservatism from a heat dissipation standpoint.

TABLE I. Heat generation rates in different components.

^{232}U	^{234}U	^{235}U	^{236}U	^{238}U
0.0%	1.0%	93.2%	0.3%	5.6%

The resulting heat generation in different components of the capsule are summarized in Table II. These values were then used as input for the thermal calculations in Table II.C. Overall the 20 W/cm^3 targeted power density was reached. The overall reactivity impact of MRTI was estimated at 1.99 ± 1.07 cents, significantly lower than the NRAD safety limit of ±40 cents.

TABLE II. Heat generation rates in different components.

	Heat Generation (W/cm^3)
Salt	19.14
Heater + thermowell	0.15
Capsule radial wall	0.24
Outer can	0.14
Lower graphite	0.02
Insulating disk	0.01
Lower capsule plug	0.26
Upper capsule plug	0.13

Initial source term calculations were performed using COUPLE (for cross-section library generation) and ORIGEN (for depletion) [4]. The activity following a 30-day irradiation was 183 Ci, which dropped to 64 Ci, ten days after irradiation. The overall production of noble gases (namely Xe, Ar, and Kr) is 1.24 mg, which constitute around a 15% in plenum gas pressure.

II.C. Thermal-Hydraulic Evaluation

The initial thermal analysis relied on ABAQUS and assumed that thermal convection was negligible in the salt [5]. This was considered the bounding conservative estimate from an experiment safety standpoint. Computational Fluid

Dynamics (CFD) analysis was then conducted using STAR CCM+ [6]. Both analyses were three-dimensional and accounted for thermal heat conduction throughout the CAD model. In the CFD analysis, the heat transfer by the natural circulation of molten salt was also modeled.

Segregated flow solvers were used in STAR CCM+, along with the Reynolds-Averaged Navier-Stokes (RANS) equations-based solver with Shear Stress Transport (SST) k-ω turbulence model. In light of the lack of data on salt properties, approximations were relied upon for their temperature-dependent equations.

Fig. 3 showcases a comparative analysis between ABAQUS and STAR. The maximum observed ΔT was 300°C in ABAQUS versus 200°C in STAR. This was expected due to the high dimensionless Raleigh number ($>10^9$) in the experiment. In both models, targeted temperature limits are met: the salt must stay above 550°C (enough margin from freezing) while the heater is rated at 1,100°C. The more representative CFD results show adequate margins (50 & 200°C respectively).

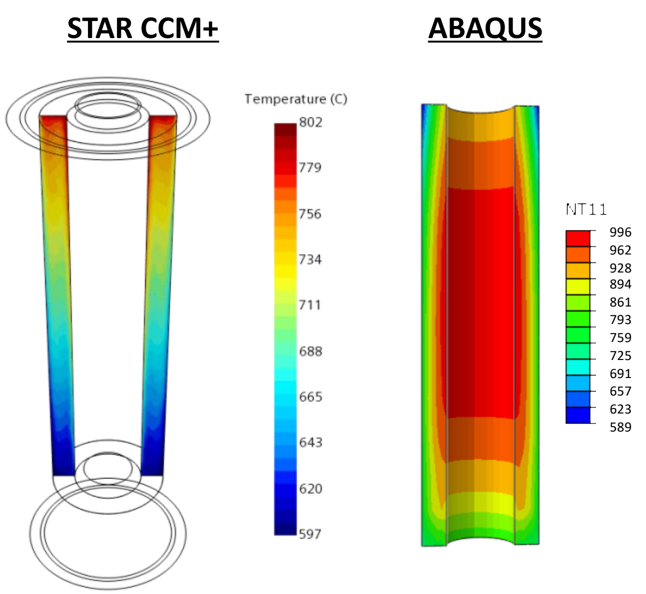

Fig. 3. Comparison of the temperature distribution in the salt between ABAQUS and STAR CCM+.

An additional analysis was conducted using ABAQUS for when fission heat is generated. The heat generation rates from Section II.B were used. This case is found to be even less limiting, with a nominal ΔT ~ 262°C. Overall the experiment thermal requirements are deemed to have been met.

III. PROTOTYPING AND EXPERIMENTS

Salt removal experiments were conducted for various inner capsule geometries. While thinner annuli resulted in lower salt ΔT, capillary forces may complicate their removals during PIE. To assess this, salt removal prototyping experiments were conducted. Three capsule variations were manufactured with salt annuluses of 0.3, 0.4 and 0.5 cm thick. A surrogate salt of LiCl-KCl eutectic was taken to be representative. Wires and steel rods were placed inside each capsule to mimic the potential impact of sensors and thermowells. After the salt in powder form was inserted, the capsules were placed in a 500°C heater as shown in Fig. 4. The salt was then left to solidify overnight.

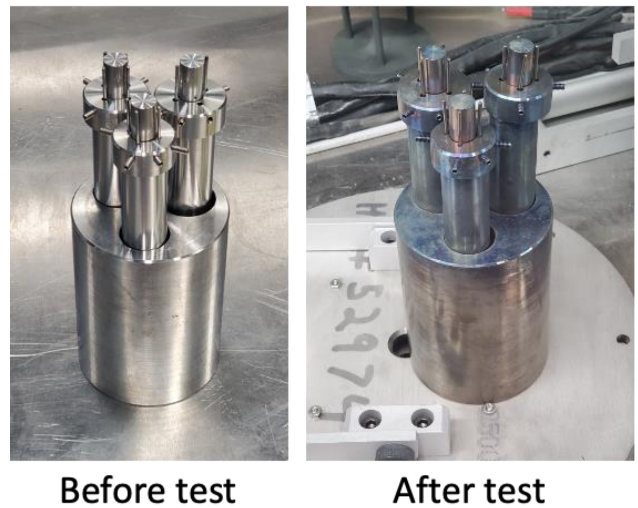

Fig. 4. Salt removal mockup test before/after insertion into heater.

Using an electric impact driver in a positive pressure glove box, the capsules were mounted and continuously vibrated to break up and remove the salt. By first removing the thermowell surrogate rod during the process, the salt could more easily extracted. Recovery of up to 99.2% of salt mass were reached as shown in Table III. It is worth noting that variations in removal rates were primarily due to experimental setup rather than annulus thickness itself.

Fig. 5. Extracted salt following prototype testing.

TABLE III. Salt removal rate after melting/freezing.

Annulus Thickness	Initial Salt Mass	Recovered Salt Mass	Recovered Fraction
0.3 cm	15.000 g	13.892 g	0.926
0.4 cm	20.001 g	15.908 g	0.795
0.5 cm	25.002 g	24.804 g	0.992

It was also deemed important to test the empirical performance of the heater in the temperature range of interest for MRTI. Heater tests with multiple Type K thermocouples were conducted on a benchtop insulated assembly as shown in Fig. 6. At the heater thermowell outer wall, a temperature reading of ~1,000°C will result in the integrated heater TC reading 1,076°C. This offset is important in controlling the temperature of the salt via the internal heater TC and identifying a overtemperature limiting value. Additionally, because the bottom 0.25" of the heater is an unheated section, measuring the temperature there is important to assess if all the lower salt is in a melted state. The recorded temperature of the coldest section of the heater tip was found to be 808°C – higher than the salt melting point – when the outer thermowell wall was 1,000°C.

Thermal contact between the heater wall and thermowell inner diameter is ensured by the application of a heat transfer T99 concrete compound over the heated section prior to insertion into the thermowell. This material was found to be suitable for ensuring good thermal contact.

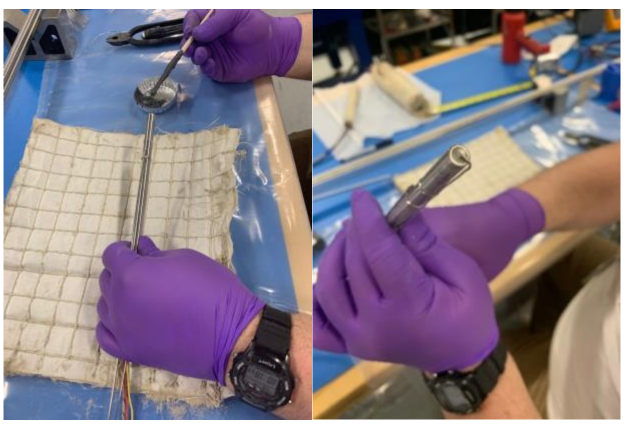

Fig. 6. MRTI heater testing on a benchtop.

Additional prototyping and testing are planned for the MRTI experiment. Brazing and weld procedures are currently being developed for the inner capsule and outer pin assembly and coupons have been manufactured for this work. Part of the assembly was 3D-printed as shown in Fig.7 to test fit the pin into the NRAD standard cluster end fitting, and to provide insight to operators and PIE on experiment handling. Prototypical capsules will be provided to PIE engineers to test the laser fission gas removal system (GASR). A full MRTI mock-up assembly will be built and tested in a representative water test chamber that simulates the NRAD environment. Lastly, a 'dry-run' experiment will be fabricated without salt and irradiated in the F1 position prior to the final fueled MRTI experiment. This last experiment will provide baselining data on the neutronics calculations.

Fig. 7. First version of MRTI 3D Print.

IV. CONCLUSION

A fissile-bearing molten salt irradiation experiment was developed and major aspects of its designs were tested. Mechanically, the MRTI pin and inner capsule design allows for integration into existing irradiation facility interfaces. An immersion heater will control the salt temperature inside of the inner capsule. Neutronics evaluation confirmed that a targeted 20 W/cm^3 power density could be reached. Thermal modeling and optimization showed that the salt can be maintained within required bounds: above melting point and below the heater temperature limits. Prototype testing demonstrated that salt can indeed be extracted during PIE phases from the thin annulus considered. Heater testing showed good thermal contact between the heater and the thermowell can be achieved. Overall, the current design for the MRTI vehicle satisfies the primary objectives of the experiment and is expected to provide high impact information on source term, corrosion, and thermophysical properties of irradiated chloride-based fuel salts.

REFERENCES

1. A. ABOU-JAOUDE, et al., "Conceptual Design of Temperature-Controlled Fueled-Salt Irradiation Experiment to Support Demonstration of Advanced Nuclear Reactors", *Proceedings of IMECE*, Virtual, November (2021).
2. D. PELOWITZ, T. GOORLEY, and M. JAMES. "MCNP6 User's Manual, v1.0", LA-CP-13-00634, Los Alamos National Laboratory, Los Alamos, NM. 2013.
3. V. N. DESYATNIK, et al., "Density, Surface Tension, and Viscosity of Uranium Trichloride-Sodium Chloride Melts," *Atommaya Energiya* Vol. 39 (1975): pp. 70–72.
4. B. T. READERN and M. A. JESSEE, "Scale Code System Version 6.2.3," Oak Ridge National Laboratory, ORNL/TM-2005/39, (2018).
5. ABAQUS Standard, Version 6.14-2, SIMULIA, Inc., Providence, RI. 2014.
6. CD-ADAPCO, "STAR CCM+ User's Manual", . http://www.cd-adapco.com/products/star-ccm/documentation.