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MITR Users' Guide

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1. MIT Nuclear Reactor Laboratory (MIT-NRL) Mission and Goals

The MIT Nuclear Reactor Laboratory (MIT-NRL Figure 1) is an interdepartmental laboratory that conducts interdisciplinary research in radiation and nuclear energy applications. MIT-NRL operates a 6 MW research reactor, the MIT Research Reactor (MITR), which is the second largest university research reactor in the United States. Our mission is to provide faculty and students from MIT and other institutions with both a state-of-the-art neutron source and the infrastructure required to facilitate use of the reactor.

As a partner in the Battelle Energy Alliance (BEA) which now operates the Idaho National Laboratory (INL), MIT intends to make the MITR a preeminent university center for research and education in the areas of materials and fuel research for advanced nuclear energy systems. The MITR is a partner reactor facility of the ATR National Scientific User Facility (ATR-NSUF) for advanced materials and fuel research and development to facilitate collaboration among national labs, universities, and industry.

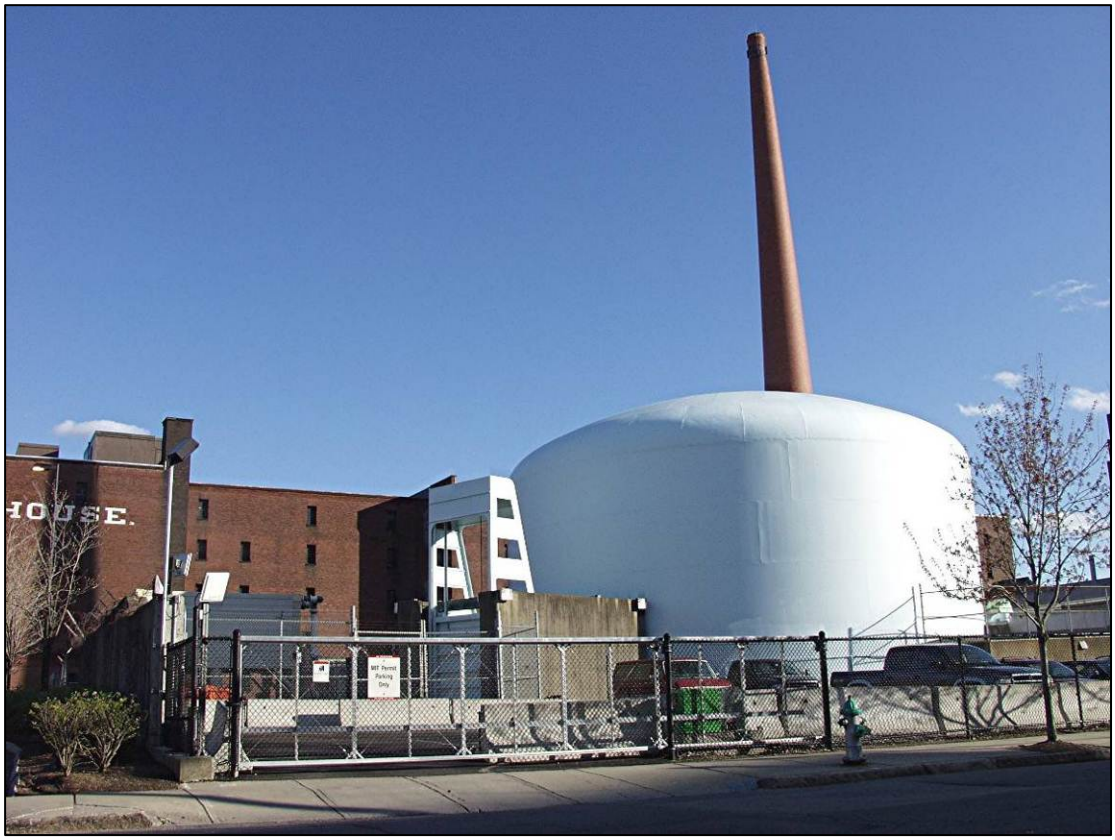


Figure 1. The MITR is located on the MIT campus in Cambridge, Massachusetts.

1.1 Educational Activities and Public Outreach

The MITR has the advantage of being affiliated with the very highly regarded Nuclear Science and Engineering Department (NSED) at MIT. The NSED and other departments use the MITR for both thesis research projects and laboratory exercises. Students are particularly enthusiastic about experimental work on the MITR because it gives them the opportunity to apply their academic learning to challenging engineering and scientific problems. Also, they acquire the skills needed to coordinate projects and are imbued with the "safety culture" needed for the proper operation of nuclear facilities. Training, research, and educational opportunities at the MITR are offered through 1) MIT undergraduate and graduate-level laboratory courses; 2) research projects as part of the Undergraduate Research Opportunity Program (UROP); 3) student reactor operator training program; 4) senior and graduate thesis research projects.

The MIT-NRL also maintains a successful public outreach program. The educational and outreach activities include:

- Tours of the MITR by the general public and high school students (about 1000-1500 tours per year);
- Assistance to students with science fair projects involving photon irradiations.

2. MITR General Description

The MITR (Figure 2) is a tank-type research reactor. It is owned and operated by the Massachusetts Institute of Technology, a non-profit educational institution, and is licensed by the US Nuclear Regulatory Commission. Its current license, issued in November 2012, authorizes steady-state 6 MW operation for 20 years. As shown in Figure 3, the reactor has two tanks: an inner one for the light water coolant/moderator and an outer one for the heavy water reflector. A graphite reflector surrounds the heavy water tank.

The reactor utilizes flat, plate-type fuel elements. Each rhomboidal fuel element consists of fifteen plates of UAl_x cermet clad with 6061 aluminum alloy. Longitudinal fins on the fuel plates increase the heat transfer area. The core has 27 fuel element positions and is normally configured with 24 fuel elements and 3 positions available for in-core experiments. The close-packed hexagonal core design maximizes the thermal neutron flux in the heavy water reflector region where the re-entrant thimbles of the beam ports are located. The light-water core, heavy-water reflector, and graphite region are all separately cooled; each transfers heat to a secondary coolant that dissipates it to the atmosphere via two cooling towers.

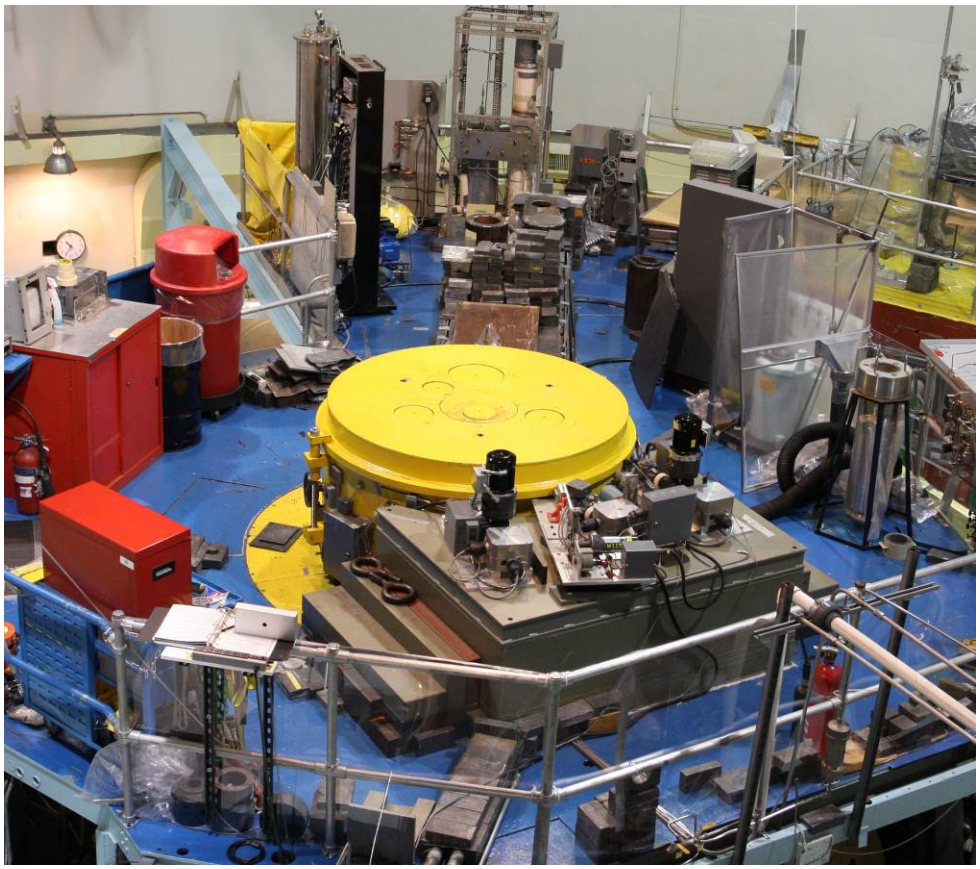


Figure 2. The MITR reactor top.

The MITR operates at atmospheric pressure. Primary coolant, at a nominal flow rate of 2000 gpm (~125 kg/s) enters the bottom of the core tank through the core shroud, flows upward through the fuel elements and then exits at the outlet piping about 2 m above the top of the core. The primary coolant core inlet temperature is approximately 42 °C and outlet temperature is about 50 °C. The hexagonal core structure is about 38 cm across with an active fuel length of about 56 cm. The compact core has an average power density of about 70 kW/l, with fast, thermal, and gamma fluxes similar to those of a commercial light water power reactor (LWR).

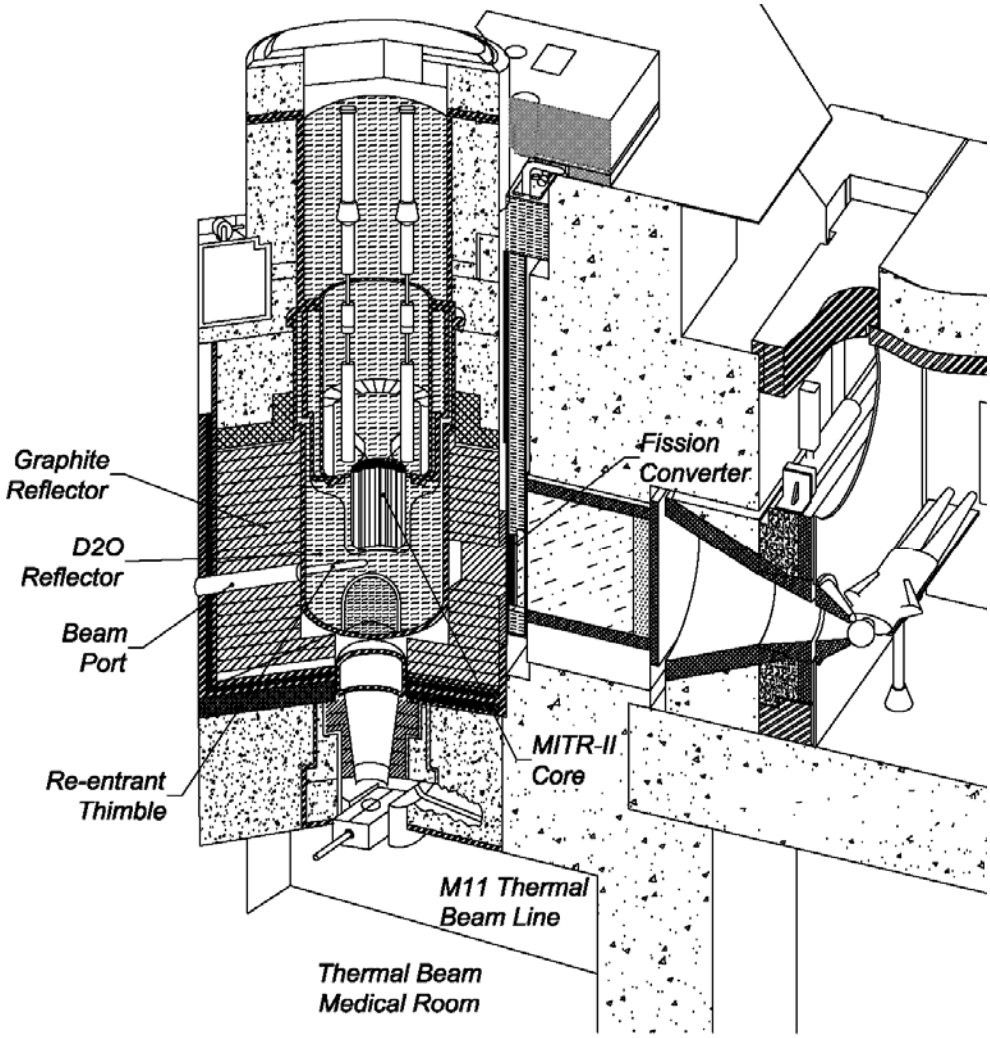


Figure 3. Cutaway schematic of the MITR.

The reactor design includes a number of passive safety features. The principal ones are negative reactivity temperature coefficients of the fuel and moderator, a negative void coefficient of reactivity, the use of anti-siphon valves to isolate the core tank from the effect of breaks in the coolant piping, a core-tank design that promotes natural circulation, and the presence of a full containment. During a loss-of-primary flow transient, decay heat is removed by natural circulation within the core tank through the natural circulation and anti-siphon valves and no forced cooling is required. The anti-siphon system prevents uncovering of the core in the event of a loss of coolant accident.

The MITR is equipped with a wide variety of sample irradiation facilities, with fast and thermal neutron fluxes up to 1.2×10^{14} and 5×10^{13} /cm^2 s, respectively; these facilities are described in the following section. The MITR operates 24 hours a day, 7 days per week. A typical fuel cycle lasts about 9 weeks followed by a 2 week refueling and maintenance outage.

3. MITR Irradiation Positions and Experimental Facilities

3.1 Overview of Experimental Facilities

The available irradiation positions at the MITR are summarized in Table 1. Table 2 gives flux and sample dimension information for the irradiation positions.

Table 1. MITR Experimental Facilities

A. In-Core*

- Inert gas-filled irradiation tube (ICSA) for sample capsule irradiation at <900°C (instrumented or un-instrumented.)
- Forced-circulation coolant loops that replicate conditions in both pressurized and boiling water reactors.
- Facilities for testing mechanical properties of samples in a light water reactor environment.
- High temperature (>1000°C) irradiation facility for materials irradiations in inert gas (He/Ne mixture).

B. Beam Ports

- One 12-inch diameter port, radial.
- Two 6-inch diameter ports, radial.
- One 6-inch diameter through-port.
- Four 4-inch diameter radial ports.
- One 4-inch diameter through-port

C. Pneumatic Tubes

- 2PH1 – 2" high flux pneumatic tube.
- 1PH1 – 1" intermediate flux pneumatic tube.

D. Graphite-Reflector Irradiation Positions

- Two 3GV (graphite-vertical) irradiation facilities located in the graphite reflector.

E. Shielded Medical Rooms

- Medical room with horizontal epithermal beam from a fission converter.
- Medical room below core with vertical thermal beam.

*In-core facilities are not permanently installed.

Table 2. Representative dimensions and flux levels in MITR irradiation facilities.

Facility	**Size**	**Neutron Flux (n/cm^2-s)**
In-core	3 available Max in-core volume ~ 1.8" ID x 24" long	Thermal: 3.6x10^{13} Fast: up to 1.2x10^{14} (E>0.1 MeV)
Beam ports	Various radial: 4" to 12" ID	Thermal: 1x10^{10} - 1x10^{13} (source)
Vertical irradiation position	2 vertical (3GV) 3" ID x 24" long	Thermal: 4x10^{12} - 1x10^{13}
Through ports	One 4" port (4TH) One 6" port (6TH).	Avg thermal: 2.5x10^{12} to 5.5x10^{12}
Pneumatic Tubes	One 1" ID tube* (1PH1)	Thermal: up to 8x10^{12}
	One 2" ID tube* (2PH1)	Thermal: up to 5x10^{13}
Fission Converter Beam Facility (FCB)	Beam aperture ~ 6" ID	Epithermal: ~ 5x10^{9}
Thermal Beam Facility (TNB)	Beam aperture ~ 6" ID	Thermal: up to 1x10^{10}

*Dimensions of the sample holders used in the pneumatic irradiation facilities are 1" diameter by 3-1/4" length for the 1PH1 location, and 1-3/8" diameter by 6-1/4" length for the 2PH1. Images of these holders are shown in Section 3.3.

3.2 In-core Experimental Facilities

The MITR core is shown in Figures 4 and 5. Figure 4 is a schematic core map showing the fuel element positions, core structures and control elements. Generally, positions A1, A3 and B4 are occupied by solid dummy elements or in-core experiments. Figure 5 is a photograph of the core with an annular fuel irradiation experiment installed in position A3. In-core experimental facilities at the MITR are installed by removing a solid dummy fuel element and replacing it with a dummy element that accommodates the desired in-core loop configuration. In some cases the dummy fuel element is integral with the in-core experimental facility structure. Note that these facilities are not permanently installed and are removed from the core tank when they are not in use. This means that a given facility can generally be tailored to meet the requirements of a particular experiment or multiple experiments/specimens.

There are two general types of in-core experimental facilities, illustrated schematically in the 3-D model of the reactor shown in Figure 6. The first, designated "In-Core Sample Assembly" (ICSA), uses an S-bend tube to give access to the in-core irradiation space from the top of the reactor core tank. This facility is generally used for sample irradiations in an inert gas atmosphere with limited requirements for in-core instrumentation (typically thermocouples only). The ICSA is cooled by the reactor coolant, but sample capsules can be insulated to take advantage of nuclear heating for elevated temperature exposures. Active heating or cooling is also potentially available in

an ICSA subject to design and safety review. Figure 7 shows the neutron spectra comparison of an ICSA and a typical PWR core.

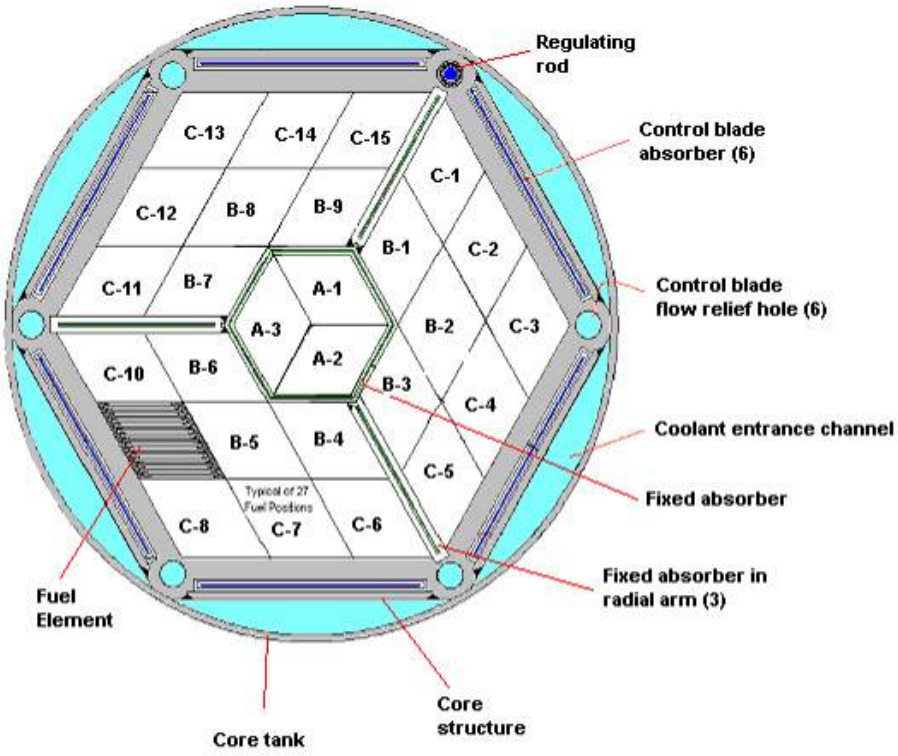


Figure 4. MITR core map showing fuel element position designations and major core structures.

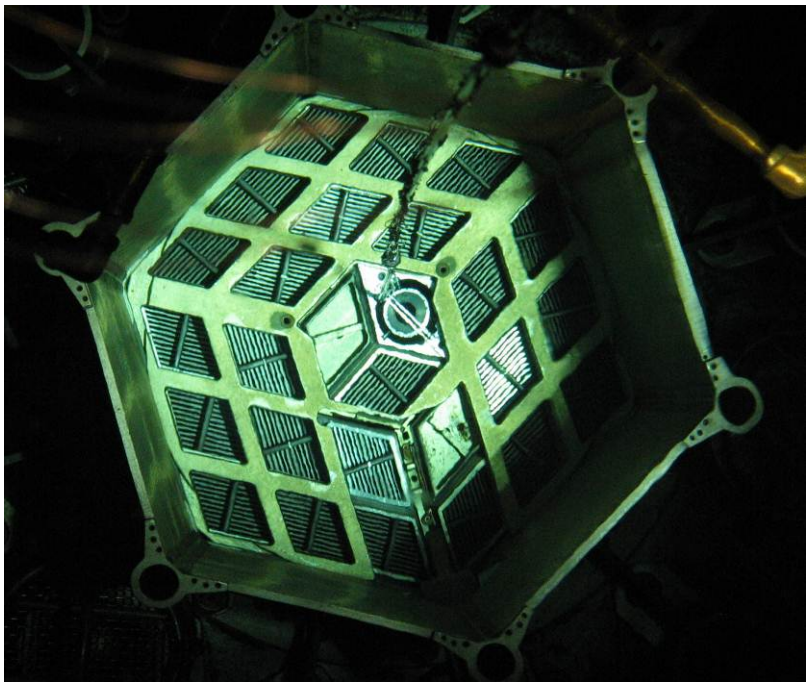


Figure 5. Photograph of the MITR core with two solid dummy fuel elements and in-core experimental facility installed.

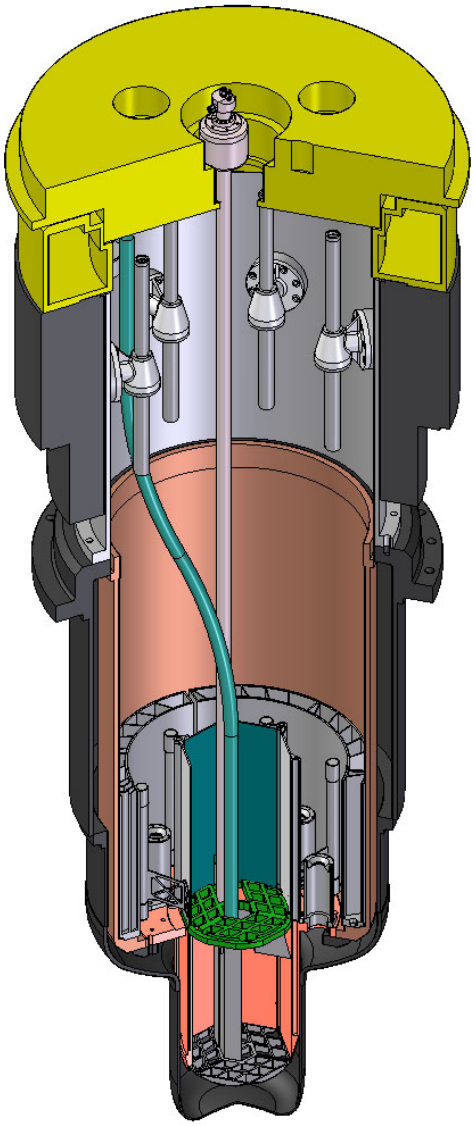


Figure 6. Cutaway 3-D model of the MITR showing an ICSA (blue) and an in-core water loop installed in the reactor core (fuel elements removed for clarity.)

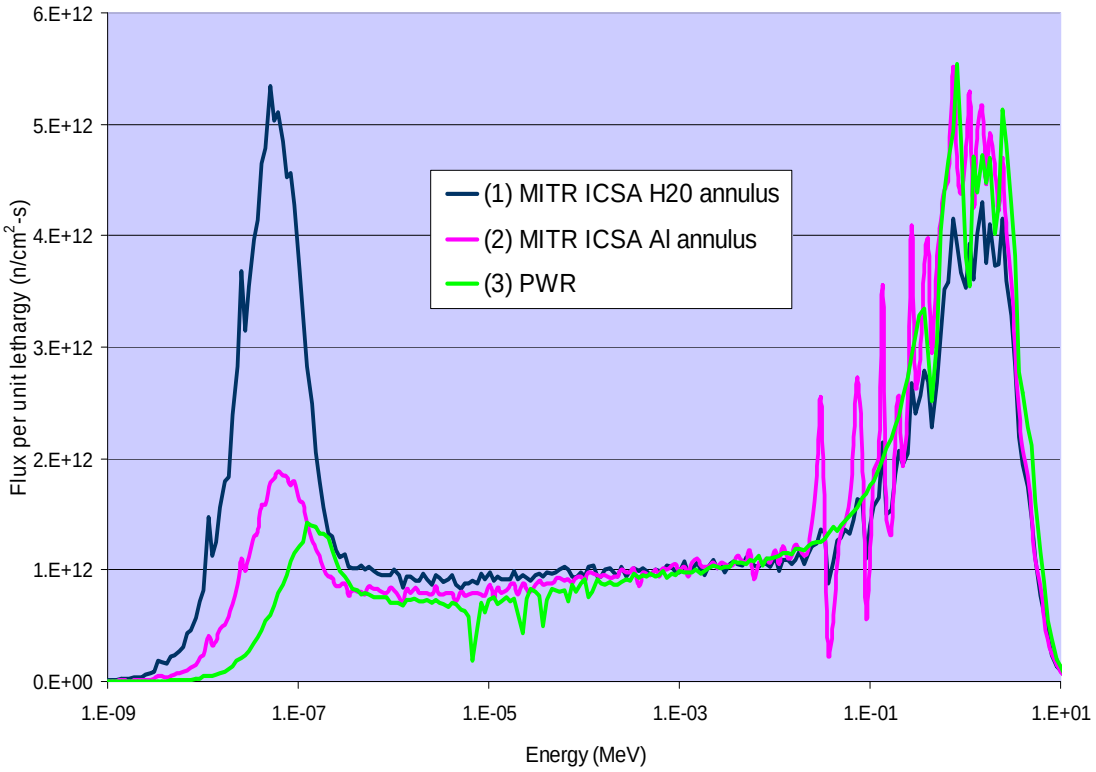


Figure 7. Comparison of the neutron spectra for (1) an MITR In-Core Sample Assembly (ICSA) with a 0.15 cm water annulus, (2) a MITR ICSA with an aluminum annulus, and (3) a typical PWR core. Calculations were performed using MCNP with a three-dimensional MITR model at 5 MW. The MITR in-core neutron spectrum and flux magnitudes are comparable to those of a commercial PWR.

The second type of in-core facility encompasses a variety of more complex irradiation rigs that are expressly designed for a specific purpose. Past examples include LWR loops used to study various aspects of coolant chemistry, passively and actively loaded mechanical tests under LWR conditions, corrosion test loops for advanced clad materials, a test of internally and externally cooled annular fuel, an irradiation test for high temperature gas reactor materials at temperatures approximately 1000-1600°C, and irradiation of multiple uranium-zirconium hydride fuel rods at PWR temperatures. These experiments are described in more detail in Appendix A to illustrate the types of in-core facilities that can be used and the design envelope that has been previously approved and demonstrated. In some cases, similar experiments can utilize existing irradiation rigs. The modular design and small size of these facilities, however, makes it possible to design and construct new facilities for specific purposes at moderate cost.

A variety of in-core and support instrumentation and measurement can be provided to support in-core loop irradiation. Thermocouple temperature measurement is the most common in-core measurement. Other in-core instrumentation that has been used includes electrochemical corrosion potential and other electrode measurements and DC potential drop strain and crack growth measurement. A variety of data acquisition equipment is

available if experimenters desire to use or test their own in-core instrumentation. A wide variety of parameters are routinely monitored and recorded at out-of-core locations for the in-core facilities. For water loops these include temperature, pressure, flow, dissolved hydrogen and oxygen, and conductivity. Real-time residual gas analysis (mass spectrometry) is also available. Radiochemical and chemical assays can be performed on site on batch-sampled radioactive coolant samples. For chemical assays, ICP-OES, INAA and prompt gamma NAA are available. MITR staff will support use of specialized instrumentation required for a particular experiment, subject to funding and manpower constraints

Table 3 summarizes the general characteristics and constraints of the in-core positions. Note that fuelled irradiations can only be performed up to a maximum of 100 g ^{235}U equivalent and that sample fuel must be double encapsulated and cooled by the reactor primary coolant. That is, test fuel cannot be operated in a loop that relies on its own system for circulating coolant.

Table 3. Characteristics and constraints for in-core sample facilities.

Parameter	Permissible values	Comments
Total in-core volume	2” ID x 22” long	Maximum available opening in an in-core dummy fuel element
LWR sample space	~ 1” ID x 22” long	Typical – dependent on autoclave design pressure and materials.
High temperature sample space	~ 0.8” ID x20” long	Dependent on temperature desired and gamma heating susceptor material choice.
MITR coolant wetted materials	Aluminum, stainless steel, titanium, zircaloy	Small amounts of other materials on case by case basis
MITR coolant heat flux	<400 kW/m^2	No Onset of Nucleate Boiling
Fissile material content	<100 g U-235 or equivalent	Fissile materials other than U-235 require pre-approval
Facility reactivity Secured Non-secured Movable	 <1.8% ΔK/K <0.5% ΔK/K <0.2% ΔK/K	“Movable” reactivity limits apply to coolant phase change and dissolvable neutron poison.

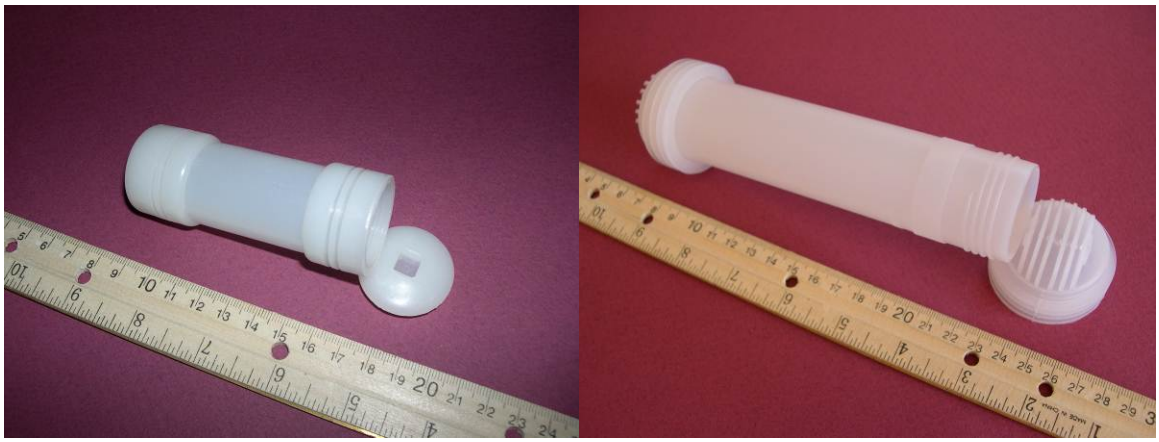
3.3 Pneumatic Tube Transfer System

The MITR is equipped with two pneumatically-operated irradiation facilities that allow materials to be exposed to high or intermediate-level neutron fluxes. A 2” pneumatic facility (2PH1) offers a high thermal flux of up to 5x10^{13} n/cm^2-s with a significant fast

neutron flux (cadmium ratio of about 20). A 1" pneumatic facility (1PH1) offers an intermediate, highly thermal flux of up to $8x10^{12}$ n/cm^2-s (cadmium ratio of about 200). These facilities are normally used to support MIT-NRL's Neutron Activation Analysis program (described in Section 4.1 below) and for isotope and radiotracer production.

Materials are transferred through the pneumatic facilities in sample holders known as 'rabbits', shown in Figure 8. The 1PH1 system uses a polyethylene rabbit with internal dimensions of 1" diameter by 3-1/4" length; the 2PH1 system uses either polyethylene or titanium rabbits with internal dimensions of 1-3/8" diameter by 6-1/4" length. Titanium rabbits are generally only used for 2PH1 irradiations longer than 10 hours.

The 1PH1 pneumatic system is capable of ejecting the sample to a hot cell within the reactor containment or to a laboratory in an adjacent building. This allows materials to be irradiated for short periods (typically from 10 seconds to 10 minutes) and examined within minutes.



(a) Polyethylene 1PH1 rabbit. (b) Polyethylene 2PH1 rabbit.

(c) Titanium 2PH1 rabbit.

Figure 8. Sample holders or 'rabbits' used in the 1PH1 (a), and 2PH1 (b and c) pneumatic irradiation systems. The titanium rabbit (c) is generally used only for 2PH1 irradiations longer than 10 hours in duration.

3.4 Beam ports

Numerous beam ports penetrate the reactor's shield, graphite reflector, and heavy-water reflector as shown in Figure 9. These provide a high-quality neutron flux for such endeavors as neutron scattering, prompt-gamma analysis, neutron physics, and neutron transmutation doping (NTD). Currently, port 4DH1 is equipped with a time-of-flight neutron spectrometer with remote access capability, and a neutron diffractometer at 4DH4.

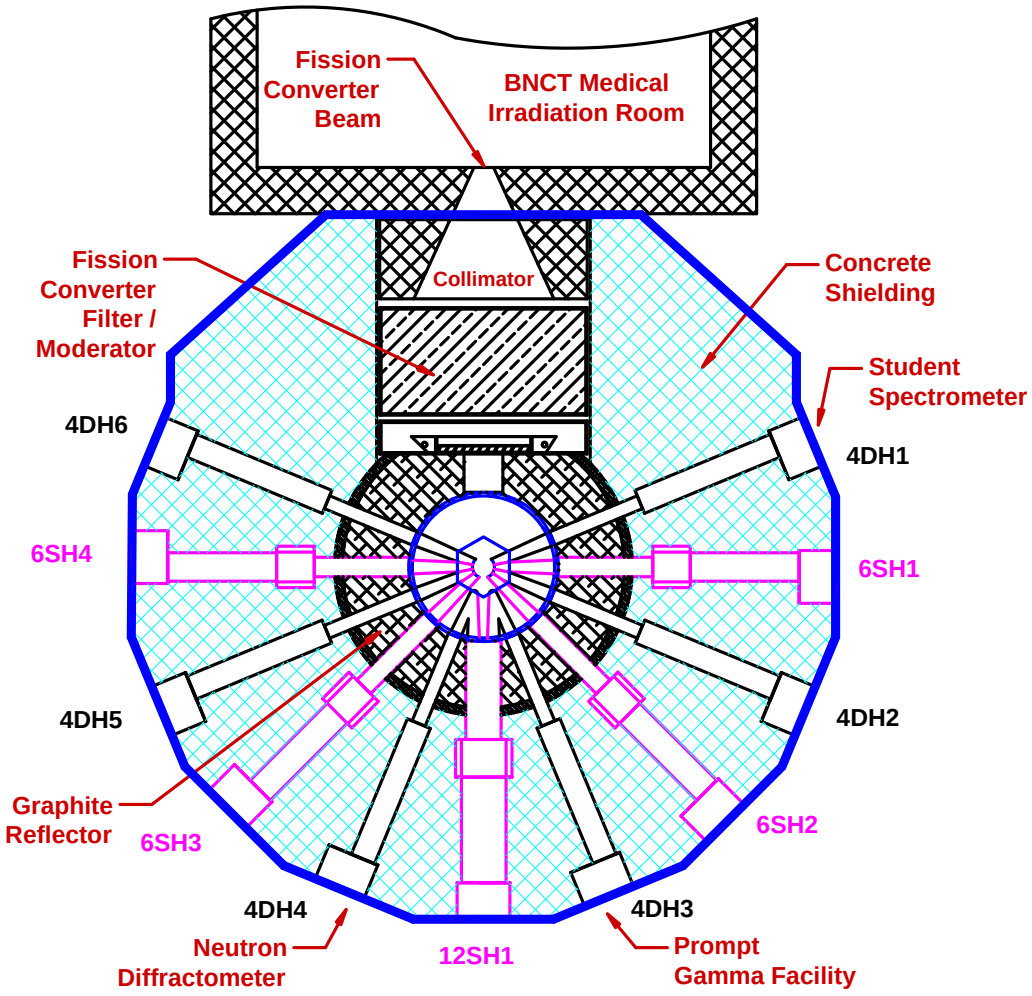


Figure 9. Cutaway view of the MITR showing the radial beam ports.

3.5 Horizontal through ports and vertical irradiation facilities

MIT utilizes the 4TH and 6TH horizontal through ports for neutron transmutation doping (NTD) of single crystal silicon. These through ports are tangent to the D_2O reflector tank as shown in Figure 10. The 4TH port can accommodate 4 inch OD samples and the 6TH port can accommodate up to 6 inch OD samples. Automatic conveyer, neutron detectors,

and control systems were installed for routine NTD silicon crystal doping as shown in Figure 10. These through ports offer high-quality thermal neutron flux and can be used to accommodate larger sample irradiations. There are also two 3" vertical irradiation thimbles (3GV) available for sample irradiations in the graphite reflector region, also shown in Figure 11. These thimbles offer a uniform, thermal flux for an active height of about 24" and are suitable for medium and long-term irradiations.

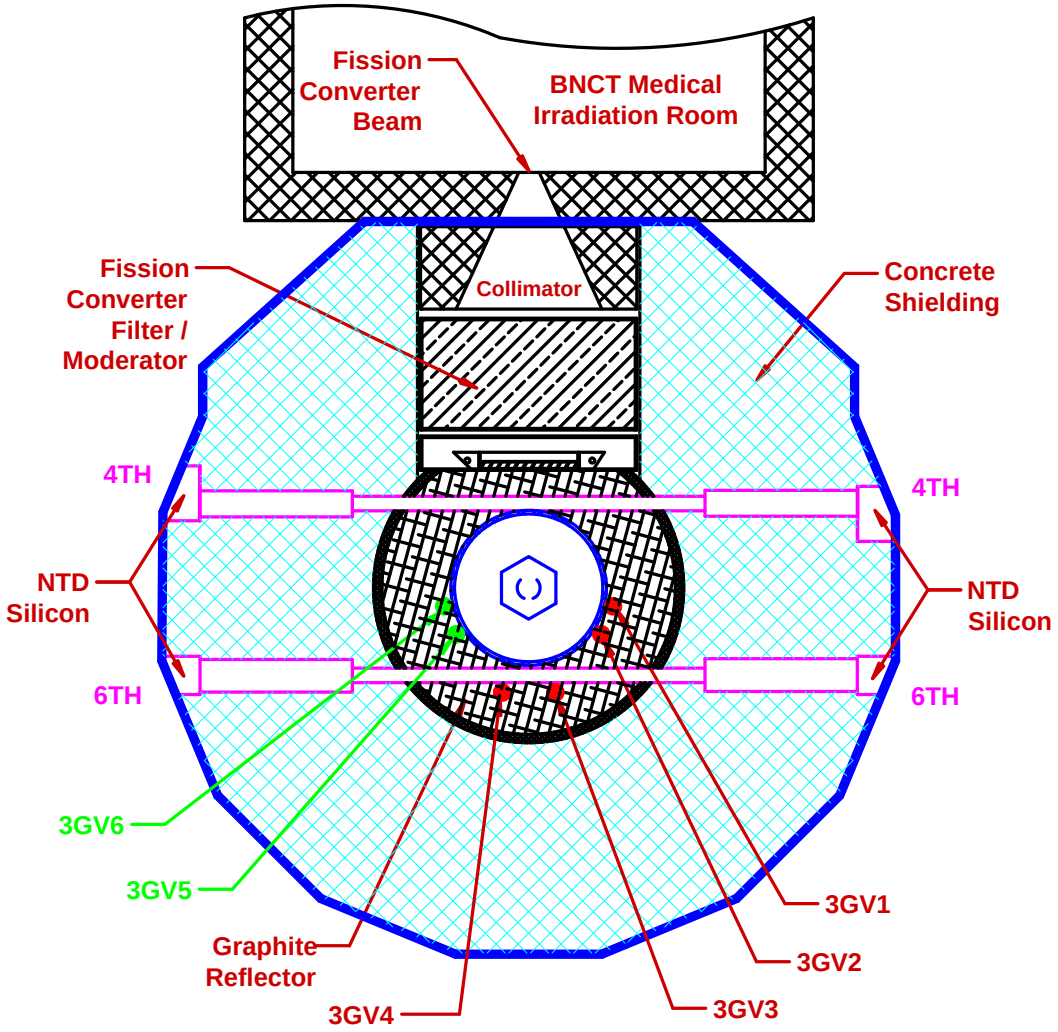


Figure 10. Cutaway view of the MITR showing the horizontal through ports and the location of the 3GV vertical irradiation facilities in the graphite reflector region.

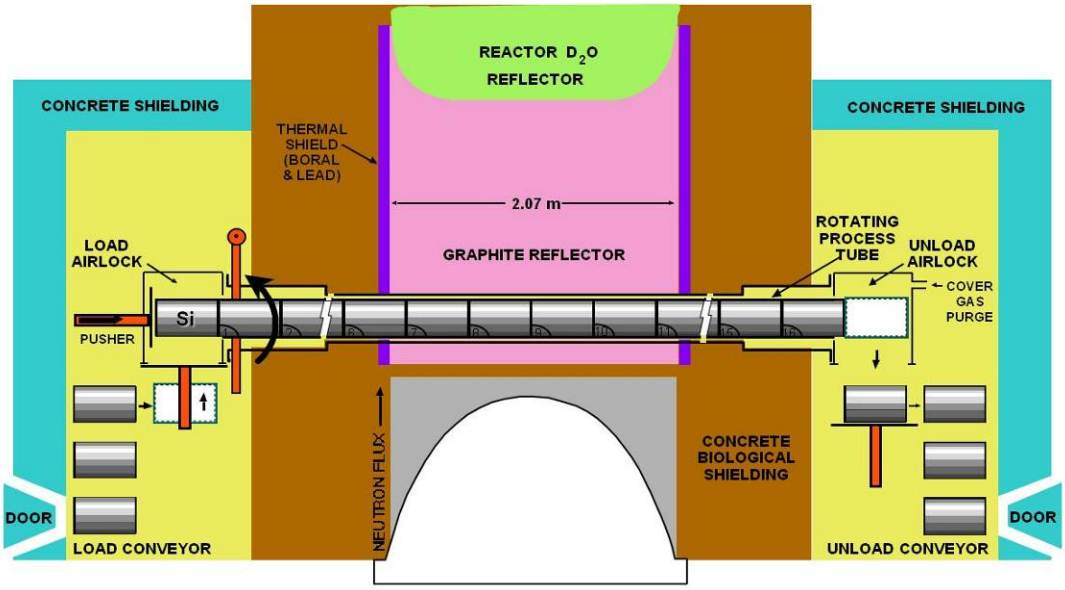


Figure 11. Schematic of the MITR NTD system.

3.6 Medical Irradiation Facilities

The Fission Converter Beam (FCB) is one of the highest intensity epithermal neutron beams available in the world. The beam quality has been experimentally verified to be near theoretical purity. This beam has a dedicated irradiation room which was constructed for Neutron Capture Therapy (NCT) clinical trials. There is a second dedicated medical irradiation room which is attached to a high purity Thermal Neutron Beam (TNB). This is a high intensity thermal neutron beam that is well suited for small animal studies and for clinical studies of superficial tumors.

3.7 Gamma Irradiation Facility

Gamma irradiations can be performed in MITR's spent fuel pool, utilizing gammas from fission product decay. Dose rates range from 10^3 to 10^5 rad/h, depending on the element configuration. Irradiations have been performed for medical equipment sterilization and high school science classes studying radiation effects on food and seed germination.

4. Support Services, Equipment and Specialized Facilities

4.1 Elemental analysis facilities

The MIT-NRL is currently equipped for both delayed and prompt gamma neutron activation analysis (NAA). Delayed gamma NAA (or Instrumental NAA) techniques are often used in environmental health studies to evaluate the transport of air pollutants, mineral uptake in the body, aerosol formation, the role of aerosols in acid rain, and

geochemical studies. In the past few years, the MIT-NRL's NAA capability has been expanded for toxic trace element analysis in biological samples such as animal and human hair and tissue. The 1PH1 pneumatic tube can be set up so the irradiated sample is transferred automatically to the hot lab in the adjacent building for short irradiations (from a few seconds to a few minutes) for the analysis of short-lived isotopes. The NAA laboratory has three High Purity Germanium (HPGe) detectors, digital multi-channel analyzers, and the Genie 2000 gamma spectrum analysis system.

A Prompt Gamma Neutron Activation Analysis (PGNAA) facility is also available for trace element analysis. PGNAA was used primarily for the quantification of trace B-10 in blood for BNCT research. The PGNAA facility is normally installed at one of the MITR's horizontal beam ports, 4DH3. The thermal neutron flux at sample location is approximately 1.8×10^7 n/cm^2 s. The PGNAA counting facility is equipped with a HPGe detector with a multichannel analyzer, and the Genie 2000 Gamma spectrum analysis.

MIT-NRL is also equipped with an Inductively Coupled Plasma - Optical Emission Spectrometer (ICP-OES) that is used to complement the NAA facilities. ICP-OES is a widely used multi-element analysis technique that offers good precision and speed and can be useful for elements that are not suitable for NAA or where interfering activities are present. The ICP-OES can be used for radioactive sample analysis and has been used for chemical analysis of in-pile loop coolant. Note that ICP-OES analysis requires that the sample be in aqueous solution or uniform suspension and that a standard is available with the matrix matched as closely as possible to the unknown.

5. Post-Irradiation Examination (PIE) Facilities

5.1 Hot cells and handling facilities

The reactor containment building is equipped with an overhead polar crane with 20-ton and 3-ton hooks. These cranes are used for installations and removals of in-core and other experiments. A variety of shielded transfer casks are also available for transfers. There are two hot cells in the reactor hall as pictured in Figures 12 and 13. The larger cell is generally used for handling and disassembly of full-height in-core experiments. This cell is accessible for installation of custom fixturing required for particular experiments. The smaller cell has been used to handle small, high activity components and fuel from in-core experiments. A collimated gamma scan facility can be installed in the small cell. The reactor spent fuel pool is also available for storage, handling and packaging of irradiated experiments. Shipping casks up to the GE2000 can be loaded dry or wet.

Figure 12. Reactor floor main hot cell.

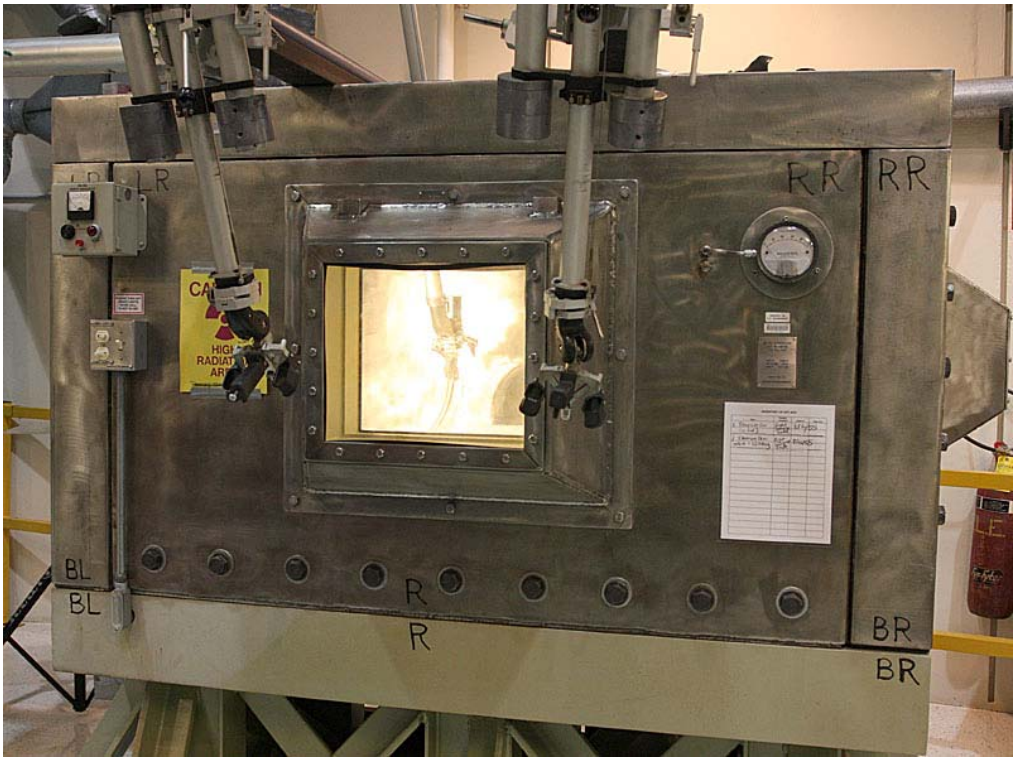


Figure 13. Reactor floor hot box.

5.2 Hot sample preparation facilities

Laboratory space is available within the reactor exclusion area (outside the containment building) with two standard fume hoods and a perchloric acid-capable fume hood for electro-polishing (a Struers electropolisher is available.) A controlled-atmosphere 4-port glove box with furnace is also available. A ventilated hot box with manipulators is located in this laboratory and is available for specialized PIE activities requiring more shielding than can be installed in the fume hoods. Standard metallurgical sample preparation (epoxy mounting, sectioning and polishing) can be carried out on activated samples. Photography and macro-photography for irradiated specimens is also available.

5.3 Electron microscopy facilities

Although there are no electron microscopes within the reactor containment or exclusion area, non-dedicated facilities can be used for hot sample microscopy at MIT. The instruments available in the MIT Department of Materials Science and Engineering Central Facility are described at http://web.mit.edu/cmse/facilities/electron.html. Use of these facilities for irradiated materials is subject to dose limits and approvals from MIT's Radiation Protection Office. In some cases, dedicated sample holders may be required to reduce the probability of contamination of shared equipment.

Appendix A. MITR In-core Loop Experience

A1. Introduction and Overview

The purpose of this Appendix is to provide a brief review of the in-core experiments that have recently been carried out at the MITR as a supplement to the outline of the in-core capabilities given above. Although future experiments are not limited to the types of experiments that have been previously performed, these examples should serve to illustrate more fully the nature and scope of in-core research that can be contemplated. Furthermore, experiments that fall in or near the envelope of conditions previously investigated will benefit from the established techniques, safety analyses and, in some cases, existing equipment that can be re-used or adapted. Recent experiments will be described in somewhat greater detail because they are more likely to have these types of advantages.

A1.1 Overview of past experiments

Many of the in-core experiments at the MITR have been used for studies of commercial light water power reactor (LWR) technology. These include facilities to study: radioactive corrosion product transport in PWRs, specifically the effect of pH and zinc injection; BWR normal and hydrogen water chemistry and techniques to reduce radioactive nitrogen carryover into the steam phase; irradiation assisted stress corrosion cracking (IASCC) using actively loaded constant strain rate and constant load tests and passively loaded crack growth monitoring specimens; corrosion and mechanical property response of candidate ceramic fuel clad materials to LWR coolant and irradiation environments; dependence of shadow corrosion on base and counter materials, gap size and irradiation environment; the feasibility of developing internally and externally cooled annular fuel (IXAF) in order to increase the power density of PWRs while maintaining or improving safety margins, and real-time measurement of the thermal conductivity of liquid metal bonded uranium-zirconium hydride fuel during irradiation.

In addition to the LWR facilities, a high temperature irradiation facility was demonstrated that provides irradiation space for a variety of mechanical and thermal property specimens for post-irradiation examination. These specimens were exposed in-core at temperatures up to 1600 ºC. A generic In-Core Sample Assembly (ICSA) capsule design was developed and demonstrated for several types of experiments with temperatures up to 900°C.

A2. Pressurized LWR Loops

A2.1 Coolant Chemistry Loops

The PWR coolant chemistry loop (PCCL) and its companion BWR coolant chemistry loop (BCCL) are described in the following sections. They were among the earliest of the advanced in-core experiments to be carried out at the MITR. These loops were operated under prototypic pressure and temperature to enable the simulation of PWR or BWR conditions.

A2.1.1 PWR coolant chemistry loop (PCCL)

The PCCL was designed as an approximately 1/3 scale model of a single flow channel in a PWR. Figure A1 is a schematic of the PCCL. Its components included a core section of Zircaloy 4 tubing representative of the flow channel between fuel pins in a PWR core, a steam generator section of Inconel tubing representative of a single steam generator tube, and connecting sections representative of the recirculation piping. PWR coolant was circulated in the loop at representative velocities by a seal-less, magnetic drive pump. In-core heating was provided by an electric resistance heater coupled to the loop tubing by a lead bath that was predominantly liquid at operating temperature. Full operating temperature and pressure for the loop were maintained independent of the reactor power and the loop was designed to run for periods of up to 3 months at constant conditions.

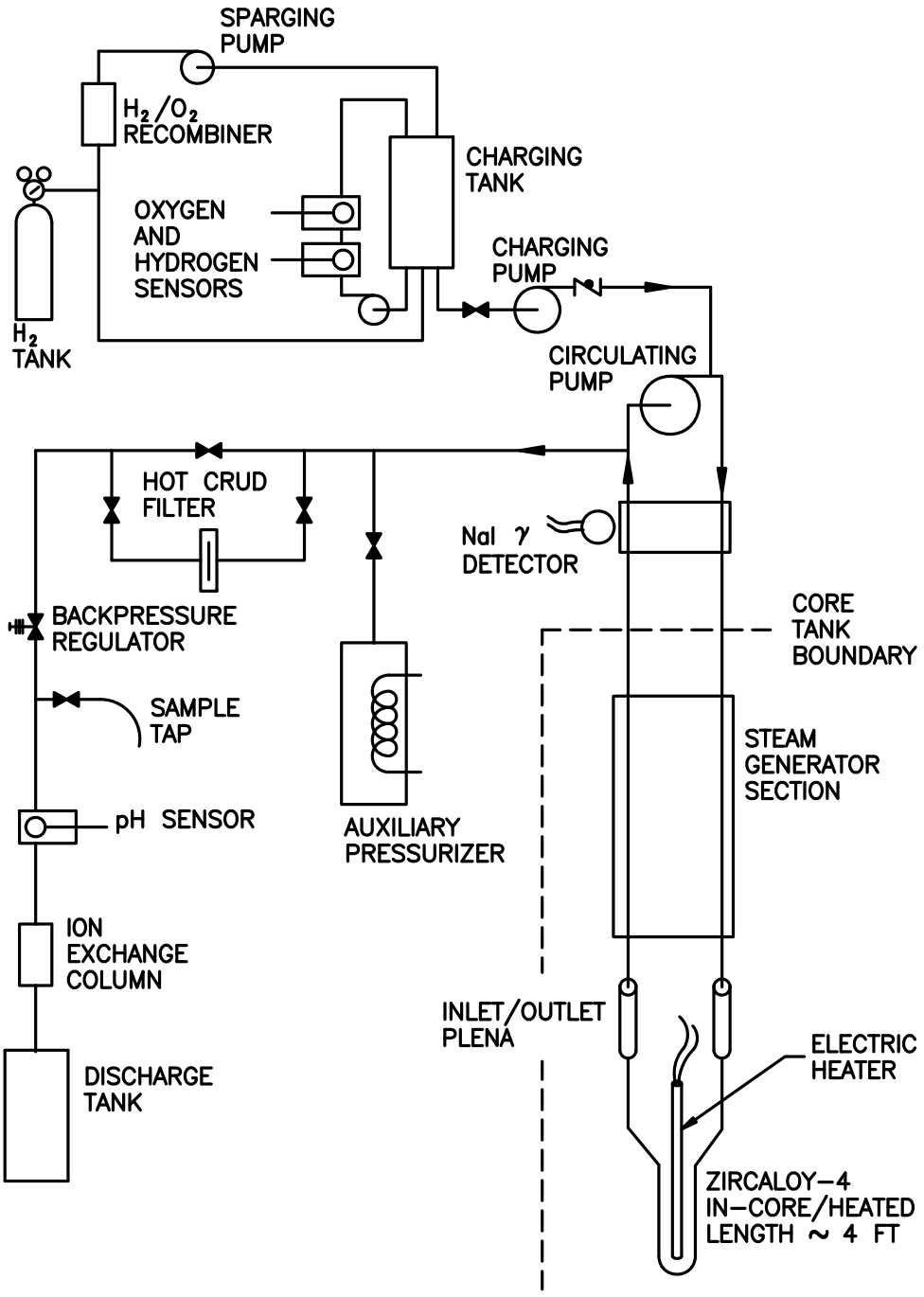


Figure A1. Schematic of the PCCL showing major components both in and out of core.

Essential parameters of the loop are summarized in Table A1. An important feature of the loop was that the bulk of the tubing was replaced for each run. This facilitated the performance of a series of closely controlled experiments where only the pH of the coolant was changed from run-to-run. The initial conditions for each run were virtually identical; there was no need to decontaminate the tubing between runs and the tubing itself was available for analysis after each run. Coolant chemistry was maintained at a constant boric acid and lithium hydroxide level for each run, with feed and bleed to the

loop set at a rate proportional to the total loop volume. Inlet dissolved hydrogen levels were controlled and monitored as was the inlet dissolved oxygen content of the coolant.

Table A1. Summary of parameter comparisons between the MIT PCCL and a representative PWR.

Parameter	Representative PWR	MIT PCCL
Core inlet T, ºC	285	277
Core exit T, ºC	320	316
Hydraulic dia., cm		
core	1.2	0.65
steam generator.	2.1	0.62
Boundary layer ΔT, ºC		
core	18	23
steam gen.	5.7	5.6
Shear Stress (relative)		
core	1	0.45
steam gen.	1	0.41
Core average neutron flux, n/cm^{2}sec		
Thermal	2.3×10^{13}	2.6×10^{13}
Fast (>1 MeV)	9.4×10^{13}	5.0×10^{13}
Material surface area ratios:		
Total Inconel/Zircaloy	2.73	5.11
Cooled Inconel/heated Zircaloy	3.44	2.79
Total SS/Inconel	0.15	0.24
Plenum SS/Inconel	--	0.10

In addition to the initial series of runs to study pH effects on radioactive corrosion product transport (seven one-month long runs) a series of longer runs were undertaken to investigate the possibility of reducing transition metal transport in PWR coolant using zinc injection.

A2.1.2 BWR coolant chemistry loop (BCCL)

The BCCL was also designed as an approximately 1/3 scale model of a single flow channel in a BWR. In this case, the parameters that govern BWR coolant radiolysis and the associated parameters of electrochemical corrosion potential (ECP) and nitrogen chemistry were simulated as closely as possible in the in-core loop. The loop operated at the same temperature and pressure as BWR core with capability for outlet quality up to

15%. Operating in single phase mode the loop could also be used to simulate bypass region conditions.

For the BCCL, makeup and letdown systems were designed to provide heated feedwater at a variety of conditions. Feedwater gas content (H_2 and O_2) was independently controllable and other dissolved species could be added through a separate injection system. The core exit stream was separated into steam and water phases that were independently assayed for N^{16} content using on-line gamma monitors. ECP was monitored in the steam separator plenum and in the liquid phase letdown line. A cooled sampling system was installed on the loop to allow core exit hydrogen peroxide concentration to be measured. Hydrogen peroxide decomposes rapidly by thermal processes at BWR temperatures in the bulk coolant and on sample line walls. The possibility of close access to the loop during reactor operation and the use of the sample cooler combined to provide a unique capability for accurate hydrogen peroxide measurements.

Several sets of BCCL experiments were carried out. The first were aimed at providing well characterized datasets to be used for benchmarking radiolysis codes. Many codes were semi-empirical in nature and had been optimized to predict parameters measured in operating plants. The availability of a test bed that was better characterized and more controllable than a commercial plant was of significant value in the improvement of the codes. Another set of experiments studied the possibility of achieving the benefits of hydrogen water chemistry (HWC) in a BWR without the concomitant increase in carryover of radioactive nitrogen into the steam line. Alternate ways to produce reducing coolant environments were evaluated, as was the possibility of injecting other species together with hydrogen to achieve reductions.

A2.2 Fuel Cladding Test Facilities

Two major types of in-core fuel cladding test have been carried out at the MITR. The first is designed to expose cladding samples to prototypical LWR coolant and neutronic environments. This facility is referred to as the Advanced Clad Irradiation (ACI) and has been used to test alumina and silicon carbide-based fiber composites. The facility is essentially the same in both cases and therefore the more recent implementation of the ACI for SiC composite candidate clad materials is described in Section A2.2.1. A separate facility used to study the phenomenon of shadow corrosion under BWR conditions is described in Section A2.2.2.

A2.2.1. Advanced Clad Irradiation facility (ACI)

The aluminum thimble for the ACI facility is shown installed in the MITR core tank in Figure 6 of Section 3. Within this thimble there is a titanium autoclave serving as the main pressure boundary, and within the autoclave there is a set of internal parts to position the specimens and direct the water flow. A cutaway view of these parts is shown at the top and bottom ends of the specimen stack in Figure A2. Water flow is downward around the return tube and the flow shrouds that isolate the specimens, and then back up

through the spacers and over the specimen surfaces. At the top of the specimen stack the water flow enters the return tube, which is fed through the autoclave pressure boundary at the top head of the autoclave. Dimensions of the in-core components are given in Table A2. Photographs of the in-core components are shown in Figure A3. It is generally possible, as was done in this experiment, to place specimens in a region immediately above the core as well as directly in core. This feature has been used in several experiments to directly study the effect of neutron irradiation on the relevant phenomena, because the in-core and above-core specimens are exposed to otherwise virtually identical coolant temperatures and chemical conditions. The ACI facility has space for 24 in-core and 15 above-core specimens about 0.45 inches OD by 2 inches long.

Table A2. Dimensions of the major ACI in-core components (inches).

Component	**Outside diameter**	**Wall thickness**	**Inside diameter**
Thimble	2.00	0.156	1.68
Autoclave	1.566	0.035	1.496
Flow Shroud	1.25	.016	1.22

The ACI facility is inserted through a port in the reactor top lid and into a dummy fuel element in the B-3 core positions. This system has the advantage that the high temperature and pressure coolant lines and thermocouples connect above the reactor top lid and are accessible for assembly and maintenance. These lines are routed to the circulating pump and heater adjacent to the reactor top. Shielding is required to reduce personnel dose, principally from the ^{16}N activity in the circulating coolant. A photograph showing the autoclave top head and connections, and the coolant lines, prior to installation of insulation and shielding is shown in Figure A4.

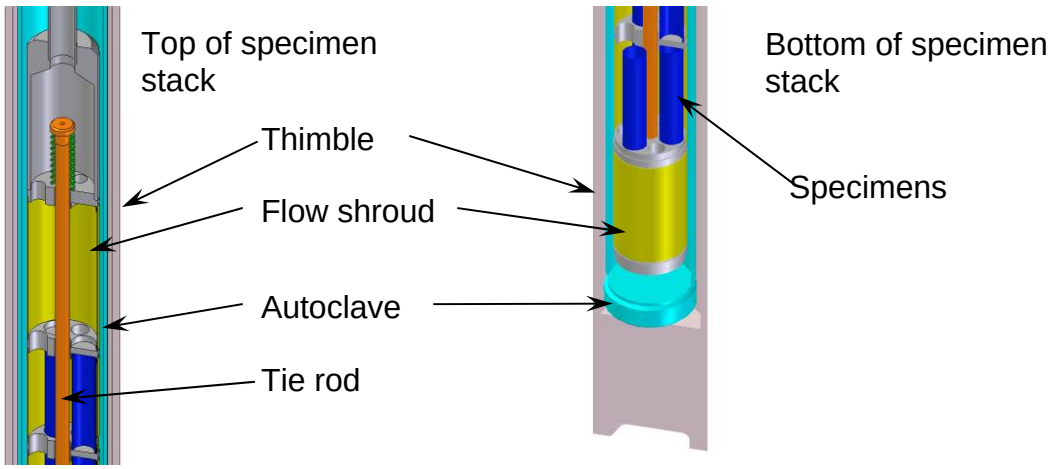


Figure A2. Cutaway models of the ACI at the top and bottom of the specimen stack.

Figure A3. Photographs of a sample module during assembly and the fully assembled sample stack before insertion into the autoclave.

Figure A4. The ACI top head and coolant lines after facility installation in the MITR and during installation of the thermal insulation and shielding.

The coolant conditions used for the SiC clad test in the ACI were chosen to represent mid-cycle conditions in a typical PWR. The coolant temperature was about 300±2 ºC at the specimen location and was controlled at that point independent of the reactor power level. In this facility, with relatively low in-core mass, the in-core heating is relatively small and the temperature change from specimen inlet to outlet is therefore also small. The coolant contained 800 ppm of boron as boric acid and 2.2 ppm of lithium as lithium hydroxide with a hydrogen content of 20 cc/kg. These parameters were generally kept within 5% of the target value throughout the run. Inlet oxygen concentration was monitored and remained at less than 2 ppb throughout the run.

In addition to the parameters above, inlet and outlet coolant conductivity was monitored throughout the run and periodic batch coolant samples were obtained for assay of radiochemical content using an HPGe gamma spectroscopy system.

A2.2.2 Shadow corrosion test facility

The shadow corrosion test facility uses essentially the same out-of-core system as the ACI but has been operated under BWR coolant conditions with no lithium hydroxide or boric acid addition. Both normal water chemistry (NWC) and hydrogen water chemistry (HWC) are used in the loop. This facility is also similar to the ACI in that it is installed through the reactor top lid into a B-ring position at the reactor and is designed to expose a set of specimens to LWR conditions including a representative in-core irradiation environment. The internals for the shadow corrosion facility are shown in Figure A5.

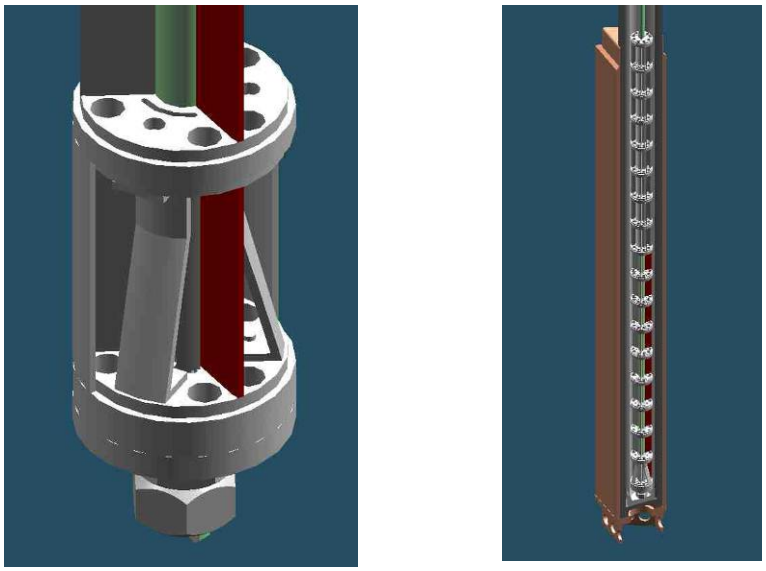


Figure A5. Cutaway 3D model views of the shadow corrosion experiment internals. A close-up view of a variable gap shadow module is on the left while the right shows the entire in-core sample stack and the dummy fuel element.

The samples in this experiment are short segments of clad tubing that are exposed to “counter” materials with a varying gap along the length of the specimen. Following

irradiation, the samples are removed from the rig and the oxide thickness is evaluated using eddy current measurement techniques. By scanning circumferentially and axially, the dependence of oxide thickness on counter material type and gap thickness can be determined.

A2.3 Irradiation Assisted Stress Corrosion Cracking Studies

This section describes three facilities that were used to study various aspects of the problem of irradiation assisted stress corrosion cracking. The first of these is a unique facility that provided for slow strain rate testing of un-irradiated or pre-irradiated mechanical property test specimens in-core under BWR conditions. The second is an adaptation of this facility for a long-term constant load test of multiple specimens. The third is a facility that provided for irradiation of passively loaded crack growth sensors and accompanying ECP electrodes. These facilities are not currently available at the MITR, so only brief descriptions are given here to provide a sense of the scope of research that can be performed.

A2.3.1 Slow strain rate test (SSRT) facility

The slow strain rate test facility used a commercially available ball-screw type test machine installed on the reactor top lid to actively load a specimen in an in-core autoclave in a B-ring position. The autoclave was operated under BWR conditions using the same out-of-pile system used for the ACI and shadow corrosion tests. Both NWC and HWC conditions were used during testing. The mechanical load on the specimen was applied through a dynamic seal at the reactor top lid and transmitted to the specimen using a pull rod and reaction tube system that extended from the reactor top to the in-core portion of the autoclave. Tests were performed at constant cross-head speed with strain rates sufficient to produce failure of the specimens in a period of 10-20 days. Both thermally sensitized (un-irradiated) specimens and specimens pre-irradiated in an inert gas environment at about 300 °C to a dose of about 1 dpa were tested. Specimen change-out was performed in a reactor floor hot cell. Fractured specimens were removed from the hot cell, sectioned to reduce background activity and examined by SEM to assess intergranular fracture percentages.

A2.3.2 Constant load test facility

The constant load test facility used the same test machine, autoclave and other major components as the SSRT facility. In this case, however, the internals were designed to accommodate a train of up to nine specimens over the active core height. All of these specimens were loaded by the test machine to the same load that was maintained for a period of months, again at BWR conditions. The objective of this test was to emphasize the effect of water chemistry on the cracking behavior of specimens by using a long-duration test at lower loads than was the case in the SSRT. The specimen train was designed to allow continuation of testing after failure of one more specimens without the need to remove the failed specimen. The design also permitted the identification of a failed specimen at the time of failure.

A2.3.3 Crack growth sensor test

This test was an approach to studying IASCC that was particularly aimed at demonstrating the ability of HWC to arrest the progress of a growing crack. To accomplish this, a set of double cantilever beam (DCB) crack growth monitors of a type adapted from those used in BWR core and recirculation piping tests was installed in an in-core autoclave. Electrochemical corrosion potential (ECP) electrodes were placed in the same autoclave to determine the ECP changes associated with coolant chemistry changes and the associated changes in crack growth behavior. The DCB specimens were pre-cracked and passively loaded with direct current potential drop measurements used to assess crack growth *in situ*. After operation at NWC to establish crack growth in one or more specimens, coolant chemistry was switched to HWC while crack growth rates continued to be monitored. At the end of the irradiation period the in-core portion of the autoclave was stored in the wet storage ring in the reactor core tank to allow for decay of the highly radioactive stainless steel specimens. The specimens were then shipped off-site for post irradiation examination.

A3. High Temperature Irradiation Facility (HTIF)

The HTIF was designed, installed and operated for a period of several months in order to demonstrate the capability for this type of irradiation in the MITR. Such a facility is an important test bed for irradiation testing of materials essential to the development of high temperature gas reactors. The demonstration test achieved temperatures up to 1600 °C in an in-core location with fast neutron flux of about 1 x 10^{14} n/cm^2-s. A variety of materials relevant to high temperature gas reactor design, including SiC, AGR matrix graphite, and non-fueled coated particles, were irradiated.

A3.1 Facility design

Figure A6 is a cutaway view of a solid model of the HTIF showing the basic components of its construction. As with other in-core experiments, the facility is installed in an in-core position after removal of a solid dummy fuel element. In this case, the experiment dummy element is an integral part of the facility. The rig is comprised of a set of tungsten sample holders with graphite-lined sample spaces designed to hold a variety of samples. For the demonstration experiment, mechanical property test specimens of two types and thermal conductivity test specimens were accommodated. In addition, one sample holder was configured to hold un-fuelled TRISO particles. Tungsten was chosen because of its high density and good high temperature properties; the large in-core mass provides a large heat input from the nuclear heating. By thermally insulating the tungsten sample holders from the aluminum dummy element using a heat shield and gas gaps, high temperatures were achieved in the facility. As can be seen in Figure A6, insulation is achieved using a Mo heat shield that creates two circumferential gas gaps. These gas gaps are filled with a helium/neon mixture in order to provide some active control over the temperature in the rig. The thermal conductivity of helium is approximately ten times that of neon. Thus increasing the helium fraction in the cover gas decreases the temperature while increasing the neon fraction increases the temperature. At the temperatures

achieved in the facility, a large part of the heat loss from the sample stack is due to radiative cooling and is not affected by the gas composition.

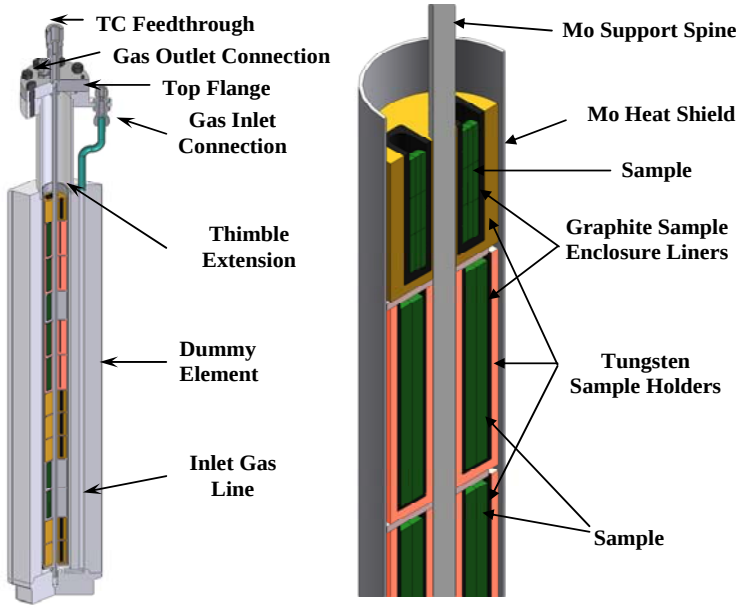


Figure A6. Cutaway views of a solid model of the complete in-core test section and a close-up of individual sample modules.

More details of the construction of the facility can be seen in Figures A7 and A8. Figure A7 shows some of the actual tungsten sample holders, manufactured by the wire EDM technique because of the difficulty of machining pure tungsten. End caps were EDM'ed separately and electron beam welded to the holder bodies. In addition to the sample cavities visible in the sample holders, there is a central hole for the Mo support spine and holes for thermocouples. Thermocouples were installed both within the tungsten holders and in the center of the tubular support spine. Figure A8 shows components of the rig prior to assembly. Graphite liners are visible in the holder on the bottom left. One of the alumina spacers that were used to thermally insulate the holders from each other and to center the sample holders within the heat shield can also be seen in this picture.

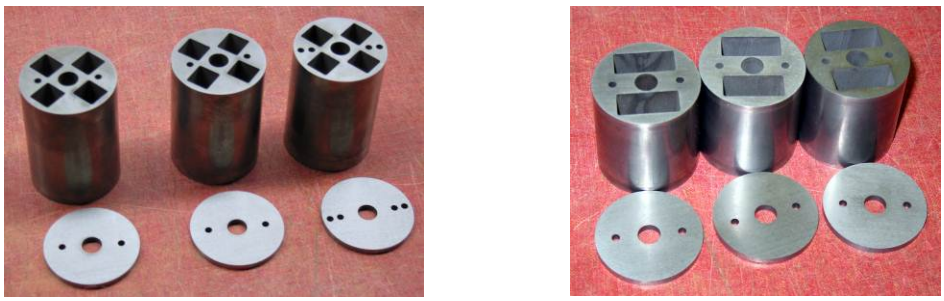


Figure A7. HTIF tungsten sample holder bodies and end caps prior to e-beam welding.

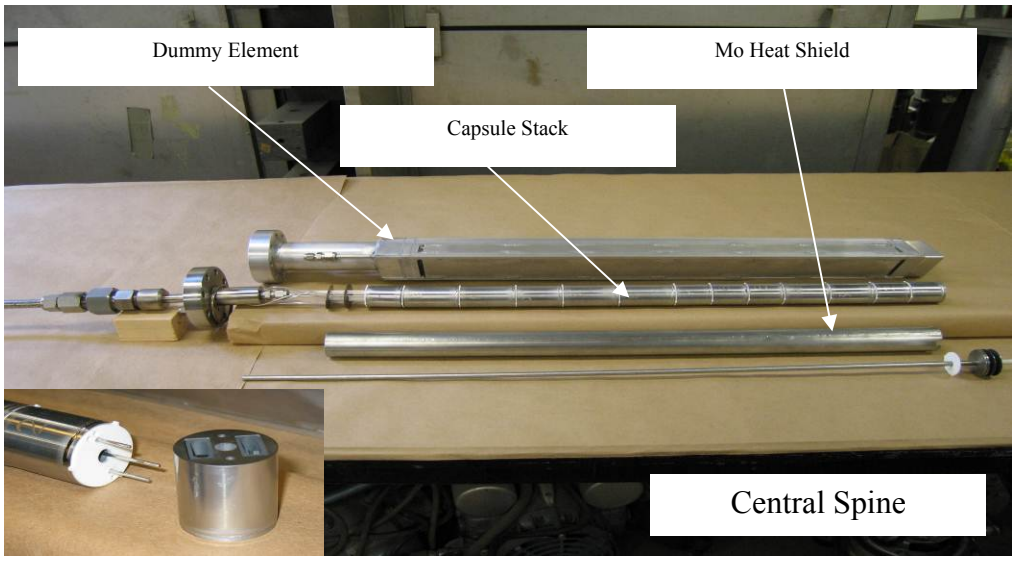


Figure A8. Photographs of the main HTIF components prior to assembly.

A3.2 Conditions and results of the demonstration test

Figure A9 gives the calculated axial flux distribution for the HTIF for neutron energies greater than 1.0 MeV and greater than 0.1 MeV. This axial flux shape drives an axial variation in the heat input to the rig and thus creates an axial temperature profile. It is possible to flatten the temperature profile by adjusting the mass in the capsules to counteract the effect of the flux profile. In order to achieve maximum temperatures, however, this technique was not applied for the demonstration test. Figure A10 gives calculated and measured temperatures in the center capsule of the HTIF as a function of reactor power and cover gas composition. A major uncertainty in the thermal calculations at high temperature is the emissivity of the capsule and heat shield surfaces. For the calculation, this parameter was adjusted to match observed temperatures at one condition (pure helium cover gas, reactor power 3 MW) to allow prediction for other conditions. Temperature measurements in the central spine did not vary significantly from those within the sample holders, as predicted by finite element modeling.

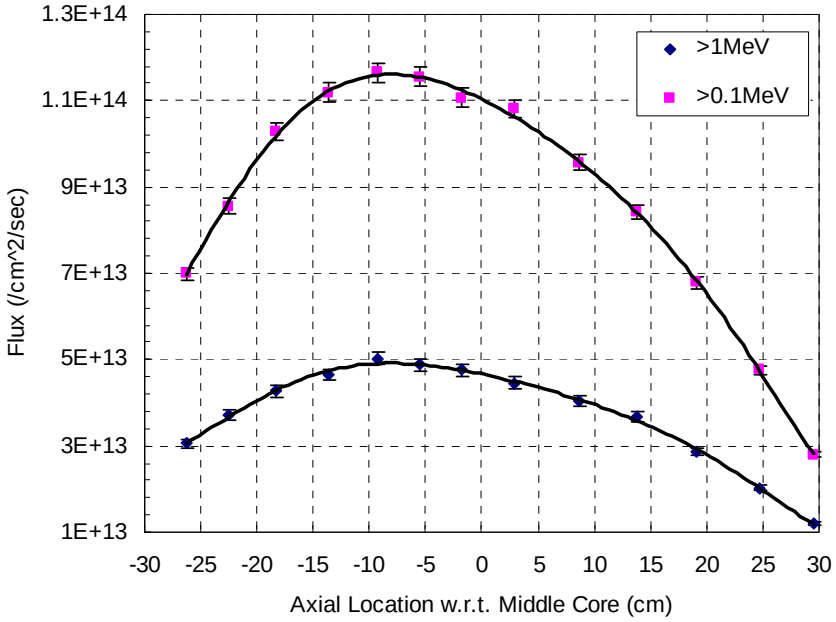


Figure A9. Axial neutron flux profile in the HTIF.

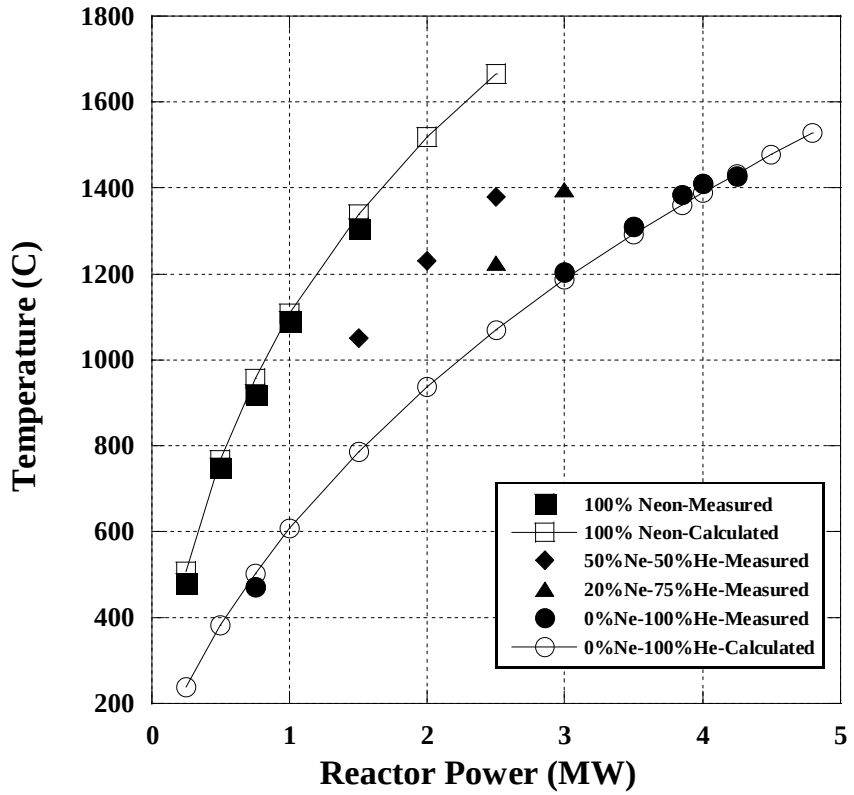


Figure A10. Calculated and measured temperatures in the HTIF as a function of reactor power and cover gas composition.

A4. In-core Fissile Materials Irradiation Experiments

A license amendment was obtained in 2003 for the MITR to conduct irradiation of fissile material in an experimental facility. Since that time, two fuelled irradiations have been performed at the MITR, as described below.

There are a number of constraints on a fissile material irradiation experiment established by the Nuclear Regulatory Commission definition of a research reactor, by the license amendment authorizing these experiments and by the technical specifications relevant to all in-core experiments at the MITR. These are: the fuel may not be contained in a "circulating loop through the core", the fissile content is limited to 100 g U^{235} or equivalent, the fissile material must be doubly encapsulated, the reactivity worth of the experiment may not exceed 1.8 %ΔK/K (2.3 beta), onset of nucleate boiling (ONB) on surfaces exposed to the MITR primary coolant must be avoided, and provision must be made to sample for fission product gas at least weekly during the irradiation and for redundant automatic over-temperature reactor scrams. Due to these constraints, the focus of such experiments at MITR is to investigate the basic properties of advanced nuclear fuels at lower burnup using small aggregates of fissile material and is intended to complement the full-scale, accelerated burnup fuel evaluation programs at test reactors.

A4.1 Annular Fuel Irradiation

Internally and externally cooled annular fuel (IXAF) has the potential to allow significant increases in LWR power density. As part of a broad study of the design, manufacturing, performance and economics of IXAF, two samples containing 5% enriched vibration packed (VIPAC) fuel were irradiated in the MITR. The goals of the irradiation test were to support the analytical and modeling effort by providing data on the actual performance of IXAF, at least at low burnup, and to serve as a demonstration that high power density annular fuel could be operated at the temperatures necessary for application in PWRs. The irradiation test itself consisted of two phases, an initial short-term test of a single fuel sample to demonstrate the capability of the facility, followed by approximately six months of irradiation of the initial sample and one additional specimen.

Avoiding the use of a separate circulating coolant loop for a fuelled experiment has a large impact on the design, particularly when high power density is desired. MITR reactor coolant has a core outlet temperature of about 50 °C and close to atmospheric pressure. It is therefore necessary to provide a heat transfer gap that will transfer the power generated by the fuel at a temperature differential of about 250-270 °C in order to operate the fuel at the temperatures expected in commercial application. This was achieved using the sample capsule design shown in Figure A11. The IXAF fuel samples were contained in aluminum sample capsules using a lead-bismuth eutectic alloy (LBE) for the heat transfer gaps. An instrumentation and cover gas tube is provided.

Thermocouples were routed through this tube to measure surface temperatures on the inner and outer clad surfaces of the IXAF sample. The tubes were also used for periodic sampling of the CO_2 cover gas to measure gaseous fission product levels.

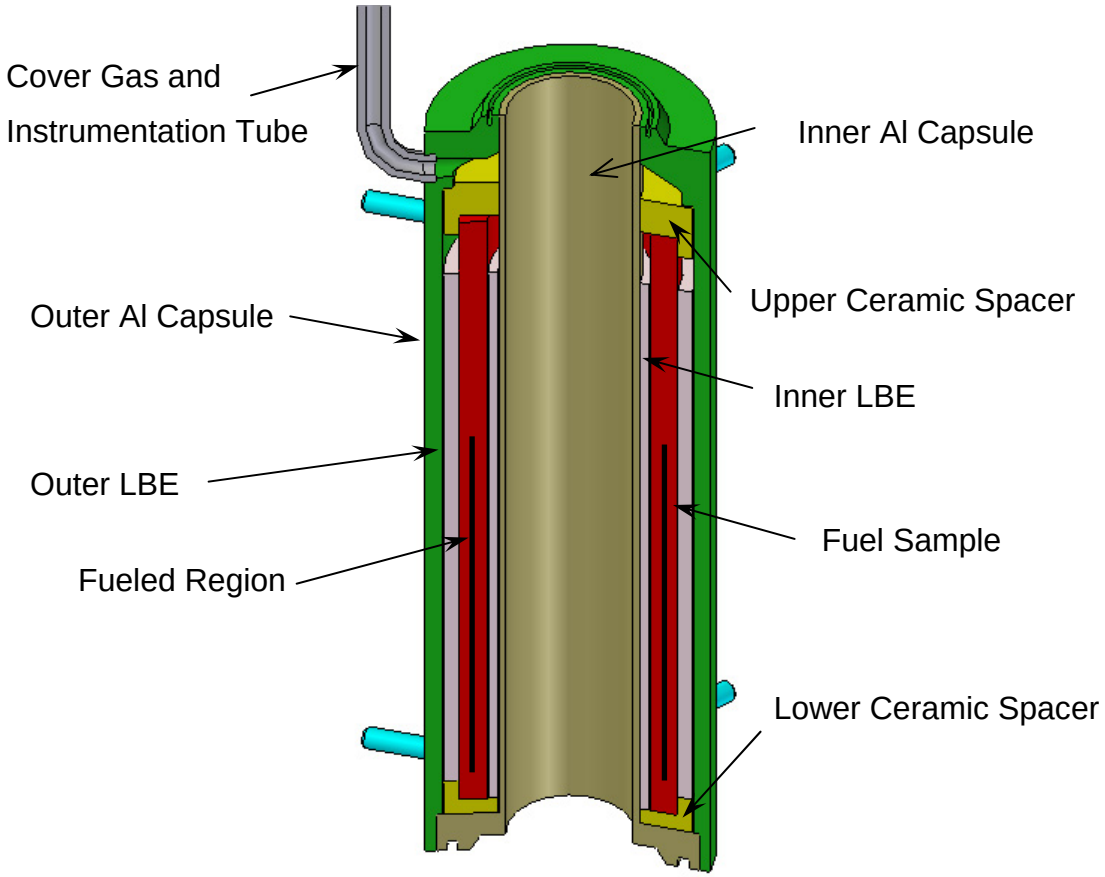


Figure A11. Fuel sample irradiation capsule design. The vertical line in the fuel sample indicates the approximate height of the fuel region within the clad fuel sample.

The linear power for VIPAC test fuel samples as a function of position in the MITR core was calculated using an MCNP model of the core and the results are shown in Figure A12. Note that most of the neutrons causing fission in the test fuel arise from the MITR reactor fuel and this gives rise to the axial variation observed in linear power. A linear heat generation rate of 110 kW/m was targeted for the sample fuel. Based on the volume ratio of the sample to the reference commercial fuel this is equivalent to 50 kW/m in the commercial fuel, which is representative of the highest power fuel pin for a core operating at 150% of the rated power of a core based on a current 17x17 Westinghouse fuel assembly. The calculated linear heat rates for the VIPAC samples in the core center are close to this target. The samples at core top and bottom are below this value because lower neutron fluxes in these regions. In the actual test, only two fuelled capsules were used and were installed at the top two positions. Figures A13 is a cutaway view of annular fuel irradiation capsules. Figure A14 is a photograph of the annual fuel experiment installed in the core.

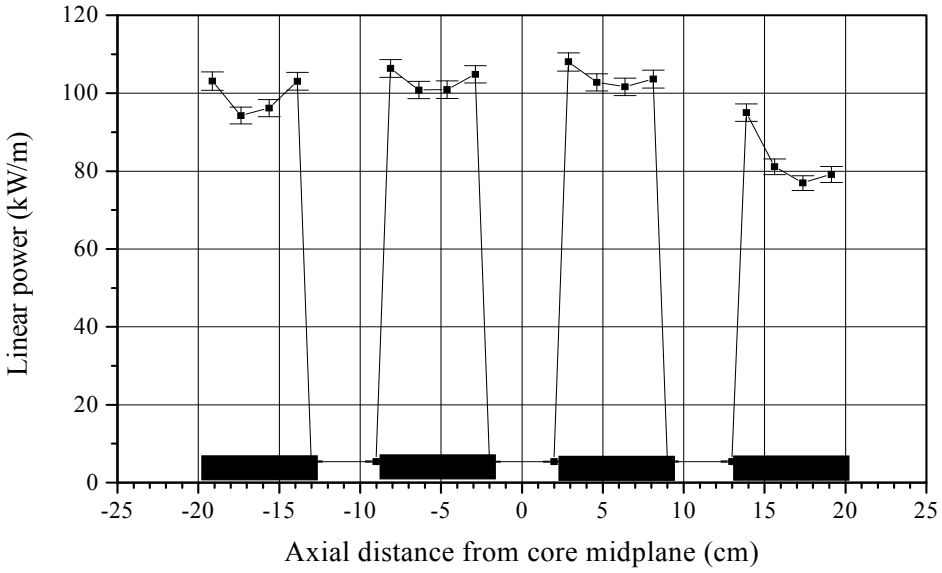


Figure A12. Linear power calculated using MCNP for four VIPAC fuel samples at various axial locations in the MITR.

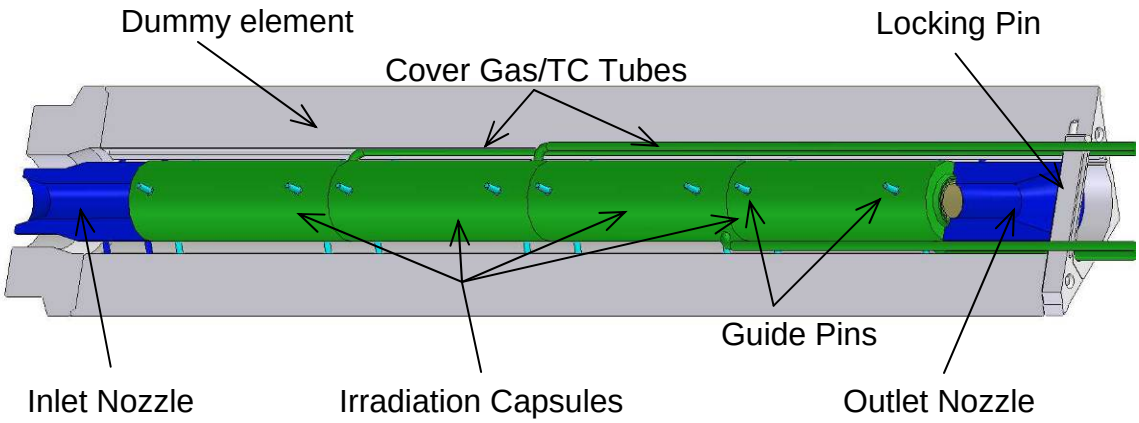


Figure A13. Cutaway view of four irradiation capsules installed in the dummy element with inlet and outlet nozzles.

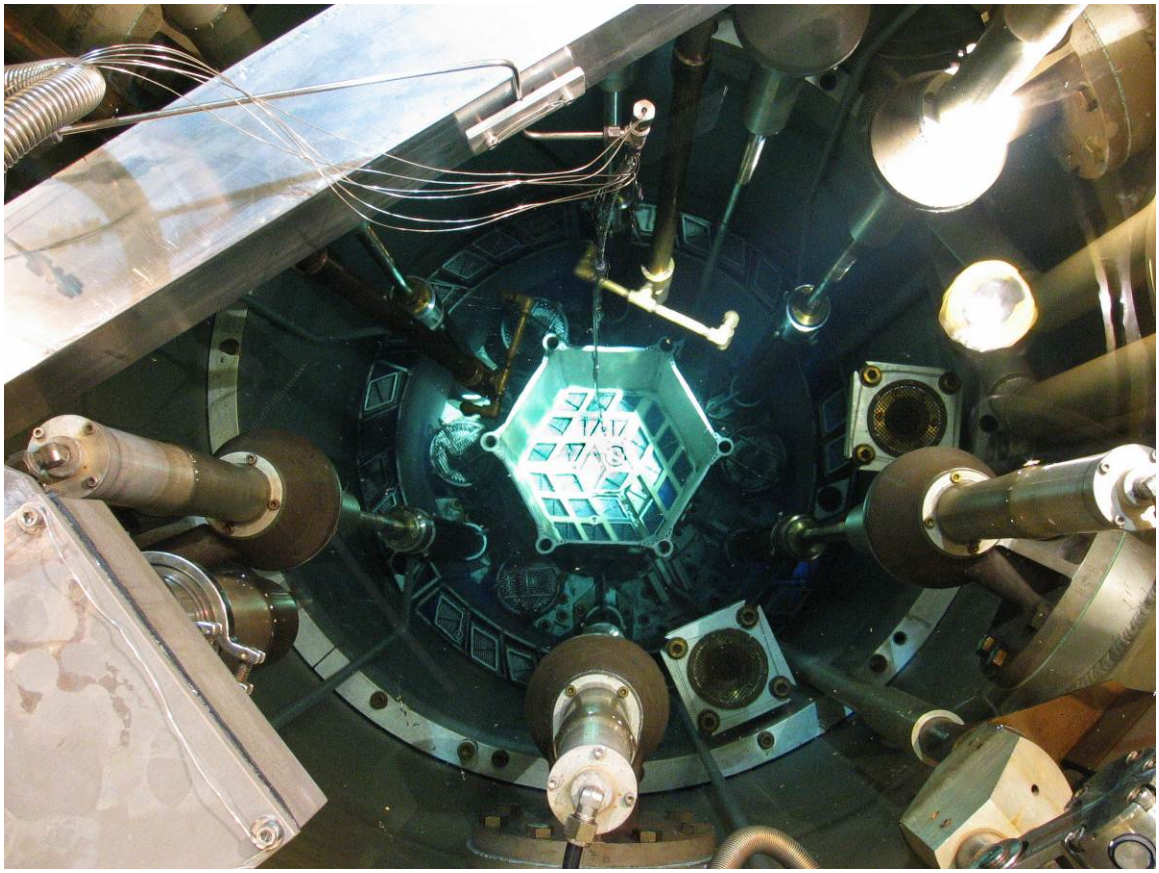


Figure A14. Photograph of the annular fuel rig installed in the MITR core.

A4.2 Hydride Fuel Irradiation

The Hydride Fuel Irradiation (HYFI) experiment is an irradiation of (U, Zr)H_X metallic fuel pellets in zircaloy cladding in a dedicated in-core facility. The fuel rod design and experiment conditions are meant to approximate the typical operating conditions (geometry, flux, and temperature) of a short length of LWR fuel. This fuel is of interest because of its higher thermal conductivity (reducing fuel temperatures) and favorable neutronics properties (built-in moderator and stronger negative Doppler feedback). The data from the experiment will reveal how the conductivity changes with irradiation.

Of additional interest is the fuel-cladding interaction, and in particular the possible transport of hydrogen out of the fuel and into the cladding, which may cause cladding degradation during irradiation. To attempt to ameliorate this interaction, the gap between the fuel and cladding is filled with lead bismuth eutectic (in place of traditional helium cover gas), which provides an excellent thermal bond and possibly chemical protection to the cladding inner surface. The cladding surfaces are also pre-oxidized to help slow hydrogen diffusion.

Three HYFI fuel rods were assembled into capsules and irradiated in the MIT reactor. The hydride fuel rod is constructed from zircaloy cladding with U(30 wt%)-$ZrH_{1.6}$ fuel pellets. The approximate dimensions are as follows: fuel pellet diameter is 0.950 cm by 1

cm height and 5 pellets are included in one fuel rod. The fuel/cladding gap is about 125 μm and is filled with lead-bismuth eutectic (LBE). Zircaloy cladding inner and outer diameters are 0.975 and 1.116 cm, respectively and the total height of the zircaloy cladding for one mini fuel rod is 7.5 cm. Figure A15 shows cross-sections of the fuel rod, titanium capsule, and loaded dummy element, while a single loaded HYFI capsule is shown in Figure A16.

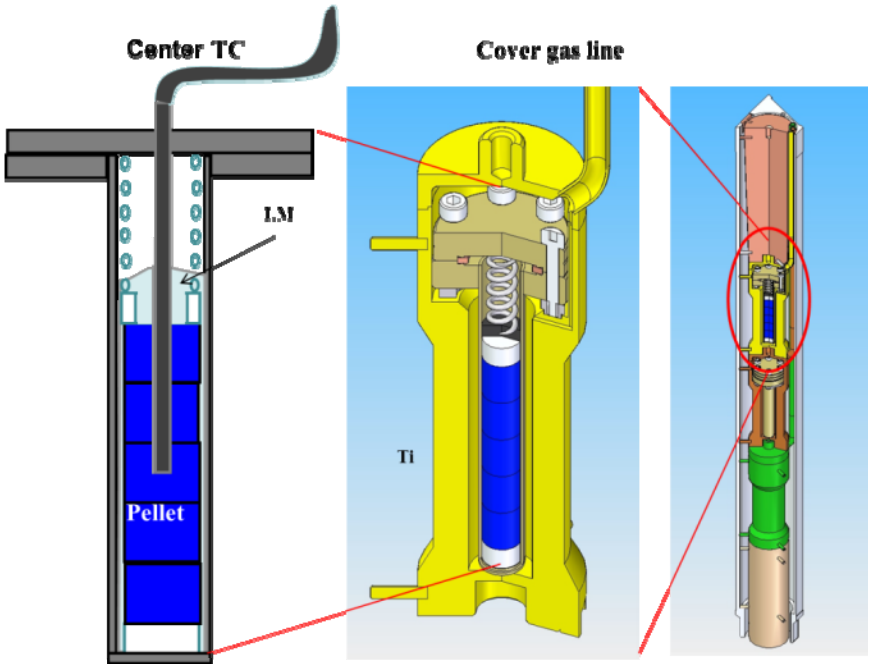


Figure A15. HYFI fuel rod, capsule, and loaded dummy element cross sections.

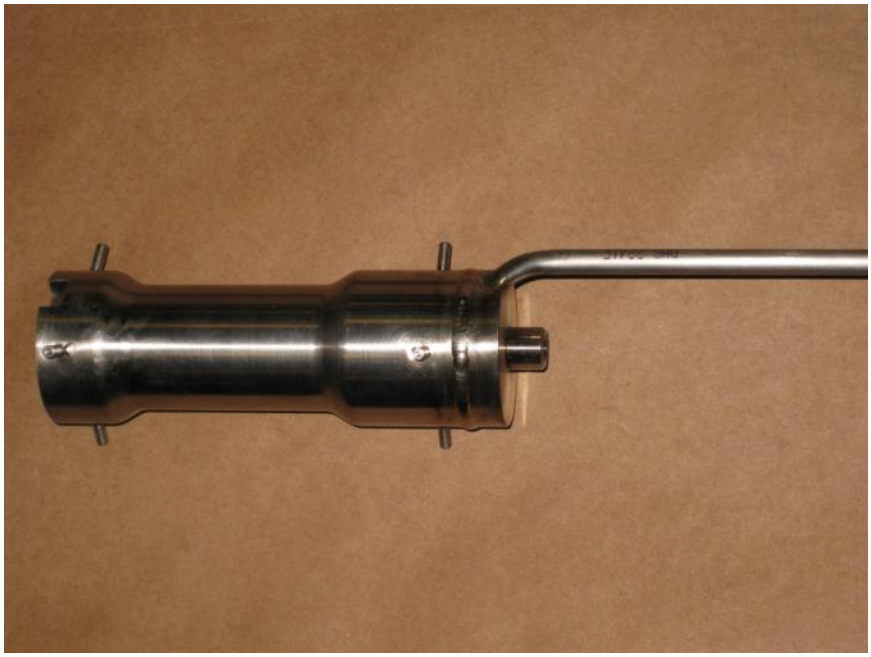


Figure A16. HYFI capsule ready for insertion in core.

Because of the MIT reactor license requirements, heat from the hydride fuel must be rejected to the primary coolant. In order to achieve typical LWR temperatures in the experiment (and provide a secondary encapsulation of the fuel), the hydride rods were enclosed in titanium outer capsules. The titanium outer capsule accommodates a mini fuel rod with a LBE-filled gap between the cladding and the outer capsule. The top of the capsule has a gas space that can be pressurized with helium and then regularly sampled in order to check for the presence of fission gas release from the fuel rod.

Up to three fuel capsules are stacked in an aluminum dummy element which forms an annular water flow channel between the fuel sample capsule and the dummy element, as shown in Figure A17. The dimensions of the titanium outer capsule (wall thickness) and the LBE gap size in conjunction with the heat generation rates in the fuel and the capsule control the experiment temperatures. The thermal design objectives for this experiment are as follows: maximum fuel centerline temperature of 650 °C, capsule outer surface temperature below saturation (107 °C) and minimum LBE gap temperature above the LBE melting point (125 °C). Furthermore, the gap width of the annular coolant channel between the outer capsule and the aluminum dummy element is determined so the primary flow rate through this experiment is adequate in maintaining the temperature constraints described above.

Figure A17. HYFI dummy element and capsule stack.

The dimensions of the capsule-LBE gap, titanium capsule thickness, and flow channel width in the fueled region were determined through an iterative process in the thermal hydraulic analysis, and these are 2.5 mm, 8.445 mm, and 11.413 mm, respectively. Each capsule also provides for instrumentation with thermocouples at the fuel centerline and on the cladding outer surface. The instrumentation tube also provides for a controlled gas atmosphere in the small gas space above the conducting gap and alumina insulator.

Up to three capsules were irradiated simultaneously in the MITR-II core. The capsules were placed close to the mid-plane position in the core with the top and bottom positions

occupied by dummy capsules. Note that the experiment design allows for independent replacement of individual fuel element irradiation capsules. The installed experiment and above-core support equipment are shown in Figures A18 and A19.

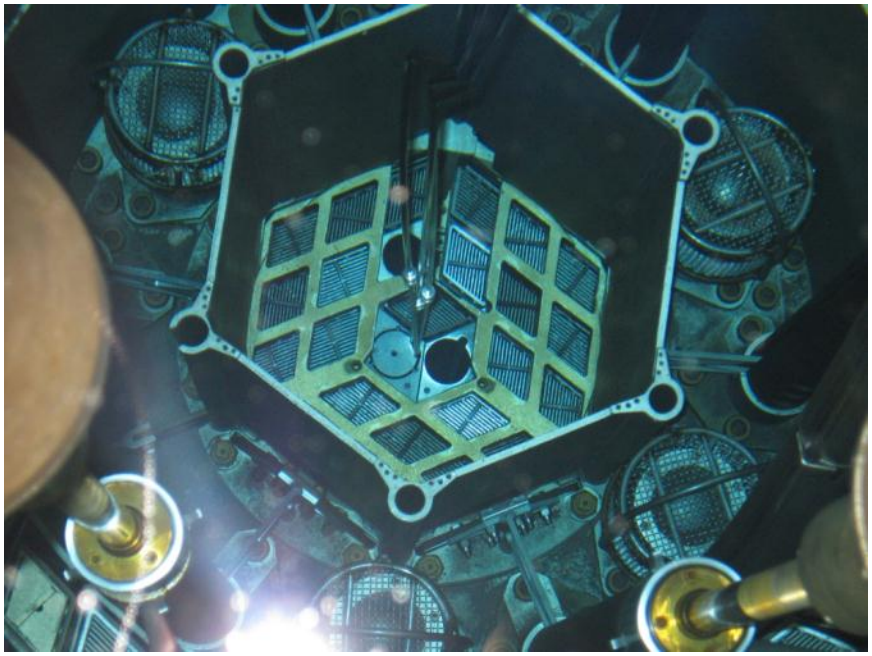


Figure A18. HYFI installed in core position A-1.

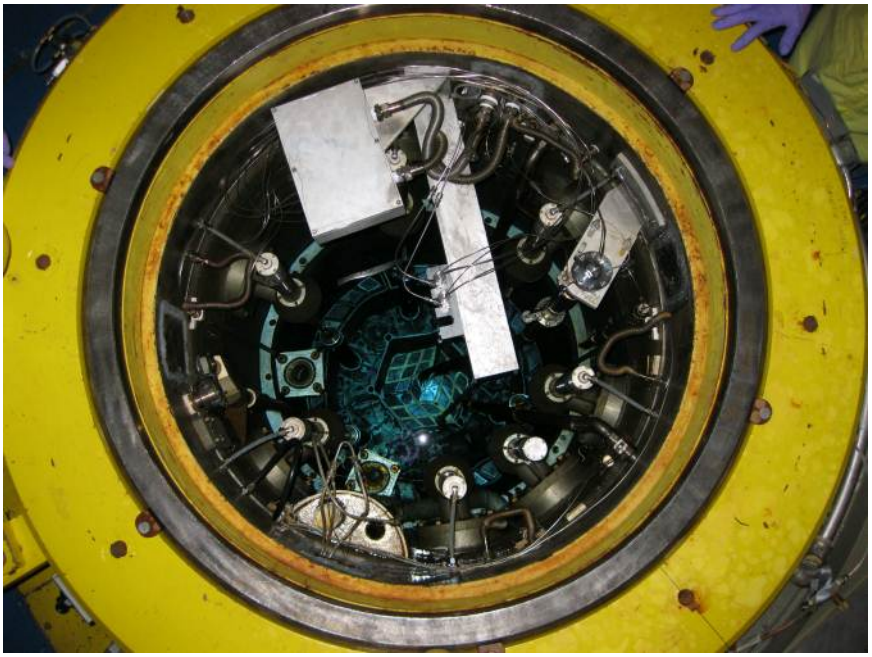


Figure A19. View of HYFI gas tube and instrumentation bridge below the reactor top lid.

A5. Inert Gas Irradiations in the ICSA

The In-Core Sample Assembly (ICSA) facility provides space for in-core irradiations isolated from the water coolant with an inert cover gas and aluminum thimble. In general, experiments in the ICSA desiring elevated temperatures utilize gamma heating (heat produced by gamma radiation from the fuel interacting with high-Z materials. Because the heat rejection from samples inside the ICSA thimble occurs across the inert cover gas gap, the conductivity of the gas mixture can be used, along with reactor power, to control sample temperatures. Figure A20 shows an experimental verification of the control of temperature in the ICSA as a function of reactor power and the supplied helium/neon inert gas mixture.

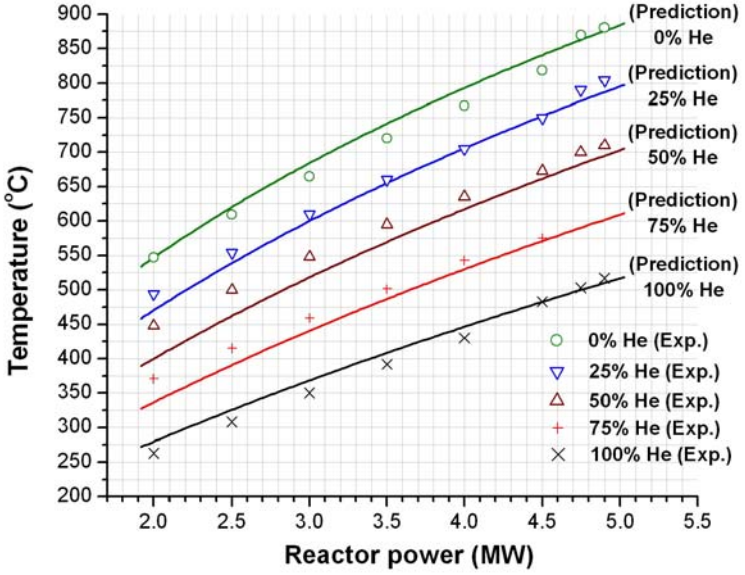


Figure A20. 2D schematic of Mo ICSA irradiation used for FLUENT CFD thermal analysis.

A5.1 Molybdenum Metal Irradiations

The objective of the molybdenum metal irradiation was to perform an in-core irradiation experiment on a realistically sized prototype Mo target. The Mo-99 activity produced from this irradiation was also used as validation of the MITR's MCNP model in order to estimate the production capacity of an optimized Mo-99 irradiation facility in the MITR. The irradiation was performed using the in-core sample assembly (ICSA) installed in a center position (A-3) of the MITR core.

A thermal analysis was performed using the FLUENT CFD code with a 2D, axisymmetric model of two Mo test capsules as shown in Figure A21. This analysis is to confirm that target temperature is below 900°C, which was previously approved by the MITR Safeguard Committee for the ICSA facility. Heat transfer is predominantly in the radial direction, and the gas gap (0.09″) is modeled as 100% helium with an outer thimble temperature of 50ºC. Gamma heating rates of titanium, graphite and Moly target were

calculated using MCNP based on 4.9 MW reactor power and are 1.58, 1.30, 1.56 W/gm, respectively. The temperature contours of the ICSA module are shown in Figure A22. No significant axial temperature gradient is expected in the target, and the predicted peak temperature is about 730°C.

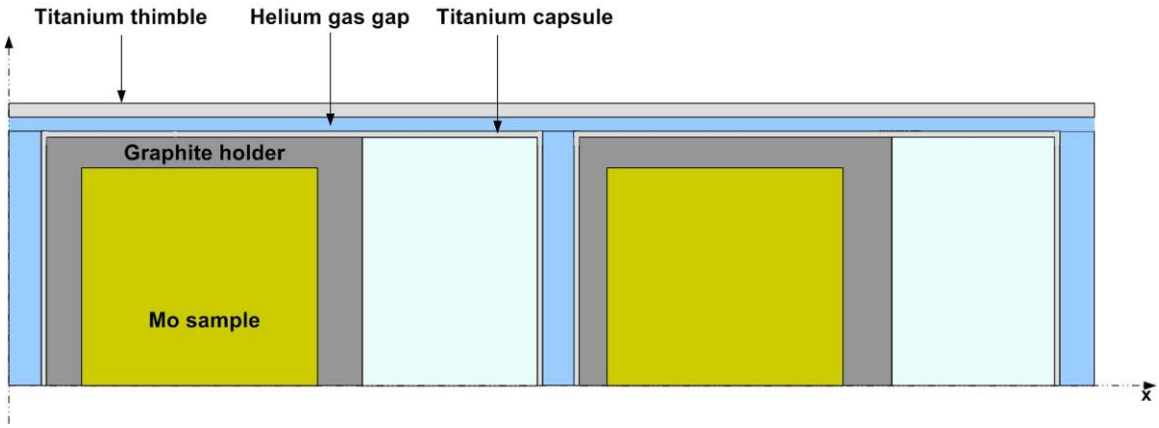


Figure A21. 2D schematic of Mo ICSA irradiation used for FLUENT CFD thermal analysis.

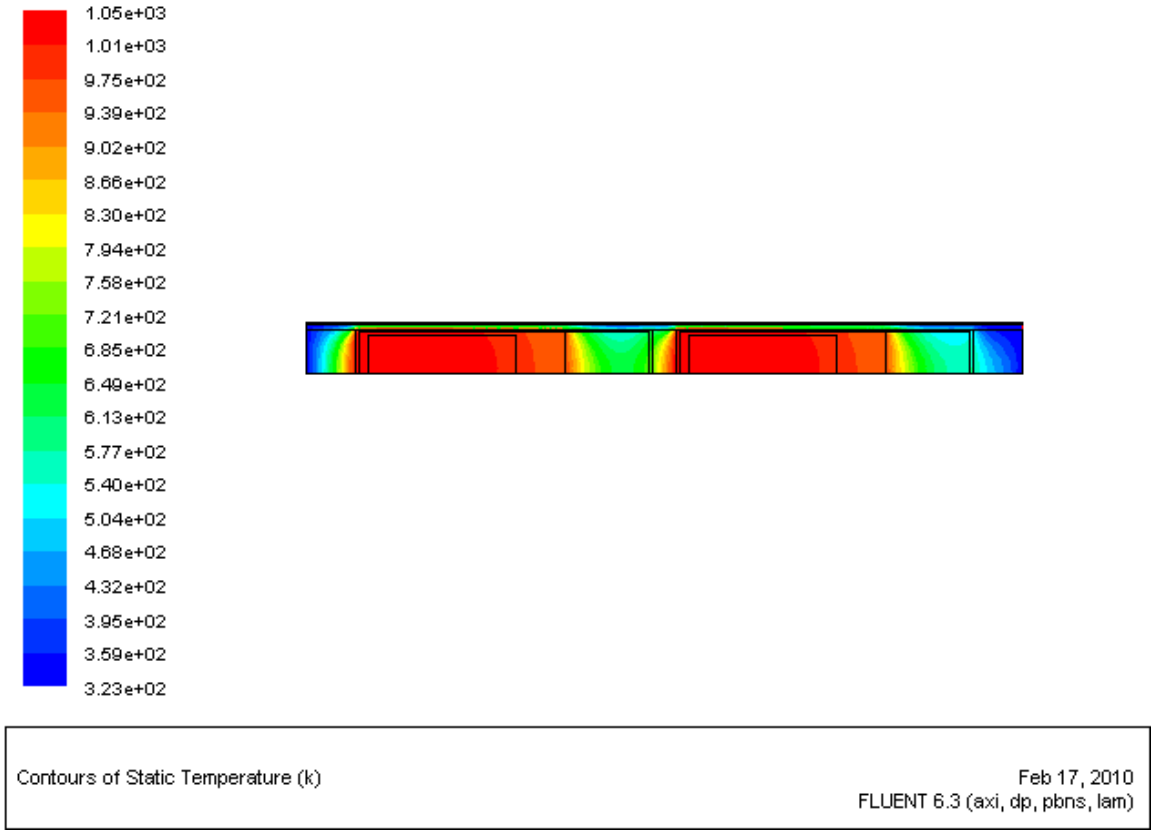


Figure A22. Calculated temperature contours in the Mo ICSA irradiation.

The metal Moly targets are 1″ or 1.5″ OD cylindrical disks. A scoping MCNP analysis indicated that the specific activities of Mo-99 of 1” and 1.5″ disks are approximately the

same, therefore the ICSA Moly irradiation capsule design is based on 1.5″ disks. Due to significant gamma heating in the Moly metal targets, a high purity graphite annulus was inserted to increase the thermal conduction between the Moly target and titanium capsule. Each capsule holds about 3″ of Moly disks. Figure A23 shows a typical capsule with samples, which were un-instrumented.

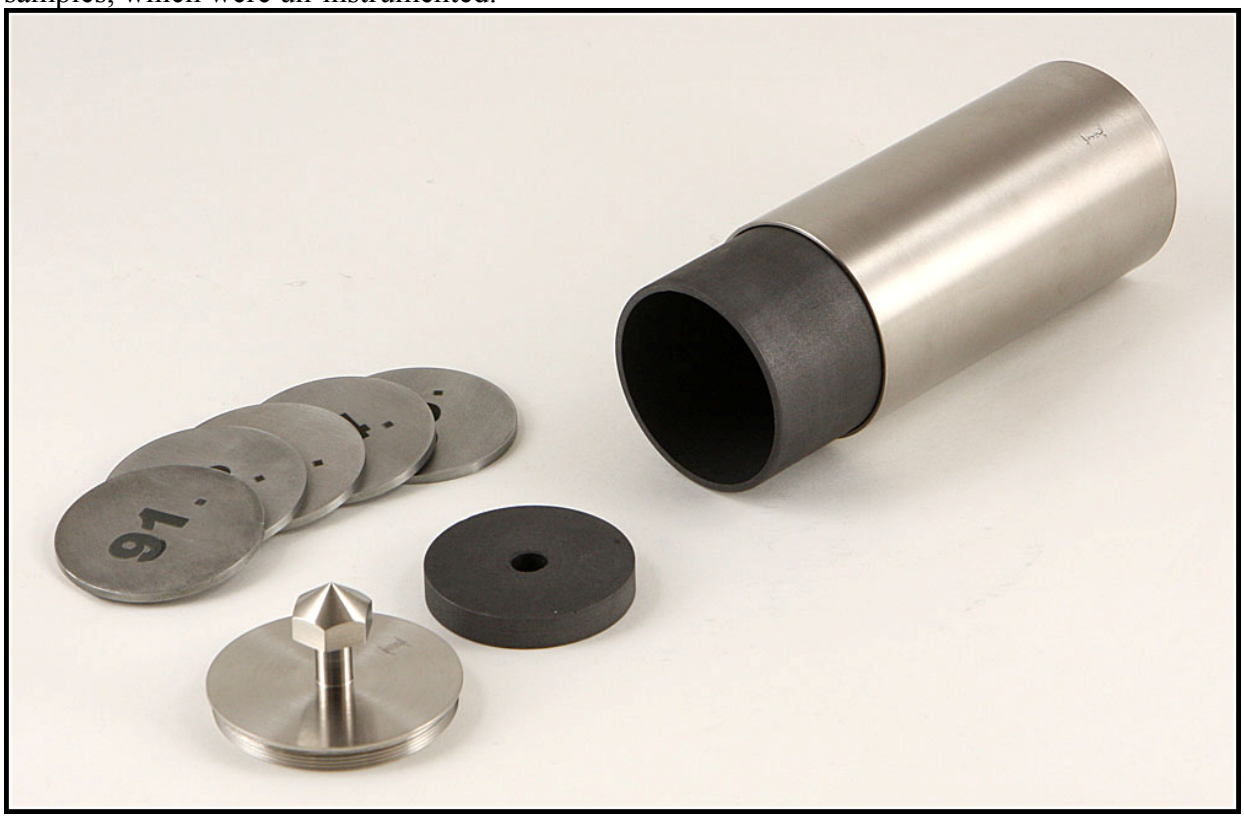


Figure A23. ICSA capsule with graphite spacer, liner, and Mo metal disks.

A5.2 MAX Phase Materials Irradiation

MAX phase materials are ternary compounds that are of interest because they can combine the properties of metals and ceramics in a single structure. MAX phases are defined as having the formula $M_{n+1}AX_n$ where M is an early transition metal, A is an A-group element, X is carbon or nitrogen, and n = 1-3.

For this irradiation in the ICSA, specimens were provided consisting of seven MAX phase materials in three shapes: tensile bars (TB), resistivity bars (RB), and thermal diffusivity disks (TD). An example of one material set is shown in Figure A24. Each irradiation capsule carried all of the types of specimens, and consisted of 28 TB, 21 RB, and 35 TD. In total six capsules (168 TB, 126 RB, and 210 TD) were prepared for irradiation at two different temperatures (300 and 600°C) and for three irradiation periods (3, 6, and 9 months).

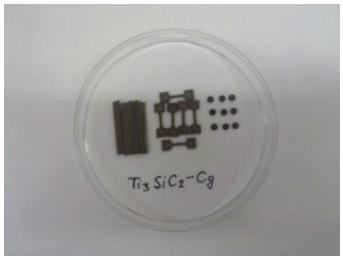


Figure A24. One set of MAX phase specimens sorted before irradiation.

During each irradiation period, four titanium capsules are used, each with sample packages inserted in a high purity graphite holder. To provide a high temperature environment, titanium gamma susceptors are included in the two High-Temperature (HT) capsules (shown in Figure A25), which are located in the vicinity of the core centerline, i.e. the highest neutron flux region. In addition, the low temperature environment can be achieved by excluding the gamma susceptors from the two Low-Temperature (LT) capsules, which are positioned in the lowest flux region (bottom and top end of the in-core position). At the start of the irradiation, four modules were loaded into the ICSA thimble; the two modules in the upper region were taken out after 3 months of irradiation. Then two fresh modules (again one HT and one LT module) were loaded to start the 6-month irradiation and the two remaining modules in the bottom region continued for a total of 9-months irradiation.

Figure A25. An empty ICSA MAX phase irradiation capsule (HT) before irradiation.

Titanium capsules for this irradiation test are similar to those used in the Mo irradiation. A HT titanium capsule is a hollow cylinder of 1.625″ OD, 0.055″ thick, and 4″ long with a titanium susceptor of 4″ long, 1.495″ OD, and 0.185″ thick, whereas a LT titanium capsule has dimensions of 1.75″ OD, 0.031″ thick, and 3.875″ long without any susceptor material. The HT capsule has a larger helium gas gap size (0.1505″) than the LT capsule (0.0.09″) so that a higher temperature is maintained in the HT capsule and lower temperature in the LT capsule. In order to control high and low temperatures over the broad irradiation field, the capsules were exposed to a helium and neon gases mixture through the ICSA gas control system. One thermocouple per capsule was used to measure the temperature in the capsule as well as the compound sample.

A thermal analysis was performed using the FLUENT CFD code on an idealized, 2D, axisymmetric model of a four-capsule stack as shown in Figure A26. Heat transfer is predominantly in the radial direction (y-direction in the figure) and this was verified through a recent ICSA demonstration. In order to achieve the high temperature environment, a larger gas gap is modeled for the HT capsules as compared to the LT capsules. The FLUENT calculation was based on 100% helium in the gap. The inlet mass flow rate was set as minimal value of 10^{-5} kg/sec, so that most of the heat transfer occurs via the gas gap to the ICSA thimble that is held at a constant 50°C. Note that helium gas properties were input in the form of polynomials obtained using the NIST database. Heating rates of irradiation components were calculated using the MCNP code on the basis of 5.0 MW reactor power.

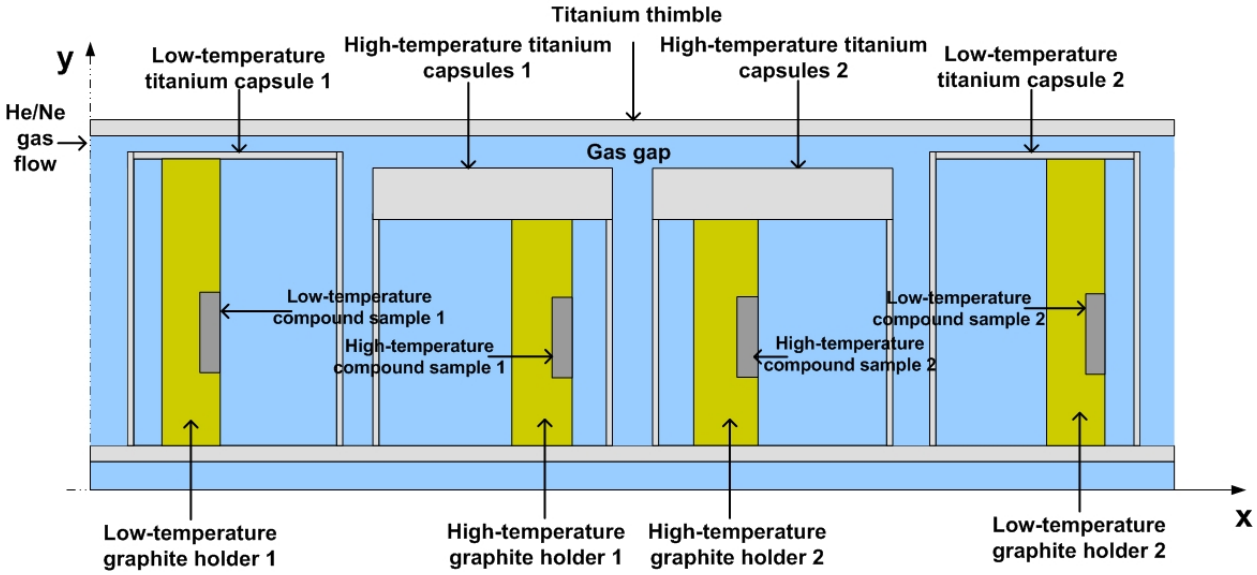


Figure A26. 2D model of the ICSA MAX phase irradiation for FLUENT CFD thermal analysis.

The irradiated materials can be secured safely because their melting points are much higher than the expected high temperature of about 650°C (see Figure A27). Furthermore, it was demonstrated that this ICSA with titanium capsule can be controlled safely at 900°C. Note that each HT titanium capsule was modeled as four segments to incorporate the axial gamma heating rates.

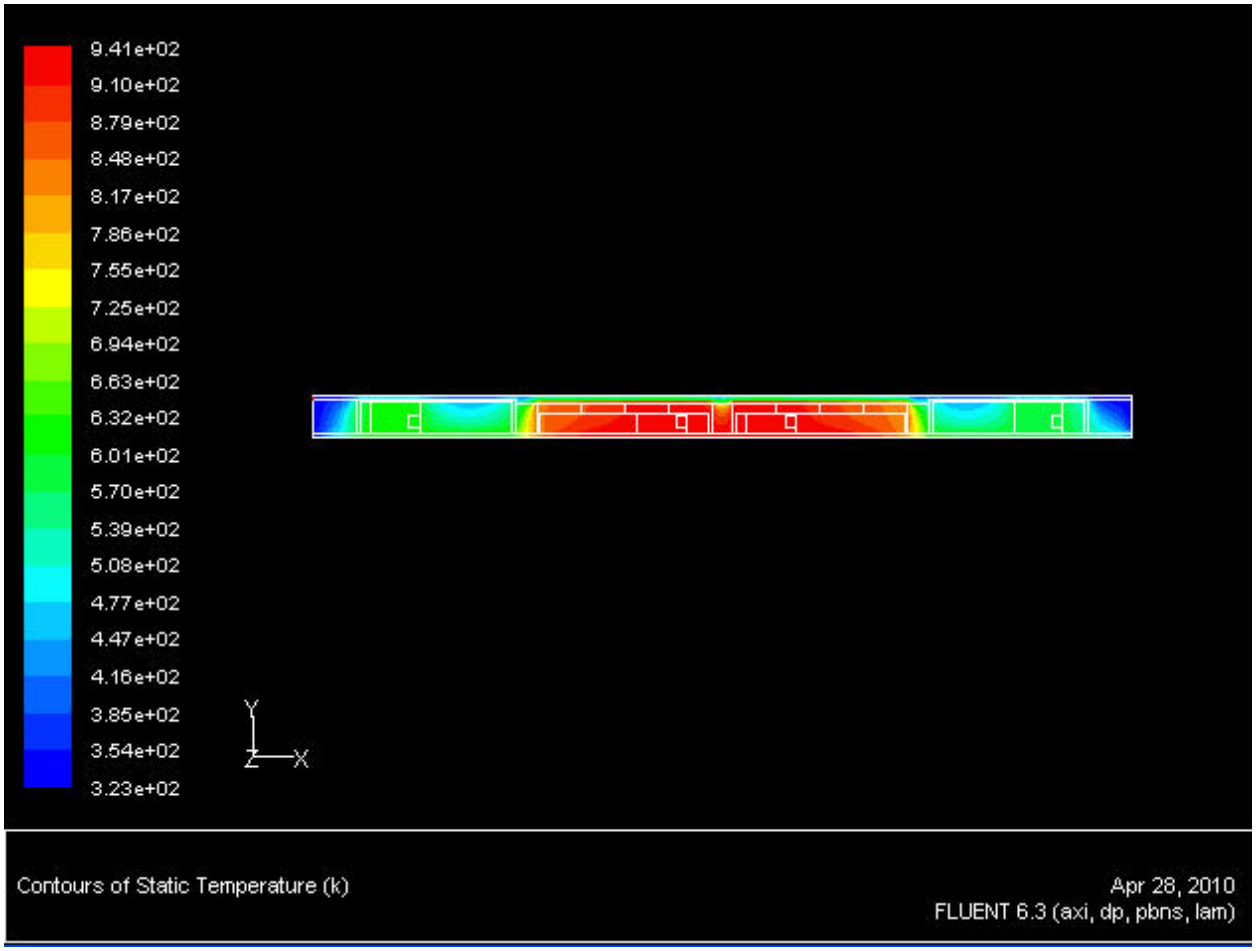


Figure A27. Results of FLUENT CFD thermal analysis for the MAX phase irradiation.

A5.3 Fiber Optic Sensor Irradiation

There has been recent interest in the use of fiber optic sensors for temperature and stress measurements. In an experiment design similar the previous MAX phase irradiations, this experiment involved the irradiation of short lengths of fiber optic sensors in the ICSA facility

Optical fiber sensors were inserted into 3.04" long metal tubes made of Nb-1% Zr. The optical sensor itself is a 3.9" long cylinder made of quartz. Six sensor assemblies were irradiated in the highest neutron flux region of the ICSA thimble. The target temperature for this irradiation ranged from 600 to 700°C and the irradiation duration was about 1 month at 6 MW reactor power. The capsule assembly design replicates the same materials and engineering features used in the previous ICSA Mo and MAX phase experiments with minor changes to the capsule dimensions and sample holders to achieve the desired operating temperatures of 600-700 °C with 100% helium gas.

The titanium capsule is a hollow cylinder 1.625″ OD by 0.055″ wall thickness, with an annular graphite holder of 0.85 ID and 1.515 OD. The capsule and holder are shown in Figure A28. The total length of the capsule is 4.375" inches. The gap size between the

titanium capsule and thimble is 0.1505″. Nominal temperatures are achieved using 100% He cover gas but in practice a small amount of Ne is added to allow for automatic control of the temperature at a constant level over a moderate range of reactor powers. Two commercially available K-type thermocouples were used to measure the capsule and sample temperatures.

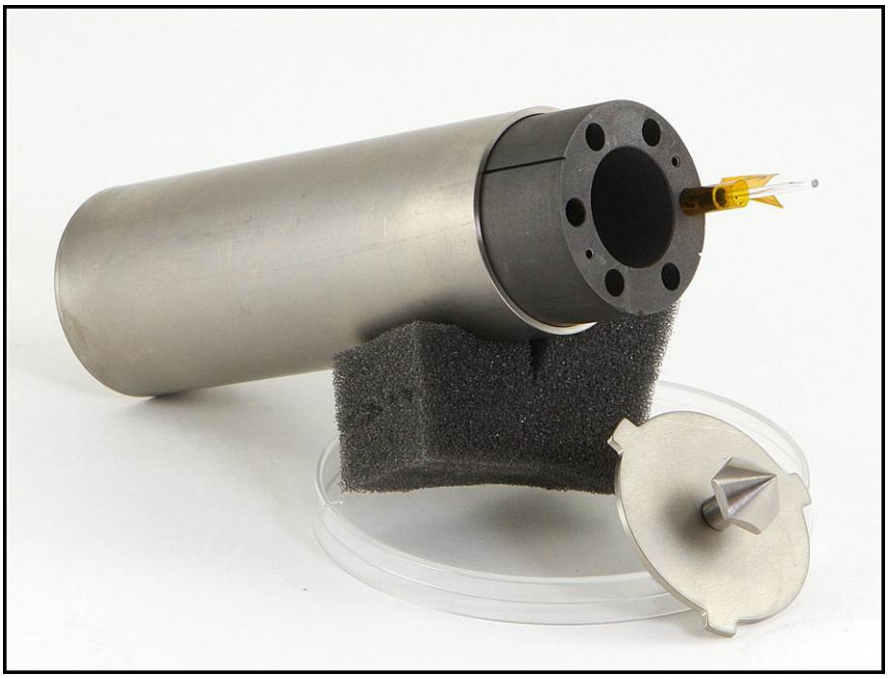


Figure A28. Titanium capsule and graphite holder with one fiber optic sensor loaded for ICSA irradiation.

Thermal analysis was used to confirm that the capsule and sensor assemblies would be within the previously approved temperature limit of 900 °C for the ICSA, and there would be an ample safety margin between the proposed operating temperature (600~700°C) and the melting temperature of the materials. This thermal analysis was performed using the FLUENT CFD code on an idealized, 2D, axisymmetric model of two titanium capsule spacers and one titanium sample capsule as shown in Figure A29.

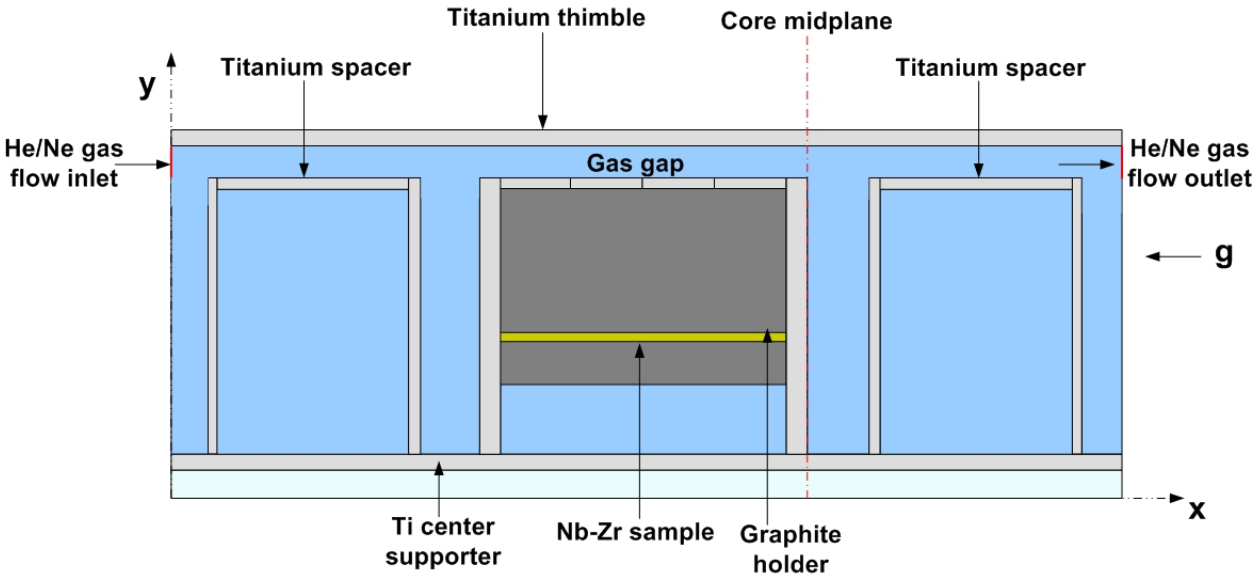


Figure A29. 2D model used for FLUENT CFD thermal analysis of the fiber optic sensor irradiation in the ICSA.

Heat transfer is predominantly in the radial direction (y-direction in the figure). The FLUENT calculation was performed with 100% helium in the gas gap. The inlet mass flow rate was set to a minimal value of 10^{-5} kg/sec (100 cc/min He), so that most of the heat transfer occurs via conduction through the gas gap to the ICSA thimble that is held at a constant 45 ºC primary water flow. Heating rates of irradiation components were calculated using the MCNP code based on 6.0 MW reactor power. As shown in Figure A30, there is an ample margin between the proposed operating temperature of 600~700°C and melting temperatures of the components.

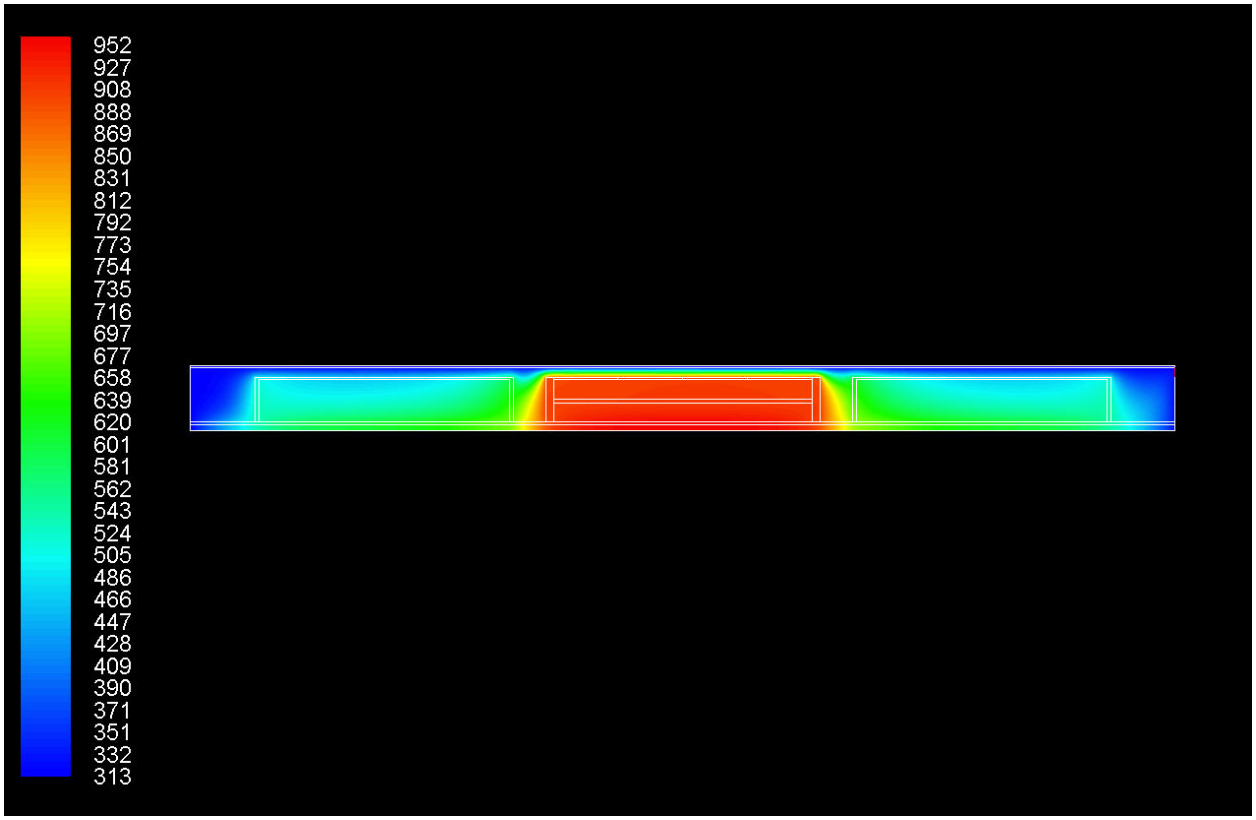


Figure A30. Results of 2D FLUENT thermal analysis of fiber optic sensor irradiation in the ICSA.

The total power generated in the ICSA assembly during this experiment is about 1.575 kW at a reactor power of 6.0 MW with a peak heat flux through the capsule outer surface of approximately 50 kW/m^2, which is close to the heat flux of MAX phase experiment HT capsule. The peak heat flux through the thimble outer surface is then 41.2 kW/m^2. The peak temperature of about 680°C is seen in the innermost area of the capsule close to the titanium center supporter. In the sample region, the expected peak temperature is 637°C. It is predicted that axial temperature profile is reasonably uniform as the temperature gradient is only about 6°C through the sensor region.