

Status memorandum on ORNL support of Vendor Irradiation Capsule design and ASME irradiation code development and new ASTM test standard development activities



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September 2024



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Advanced Reactor Technologies

**STATUS MEMORANDUM ON ORNL SUPPORT OF VENDOR IRRADIATION
CAPSULE DESIGN AND ASME IRRADIATION CODE DEVELOPMENT AND NEW
ASTM TEST STANDARD DEVELOPMENT ACTIVITIES**

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September 2024

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ABBREVIATIONS & ACRONYMS

AGC	Advanced Graphite Creep
ASME	American Society of Mechanical Engineers
ASTM	ASTM International (formerly American Society for Testing and Materials)
ATR	Advanced Test Reactor
DOE	US Department of Energy
HFIR	High Flux Isotope Reactor
INL	Idaho National Laboratory
MIF	Materials Irradiation Facility
NGNP	Next Generation Nuclear Plant
ORNL	Oak Ridge National Laboratory
U.S. DOE-NE	U.S. Department of Energy office of Nuclear Energy

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ABSTRACT

This report is in submission of completion of the Level 4 milestone number M4TG-24OR0501111 within the larger Advanced Reactor Technologies Gas Cooled Reactor program at Oak Ridge National Laboratory (ORNL). The focus of this year's efforts is to continue support of industry needs related to the development of future industry-funded irradiation programs. At ORNL these efforts included the development of a generalized three-dimensional CAD model of an irradiation creep capsules for graphite, cost estimates of the expansion of the Materials Irradiation Facility (MIF), involvement with ASTM D02.F0 "Manufactured Carbon and Graphite Products" and the ASME Boiler and Pressure Vessel Code, and publication of papers supporting these efforts. This report will document the status of these activities.

1. INTRODUCTION

This report summarizes the work performed during FY24 in support of future graphite irradiation needs and the continued support of graphite codes and standards development. This report is the culmination of the Level 4 milestone “Status memorandum on ORNL support of Vendor Irradiation Capsule design and ASME irradiation code development and new ASTM test standard development activities” M4TG-24OR050111. The work undertaken in this FY was directly in support of industry needs to advance the deployment of graphite moderated nuclear reactors.

The efforts within FY24 were intended to progress the design of irradiation creep capsules for the ORNL HFIR flux trap, support of the reactor developer needs related to design and construction codes, materials property testing, and release of preliminary irradiation data for multiple graphite grades. The preliminary design of the irradiation creep capsule includes the development of a 3D CAD model with an undefined internal design that can be tailored to the needs indicated by a reactor designer or graphite manufacturer. Any future irradiation creep capsule will also require use of the Materials Irradiation Facility (MIF), that is utilized to instrument and control capsules. Any irradiation creep capsule will require expansion of the MIF, so current cost estimates to replicate the MIF for a new irradiation creep capsule are included. Another critical component of industry support is the development and revisions of standards used for material property testing within the American Society for Testing Materials (ASTM) International. These testing standards are critical to ensuring confidence in the properties measured before and after neutron irradiation damage and quantifying these changes. The continuous updating and revisions of building and operating codes within the American Society of Mechanical Engineers (ASME) and the Boiler and Pressure Vessel Code (BPVC) is critical to advancing the construction of future reactors. These codes include the requirements of quantifying graphite property changes due to neutron irradiation damage, and the development of plans for in-service monitoring of graphite cores. Multiple preliminary graphite irradiation programs have been performed at ORNL, and the final thrust to support vendor needs is the publication of these results in the open literature.

2. BACKGROUND

The U.S. Department of Energy, through the office of Nuclear Energy (DOE-NE), has been responsible for the Advanced Graphite Creep (AGC) Experiment to produce irradiation effects data in graphite grades for advanced nuclear reactors. The AGC was initiated under the Next Generation Nuclear Plant (NGNP) program in 2005 and was transition to the Advanced Reactor Technology program in 2016. The AGC program consists of six irradiation capsules to be irradiated in the Advanced Test Reactor (ATR) at Idaho National Laboratory (INL) and one capsule irradiated in the High Flux Isotope Reactor (HFIR) at ORNL. The irradiation capsule at ORNL has been completed and INL is currently irradiating the fifth capsule. The sixth planned capsule is scheduled to begin irradiation in 2025. To-date the AGC program has cost ~\$120M, by the time the program finishes in 2028 the total cost is expected to be \$135M.

The AGC experiment was initiated under the NGNP program, which was dedicated to a gas-cooled reactor design. As a result, the six graphite grades selected for the “qualification” level study were those thought to be most desirable for gas-cooled reactors. In the ~20 years since the start of the AGC program, graphite requirements for HTR commercial applications have changed : 1) new graphite grades, different from the grades chosen for previous irradiation programs, are now being considered by HTR designers, 2) gas-cooled reactor designs have shifted from large thermal power plants (600 MW_t per module) to small-modular (100-300 MW_t) and micro-reactor (< 30 MW_t) concepts, and 3) the resurgence of interest in graphite-moderated molten salt-cooled reactors. These shifts mean that the six grades included in the AGC for ASME code qualification may not be as relevant for the newer reactor designs, or that new grades now require ASME code qualification programs. As shown with the AGC program, and with others, the irradiation programs to provide ASME code qualification data are not small undertakings [1].

The U.S. DOE-NE is not currently interested in directly funding another graphite irradiation qualification experiment such as the AGC program, and instead has expressed a desire to allow the commercial vendors to assess their individual and specific requirements in this area. The desire is that these irradiations will most-likely be conducted either at ORNL or INL, and that funding for them will either be from private funding sources or through DOE-awarded funds like those from the Advanced Reactor Demonstration Program (ARDP).

However, it is understood that scheduling, designing, and operating an irradiation campaign is a complex process and the commercial HTR community will require considerable assistance. To this point, DOE tasked INL and ORNL to host a series of workshops to bring together the DOE technical experts to discuss how such an irradiation experiment can be achieved to meet the remaining irradiation requirements for graphite components in gas-cooled and salt-cooled advanced nuclear reactor applications [2, 3]. The efforts within FY24 were targeted to advance readiness at ORNL for future graphite irradiations and support other vendor needs. The first effort within this year was the development of a generalized creep capsule model for the ORNL HFIR flux trap. The second effort was the scoping analysis related to materials and setup of additional MIF capacity to support the irradiation creep capsules. The third effort was involvement with the continued development of testing standards and construction and operation codes. The last effort was the completion of papers detailing the effects of neutron irradiation on the properties of graphite.

3. 3D GRAPHITE CREEP CAPSULE DESIGN

In-situ creep testing of reactor structural materials is of great interest to the advanced reactor research and development community. There are generally two options for designing in-situ creep experiment which involve either an actively or passively loaded specimen region. Various irradiation facilities within the High Flux Isotope Reactor (HFIR) are well suited for both passive and actively controlled creep experiments. See Figure 1 for a schematic of the HFIR core, with basic layout information and irradiation facilities. The primary candidates include the flux trap and removable beryllium, as shown in Figure 1. The flux trap enjoys relatively constant power and peak neutron energies, which make passive temperature control an option there, as well as access to active temperature and creep loading controls. The removable beryllium facility resides radially adjacent to the reactor control plates, which are used to maintain reactor power over its operating cycle. This control behavior causes localized power and gamma flux reduction during the cycle, which can result in change the specimen temperature. The time dependent power shift is mitigated by actively instrumenting an experiment to control temperatures as the cycle progresses and power changes. This capability can be expanded to include controlling creep loads as needed.

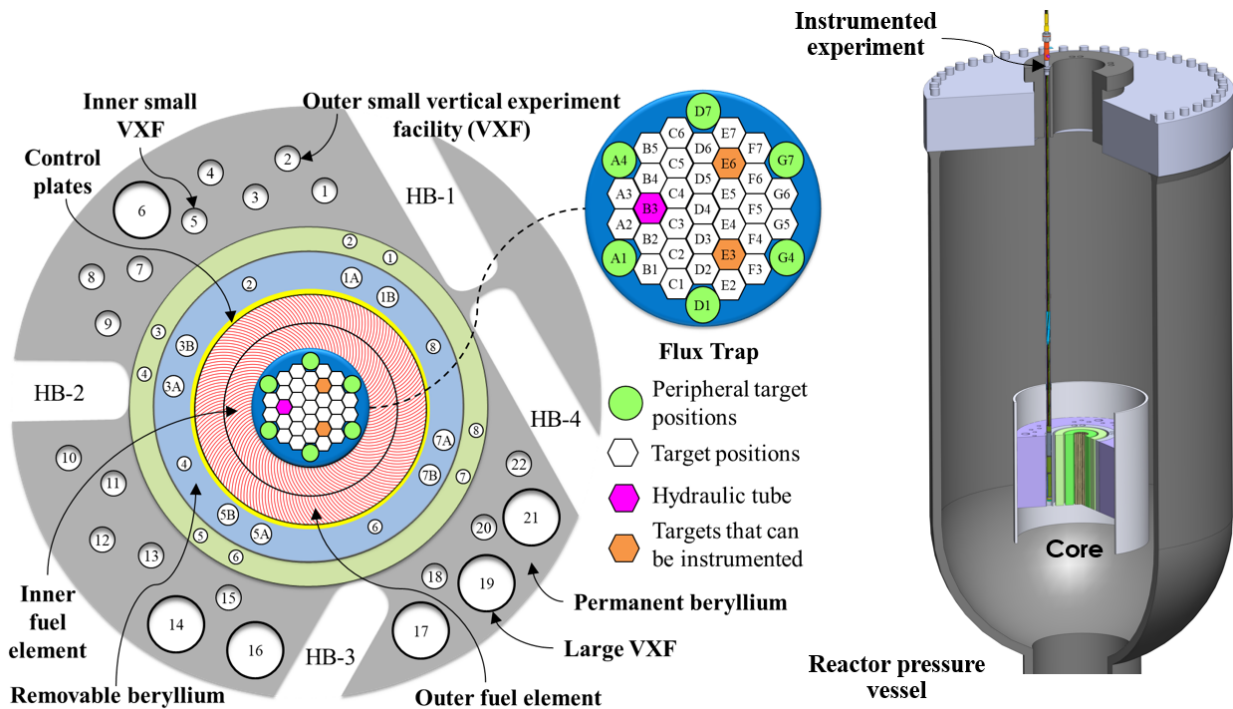


Figure 1. High Flux Isotope Reactor core cross section, and simplified isometric view.

3.1 FLUX TRAP DESIGN DESCRIPTION

More recent work to perform un-instrumented (passive) creep experiments within the High Flux Isotope Reactor (HFIR) flux trap demonstrated the feasibility of applying sufficient load to creep nuclear grade graphite under various temperatures and neutron fluences. This experiment platform utilized a tungsten mass to apply a ~14 MPa stress on the specimen train and was installed in the “instrumented” flux trap target positions (E3, E6 in Figure 1). Further, the experiment design provided three discrete temperature

zones (300°C, 450°C, 600°C) for the 6mm diameter specimens. The HFIR flux trap power shape resembles a chopped cosine with the peak power located at the reactor core centerline. The passive creep experiment housed the loaded specimen set above the reactor centerline, and the unloaded control specimens were installed below the reactor centerline so that the temperature zones for both loaded and unloaded specimens were reflected about the HFIR core centerline.

This experiment was reasonably successful and is used to form the design basis for the experiment described in this work. The current experiment is expected to be instrumented (active) and will control creep loading with a pressurized bellows. The active experiment is also further simplified from its passive predecessor by 1) only targeting a single temperature zone within the experiment and 2) separating the loaded specimen train from its unloaded control counterpart. Active controls for this experiment platform are managed by the HFIR Material Irradiation Facility (MIF), which is described in section 4. The active capsule design, enters the HFIR through its quick opening hatch, spans the length of the HFIR pressure vessel and the “experiment region” is installed within one of the E3 or E6 positions. The experiment region is designed to be compatible with the existing flux trap irradiation facility, and the specimen region is constrained to roughly an 11 mm diameter, 550 mm tall cylindrical volume. The specimen region is contained within Al-6061 T6 outer housing tube (11 mm inner diameter) and holds a creep resistant grade 91 steel specimen holder and graphite specimens. These parts are assembled in a coaxial fashion. Further details the experiment layout and loading are described in section 3.3.

The control specimens will be irradiated within a separate “full length” target capsule and is planned to be irradiated simultaneously with its loaded counterpart in the HFIR flux trap so that both specimen sets observe the same irradiation schedule and power history. A rendering of the full-length target is shown in Figure 2. Note the full-length target specimen region is identical to the creep target.

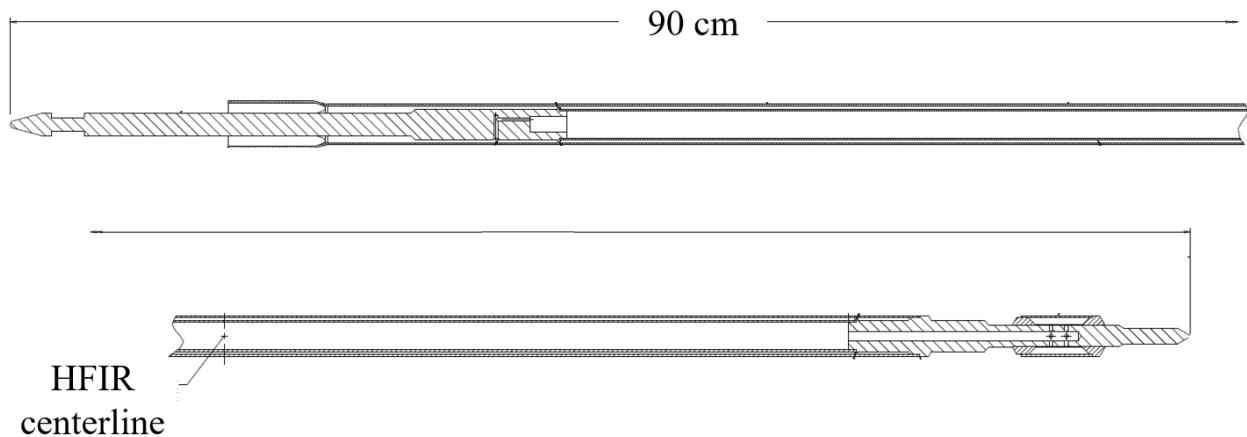


Figure 2. Engineering rendering of the full-length target.

3.2 REMOVABLE BERYLLIUM DESIGN DESCRIPTION

The passive and active experiment designs can be directly translated to the removable beryllium irradiation facility. External access to the experiment is facilitated in the same fashion as with the flux trap style design. The specimen region is larger for the removable beryllium irradiation facility, which consists of a 37 mm diameter, 550 mm tall cylindrical volume. The additional volume could be used to house an unloaded control specimen set, but fully outlining such a hybrid capsule is beyond the scope of

this initial work. Standalone control experiments can be designed, but they would also require active temperature control, given to the power shift described earlier.

Usage of removable beryllium irradiation facility comes with some operational risk. Removable beryllium irradiation experiments displace reflector material while in use, which can negatively impact HFIR cycle durations. The reactor operating authority may choose to prohibit any experiment configurations that lead to a total cycle reduction greater than 1.5 days. This constraint may make the removable beryllium irradiation facility unusable for this research application.

3.3 FLUX TRAP CREEP EXPERIMENT THERMAL DESIGN PERFORMANCE

The flux trap design consists of three sections: 1) the reactor flange penetration, 2) the bellows/load transfer section, and 3) the active experiment section (represented in Figure 3). The reactor flange penetration provides an interface for all active control infrastructure (gas lines, thermocouples, signal processing cabling, etc.) and experiment structure to pass from the low-pressure pool through the reactor vessel containment. There are penetrations above the reactor positions (in this case, flux trap positions E3, and E6 but similar penetrations exist for the RB sites) where the experiment is inserted and secured. Note that certain parts such as watertight hoses and wire passthroughs are not shown in this design to reduce complexity of the figures. The bellows/load transfer section includes a stationary shield plug, bellows, linear variable differential transformer (LVDT), and a push rod. The shield plug provides multiple functions including creating an irradiation shield, restricting coolant flow in the case of an experiment containment breach and establishing a fixed axial location for the bellows to be seated. The bellows is connected to a gas line to supply the creep load, and the LVDT measures axial displacement due to creep. The active experiment houses the creep specimen and can be instrumented with thermocouples to report specimen temperature during irradiation. This section is also supplied fill gas to modify the backfill gas between the specimen holder and housing to control specimen temperature. More detailed images of these three sections are included in Figure 4 through Figure 6. All components described in the flux trap design shown here can be directly translated to an RB style instrumented creep experiment.

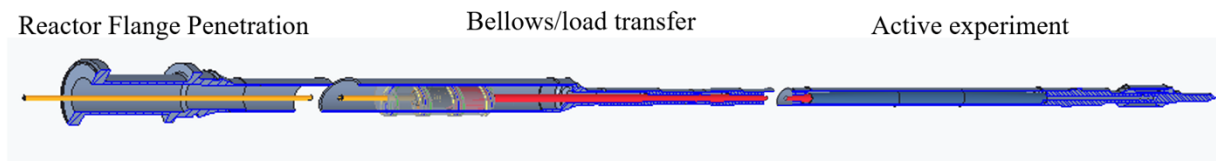


Figure 3. General anatomy of the flux trap creep experiment design.

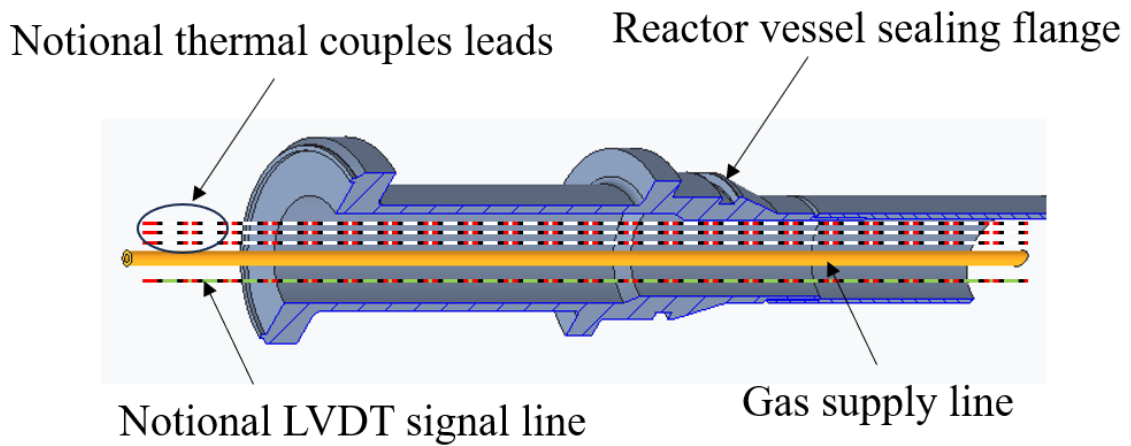


Figure 4. Section 1, the reactor flange penetration.

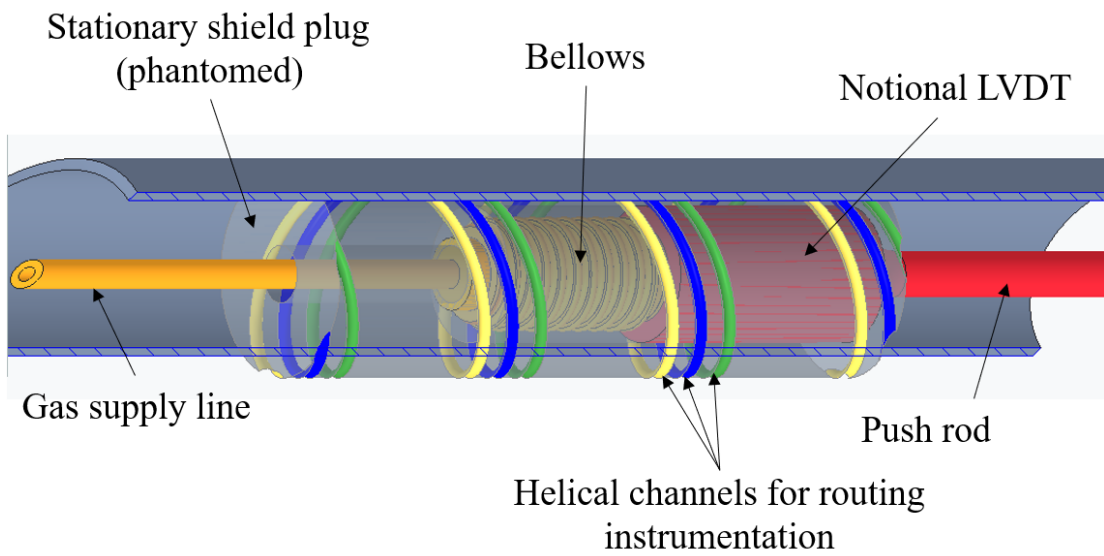


Figure 5. Section 2, the bellows/load transfer section.

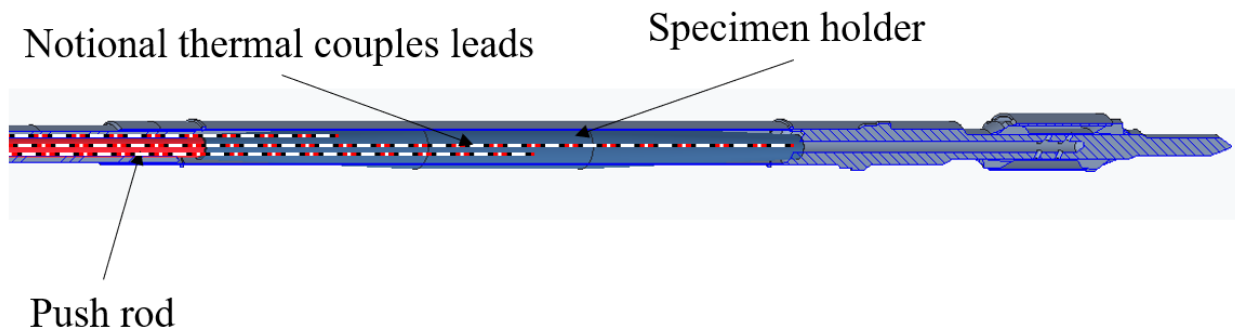


Figure 6. Section 3, the active experiment section.

3.3.1 Methodology

This work performed scoping calculations on the flux trap target design to understand thermal performance of graphite specimens operating at a single design temperature. Temperature is controlled in irradiation experiments by optimizing the gas gap thickness between the housing and holder components, and then backfilling this gap with an inert gas with a known thermal conductivity. Passively controlled experiments are preloaded with single gas composition, where an actively controlled experiment can modify the gas composition based on embedded thermocouple data at a region of interest. This allows real-time temperature correction in actively controlled experiments. Note that this work assumes that the control specimen capsule will not have active temperature control.

For this analysis, the grade 91 holder was designed to have tapers that provide a variable gas gap to counteract the chopped cosine axial power shape of the reactor. The tapers are optimized to form an axially variable gas gap that achieves the specimens' desired average temperature. Note that these design attributes can be directly applied to a removable beryllium irradiation experiment.

For this case, the desired specimen temperature is 550°C ($\pm 10\%$), and the inert gas of choice is helium. The housing inner diameter was set to the nominal dimension of 11.43 mm (Drawing X3E020977A590, Rev. A). Degrees of freedom include:

- 1.) Specimen diameter and corresponding holder inner diameter
- 2.) Holder midsection outer diameter
- 3.) Holder midsection length
- 4.) Holder midsection axial position
- 5.) Holder top end outer diameter
- 6.) Holder bottom end outer diameter

All thermal analyses were evaluated using the Ansys finite element modeling software coupled with a custom, internally maintained set of APDL macros and a Python script used to define material properties and determine thermal conductance between components. Material properties for this calculation are taken from the design and analysis calculations (DACs), and heat generation rates (HGRs) for materials were taken from a previously approved RRD calculations [4]. Table 1 displays the heat generation rates and material property references for the materials used in this calculation.

Table 1. Material heat generation rates and material property references

Part	Material	TRRH HGR (W/g)	Material Property Reference
Housing	Al-6061	30.64	DAC-10-03-PROP_AL6061, Rev. 2 [5]
Holder	F82H-Steel	38.1	DAC-10-10-PROP_F82H, Rev. 1 [6]
Specimen	Graphite	32.5	DAC-10-15-PROP_POCO-GRAPHITE, Rev. 1 [7]

The model geometry consisted of only the active experiment region, with the top and bottom of experiment is set to be adiabatic. A convective boundary condition applied to the outer surface of the housing. These boundary conditions are representative design conditions within the Target Rod Rabbit Holder (TRRH), with a heat transfer coefficient (h_{film}) of 47.1 kW/m²K and a bulk coolant temperature of 52°C. Due to the radial symmetry of the experiment, a 2D axisymmetric model could be utilized, and a nominal mesh element size of 0.5 mm was applied. Figure 7a displays the model with a zoomed in portion to display the mesh used; however, because the component is difficult to see at true scale, Figure 7b displays a not-to-scale cartoon to highlight the component features.

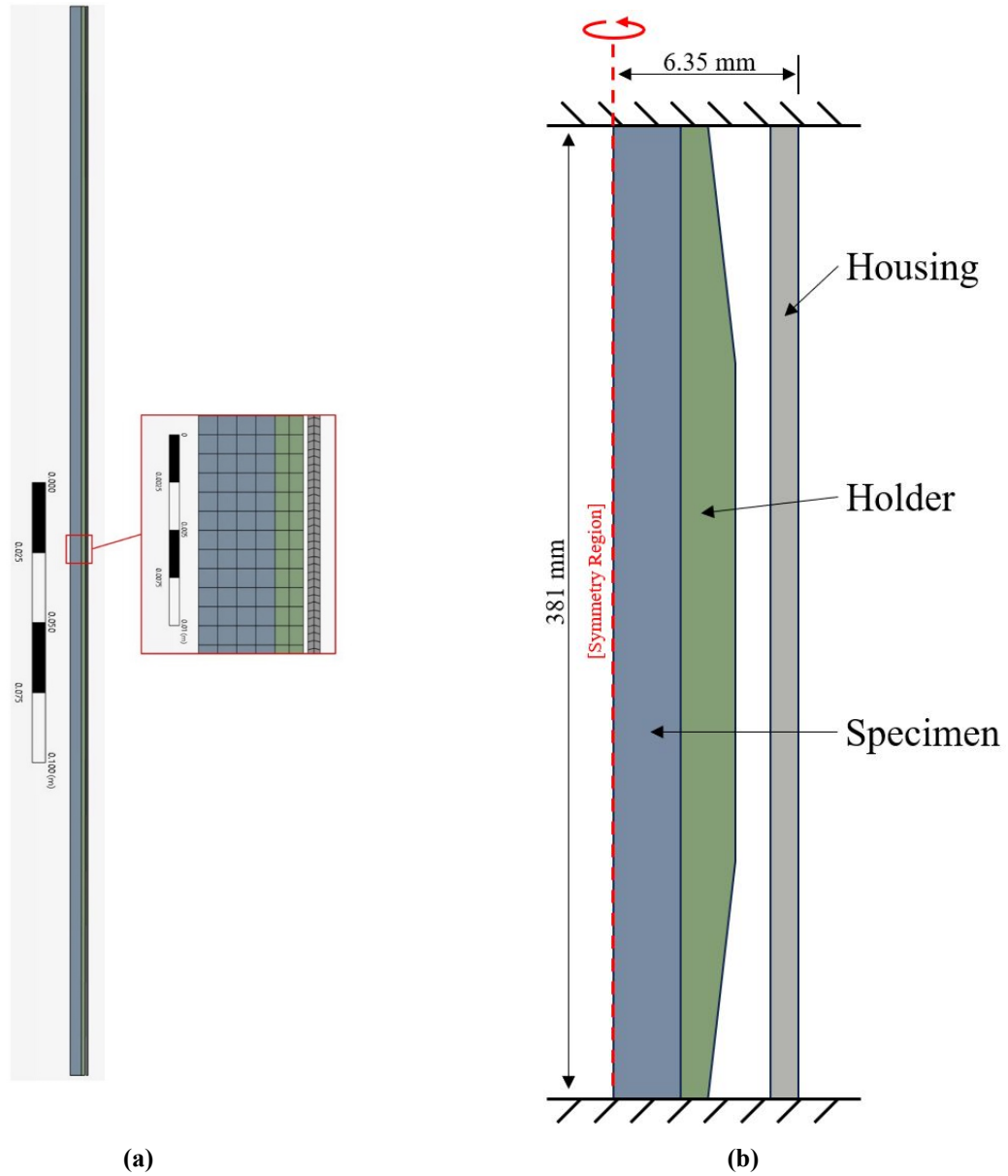


Figure 7. Thermal model design

3.3.2 Thermal Results and Discussion

The following data shows preliminary thermal design performance of the flux trap creep experiment for various specimen diameter cases. Table 2 presents the optimized geometry parameters to achieve bulk specimen target temperatures, while Table 3 displays the relevant thermal output for each case.

Table 2. Optimized component dimensions (mm) for 550°C specimen target temperature

Case #	Specimen OD	Holder Midsection OD	Holder Midsection Length	Holder Midsection Axial CL Pos. [Exp CL = 0]	Holder Top OD	Holder Bottom OD
1	6	11.1	200	50	11	10.8

2	7	11.05	200	37.5	10.9	10.65
3	8	10.95	200	25	10.8	10.55

Table 3. Optimized Case Thermal Results

Case #	Holder Temps (°C)			Specimen (°C)		
	T _{Min}	T _{Max}	T _{Avg}	T _{Min}	T _{Max}	T _{Avg}
1	490	561	534	521	567	550
2	494	563	535	518	571	550
3	498	561	535	515	574	549

These preliminary results are largely constrained by the degrees of freedom present in this simplified model. Two inflection points on the holder outer diameter does not perfectly compensate for the axial power factors of the reactor, which makes the thermal response extremely sensitive to changes in outer diameter at the top and bottom of the holder. Figure 8 below demonstrates this phenomenon. It displays the temperature gradients for a.) the 6 mm diameter specimen nominal case (dimensions detailed in Table 2) and b.) a variation of this case with the bottom holder OD increased by 0.1 mm. In Figure 8a, the high temperature area (in red) along the bottom taper is a product of the linearly increasing shielding not adequately matching the cosine shape of the reactor flux. In Figure 8b, the holder bottom OD was increased in to compensate for this high temperature zone, but this also results in the temperature at the bottom decreasing drastically (~50°C).

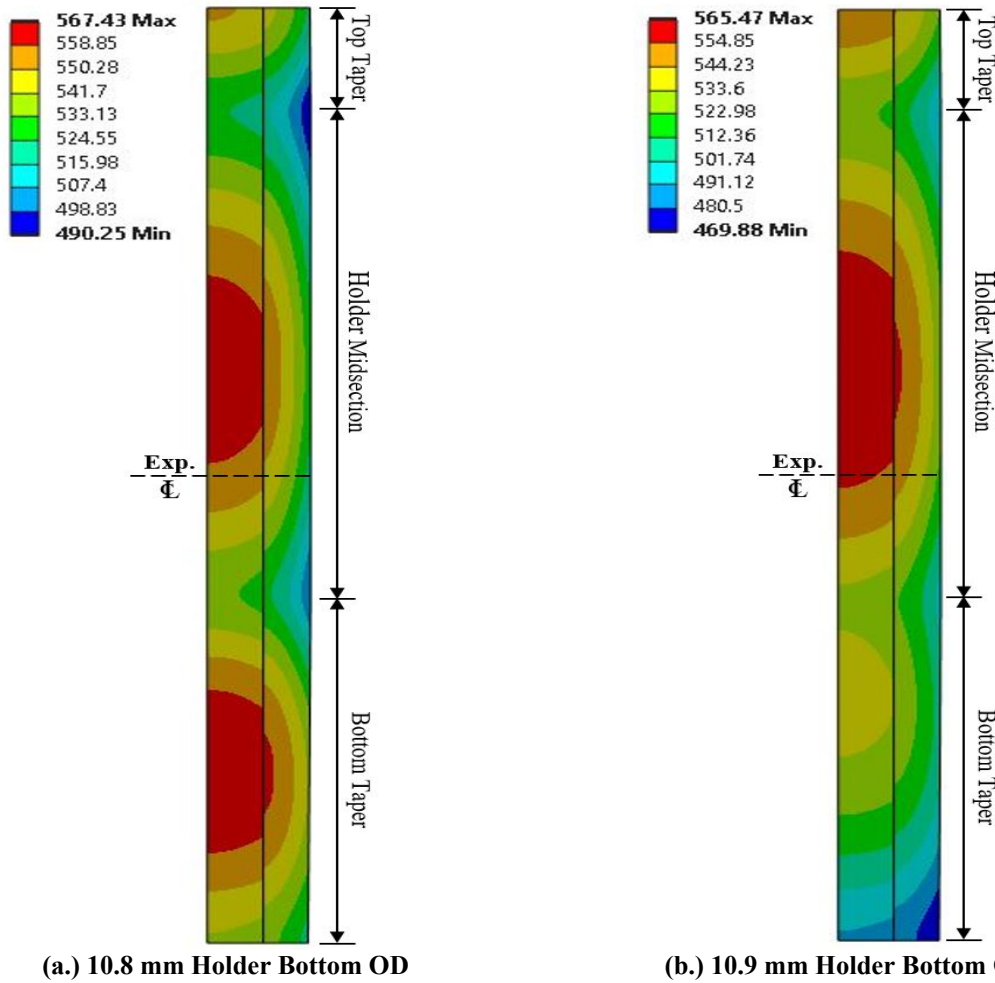


Figure 8. Highlight of thermal response sensitivity to holder end diameter [6 mm OD Specimen].

Adding additional inflection point along the holder outer diameter will allow the shielding to better match the cosine shape of the reactor flux, reduce areas of heat buildup along the tapers. Note: the images in Figure 8 are not to scale; the model was rotated in the along the x-axis (aligned with the experiment bottom edge), to improve the visual representation.

In addition, due to the higher heat generation rate of the F82H stainless steel of the holder relative to that of the graphite specimen, to reach the equivalent design temperatures, the required maximum holder OD must be increased as the specimen diameter is decreased. This feature will need to be taken into consideration when determining the specimen diameter, as the possibility for thermal expansion of the holder into the housing will be a concern for safety basis scenarios.

4. EXPANSION OF MATERIAL IRRADIATION FACILITY (MIF) FOR FUTURE GRAPHITE CREEP EXPERIMENTS

The Materials Irradiation Facility (MIF) has a long history providing scientists the ability to monitor real-time data from instrumented irradiations. The MIF enables access to through the HFIR pool wall for both gas lines (using the dry wall junction box) and electrical leads (using the electrical junction box). Gas mixtures such as Argon, Helium, and Neon flow into the experiment and change heat transfer across multiple zones in the experiment housing. The gas mixture also provides a cover gas for off-gas monitoring of fuel undergoing burnup. Finally, the gas ensures continuous cooling of the experiment for the duration of the irradiation. A second port provides electrical lead access including thermocouple extension cables, triax extension cables, and fiber optic extensions. The MIF lab space is located south of the HFIR core, on the experiment hall, see Figure 9.

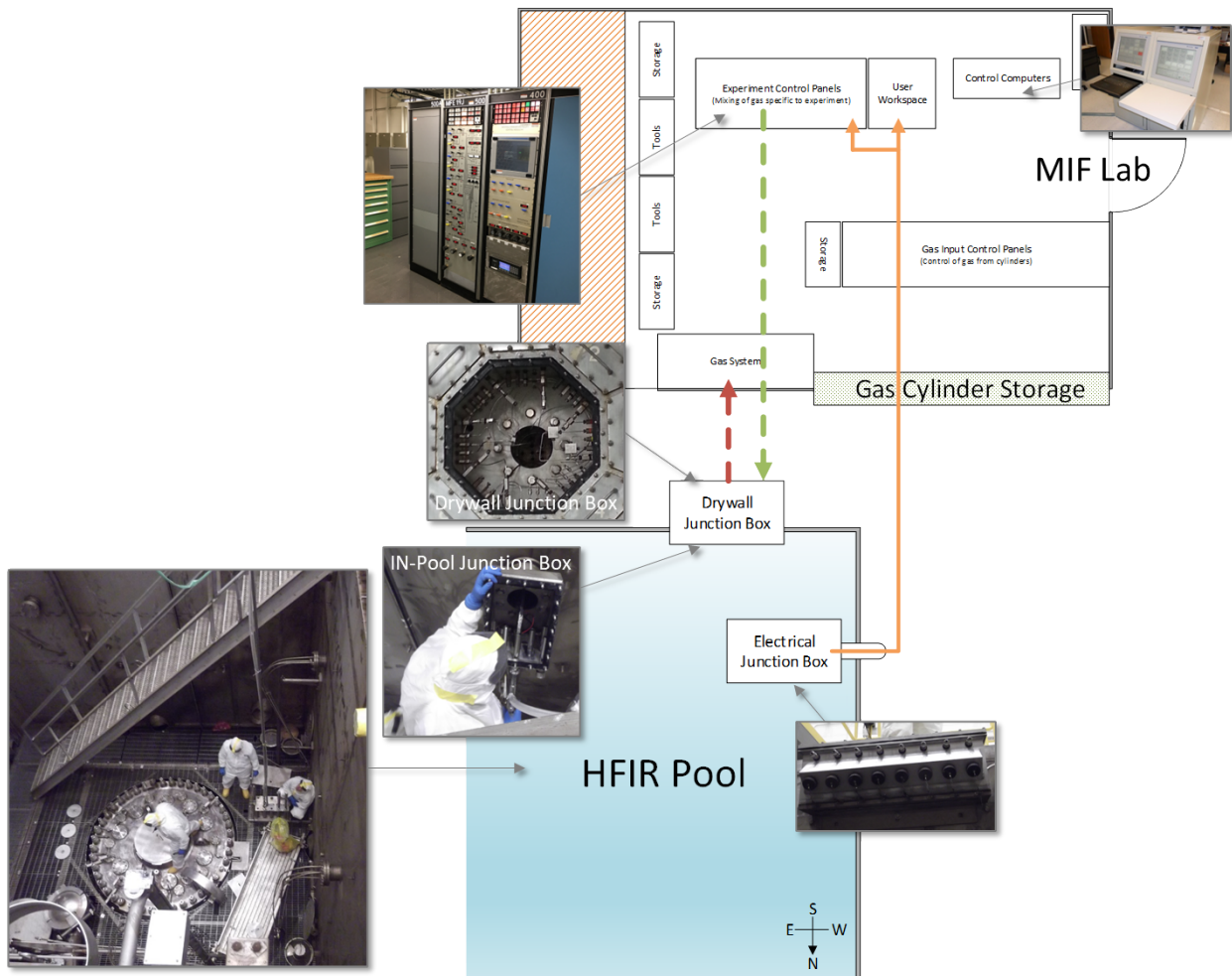


Figure 9. Materials Irradiation Facility (MIF) floor plan illustration with images included of key parts of the facility

Allen Bradley industrial controls, with software that undergoes a software quality assurance (SQA) review, are programmed to always maintain control of the experiment. Assurances of pressure, leak detection, and over-temperatures are monitored during the irradiation and alarm if they exceed the maximum set-point range. These alarms notify the HFIR control room of an issue, and in turn they notify

additional cabinets on either side of the MIF500 that have been leveraged for various purposes and could be equipped with the same or similar hardware. Also, all four cabinets are accessible to the computer control system.

The current internal configuration of the MIF500 cabinet components is listed in Table 4. To reproduce an exact replica of this configuration, the cost for materials is approximately \$252,000. It should be noted that due to the current HFIR user facility agreement, all modifications to the facility are performed by HFIR staff and that requires additional labor cost.

Table 4. MIF 500 bill of materials with quantity and cost at the time of quotation from vendors, including installation and software quality assurance labor cost.

Part	Manufacture	Part#	QTY	Price Each	Total Price
PT	Viatran	Model 345	12	\$967.00	\$11,604.00
MFC	Brooks	SLA5850S	12	\$3,693.00	\$44,316.00
Cable Kit	Brooks	778Z010ZZZ	1	\$189.00	\$189.00
Power Supply	Brooks	0254AA1B15A	3	\$4,206.00	\$12,618.00
PANEL	Custom		1	~\$5k (Optional)	
Displays	Trident	PD765-6R0-10	30	\$542.00	\$16,260.00
25' Cables	Brooks	S124Y059AAA	12	\$332.00	\$3,984.00
Enclosure			1	\$15,422.00	\$15,422.00
T BLOCK	Alan Bradley	2136323	100	\$3.95	\$395.00
T Anchors	Alan Bradley	2116632	70	\$2.25	\$157.50
FUSE Disc.	Alan Bradley	2898	50	\$12.54	\$627.00
fuses	Klien 500ma	3939254	50	\$11.33	\$566.50
Jumpers		2140127	4	\$4.66	\$18.64
DIN RAIL		2116	3	\$28.90	\$86.70
Valves	Swagelok	SS-41GS2-A	18	\$121.22	\$2,181.96
Tubing	Swagelok	SS-T2-S-028-20	100	\$8.08	\$808.00
PCV	Swagelok	KPR1FJA415A20000	7	\$379.56	\$2,656.92
PLC	Alan Bradley				\$139,821.96
Hardware (at time of quotation)					\$251,713.18
HFIR Staff Hours (at time of estimation)					\$243,960.00

There is an on-going discussion between the MIF operators and HFIR staff about modifying the MIF configuration to modernize the system to be a fully digital system with autonomous controls. There are several benefits to this approach, including enabling remote access for external users to log into the lab and view real-time data. However, remote users would be in viewing mode only and not have access to modifying the operation. The autonomous controls will undergo a safety review to ensure safe operation, protecting HFIR and the experiment.

5. DEVELOPMENT OF GRAPHITE CODES AND STANDARDS

FY24 included ORNL involvement with two professional societies to continue to support the development of codes and standards for graphite testing including ASTM International and the ASME BPVC. The ASTM testing standards are critical to ensuring confidence in the properties measured before and after neutron irradiation damage and quantifying these changes. The ASME BPVC includes the requirements of quantifying graphite property changes due to neutron irradiation damage, and the development of plans for in-service monitoring of graphite cores.

5.1 ASTM INTERNATIONAL

The primary focus within FY24 was the revision and reapproval of multiple ASTM standards. The first focus within this effort was the revision of ASTM C559-20 [8]. This “Standard Test Method” underwent a change to a “Standard Practice” due to the lack of an intra- and inter- laboratory round robins. The other primary revision was the removal of the 500 mm³ minimum volume, and instead making the minimum dimensions dependent on the resolution of mensuration equipment and the size of features within the different graphite grades.

The other on-going effort during FY24 was the revision of ASTM D7219-19 [9]. The revision of this “standard specification” is to clarify the terminology section. Within this specification there are multiple uses of the same words (i.e. charge, lot, batch) that when references in other cases, including the ASME BPVC, that the full term is not utilized making for a continued confusion. While revising this “standard specification”, a comparison with ASTM D7301-21 [10] will be performed. There is significant overlap between the two specifications and the committee believes that sunseting of D7301 may be in order.

The final effort within the ASTM D02.F subcommittee is revising D7846-21 [11]. The primary revision will be reverting Tables 2 and 3 to include all the values that were listed in the 2016 version of this standard. The original tables included the Upper and Lower bounds for the Maximum Likelihood Estimator (MLE) in Table 2, and Upper and Lower Confidence bounds in Table 3. In the 2021 version, all the values except those for the lower 5% were removed due to a misunderstanding that users of the standard may be using this for items other than the ASME code analysis. Statisticians at INL have performed numerical analysis of these confidence trends and the resulting best-fit equations for these values in Table 2 and Table 3 in the standard will also be added. Lastly, section 1.2 that provides information about how the results of the Weibull analysis can be used will be removed in this revision since the testing standard should remain agnostic of use.

The ORNL voting representative on the ASTM D02.F subcommittee was selected by ASTM as a recipient of the D02 Emerging Professionals Award for the June 2024 committee meeting. This award included two days of training about all the various aspects of ASTM, from topics as high level as society structure, to the process of getting standards written and approved, to the minutiae of how to handle voting result. During the D02.F subcommittee meeting, this same representative was elected to be the chair of the D02.F0.0B “Nuclear Applications” as the current chair decided to step down from position.

5.2 ASME BPVC

Continued involvement in the development of the ASME BPVC is paramount to the future construction and operation of high temperature nuclear reactors that utilize graphite as a moderator and/or reflector. ORNL has staff involved within ASME include both Section III Division 5 “High Temperature Reactors” and Section XI Division 2 “Requirements for Reliability and Integrity Management (RIM) Programs”. Within III-5 membership in the Working Group on Nonmetallic Design and Materials (SG HTR) (BPV III), Working Group on General Requirements for Graphite and Ceramic Composite Core Components

And Assemblies (SG GR) (BPV III), and Special Working Group on High Temperature Reactor Stakeholders (SG HTR) (BPV III) is critical to supporting the design and construction of graphite-moderated advanced reactors. Efforts within XI-2 Task Group on Non-metallic Component Degradation and Failure Monitoring (SG-RIM) (BPV XI) are focused on the development of plans for in-service monitoring of graphite cores.

Within the Working Group on Nonmetallic Design and Materials there are two task groups that are supporting reactor deployment: “irradiation effects in graphite task group” and the “design task group”. The irradiation effects in graphite task group is focused on developing an addition to the code of “generalized trends” that describe the neutron-irradiation induced property changes in graphite from temperatures of 300-900°C up to the turn-around fluence. The design task group, within the Working Group on Nonmetallic Design and Materials, is focused on revisions of the code that enhance the understanding of the failure calculations that utilize the Weibull analysis. The effort for this year included multiple thrusts, including revisions to article HHA-3000 where the “recalculation” of the threshold strength will also include the recalculation of the modulus, making the code conservative but not to the point that it is too conservative to be used by designers.

6. PUBLICATION OF ORNL GRAPHITE IRRADIATION PAPERS

The primary thrust of the first half of FY24 was the completion of two paper drafts on the neutron irradiation studies performed at ORNL. The first paper completed is titled “Materials Property Changes in ETU-10 Graphite due to Neutron Irradiation at Elevated Temperatures”. This paper captures the neutron irradiation effects in ETU-10 graphite that was irradiated at nominal temperatures of 300°C-900°C up to a total damage of ~30 dpa. The second paper is titled “Effect of neutron irradiation at elevated temperatures on the materials properties of IG-110 and IG-430”. This paper captures the neutron irradiation effects on graphite grades IG-110 and IG-430 for nominal temperatures of 300°C-1000°C up to a total damage of ~27 dpa. Both papers captured the changes in properties including mass/volume/density, elastic properties, coefficient of thermal expansion, thermal diffusivity, and strength. These papers are undergoing final input from co-authors and will be submitted to journals in early FY25.

The second half of this effort analyzed the effects of neutron irradiation on the equiaxial strength of ETU-10. These results were presented at the 2024 ASME Pressure Vessel and Piping (PVP) meeting, and published in the proceedings [12]. This work looked at the effect of neutron irradiation on the 2-parameter Weibull distribution. The primary findings from the work were that the change in the 2-parameter Weibull modulus was best described by a rapid decrease when exposed to neutrons that was not affected by increasing dose. During the ASME PVP meeting, a colleague from industry suggested that they had observed a similar decrease without irradiation and that the drop may be caused by annealing of residual surface strains/damage from the specimen machining process. This requires further investigation, and more discussions as to whether as-machined or post-machining annealed is a more representative surface for reactors.

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