THE IN-CORE EXPERIMENTAL PROGRAM AT THE MIT RESEARCH REACTOR

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Abstract

This paper describes the program of in-core experiments at the Massacusetts Institute of Technology Research Reactor (MITR), a 6 MW research reactor. The MITR has a neutron flux and spectrum similar to those in water-cooled power reactors and therefore provides a useful test environment for materials and fuels research. In-core facilities include: a water loop operating at pressurized water or boiling water reactor conditions, an inert gas irradiation facility operating at temperature up to 850 °C and special purpose facilities including fuel irradiation experiments. Recent and ongoing tests include: water loop investigations of corrosion and thermal and mechanical property evolution of SiC/SiC composites for fuel cladding, irradiation of advanced materials and in-core sensors at elevated temperatures, irradiation in molten fluoride salt at 700 °C of metal alloy, graphite and composite materials for power reactor applications and instrumented irradiations of metal-bonded hydride fuel.

1. Introduction

The Massachusetts Institute of Technology (MIT) Research Reactor (MITR) is a 6 MW, tank-type reactor operated by the interdepartmental Nuclear Reactor Laboratory (NRL). The reactor is located on the MIT campus in Cambridge, Massachusetts, USA. It is the second research reactor to operate at the site and achieved first criticality in 1974. The MITR received a new 20-year license with power uprate from 5MW to 6 MW in November 2010. The reactor is a partner facility of the Advanced Test Reactor National Scientific User Facility, the US Department of Energy's designated nuclear energy user facility. Activities at the MITR include teaching and training, medical irradiations, neutron activation analysis, neutron imaging and beam optics experiments and a very active program in materials and fuels development using a variety of in-core experimental facilities. The in-core experimental program is the focus of this paper.

The MITR is a very useful test bed for materials and fuels experiments relevant to nuclear energy systems because it has an in-core neutron flux and spectrum similar to those of typical light water power reactors. The core consists of twenty-seven fuel element positions of which three are available for installation of in-core experiments as shown in the photograph in Figure 1. These irradiation positions are approximately 5 cm in diameter with a useful irradiation height of about 40 cm. The top of the core shown in Figure 1 is approximately 3 m below the coolant surface and 3.5 m below the reactor lid upper surface. Connections to in-core experiments are typically made through the lid or the upper section of the core tank, which is approximately 2 m in diameter. The area adjacent to the upper core tank is accessible even during full power reactor operation, facilitating the operation and maintenance of in-core facilities. This paper discusses the three types of facilities that are operated in-core: a pressurized water loop, a general purpose inert gas irradiation facility that can operate up to 850 °C, and special-purpose in-core facilities that are designed to extend the in-core experimental envelope beyond what is achievable in the more standardized facilities. The general characteristics of each facility type are described and examples of recent, ongoing and planned experiments are briefly discussed.

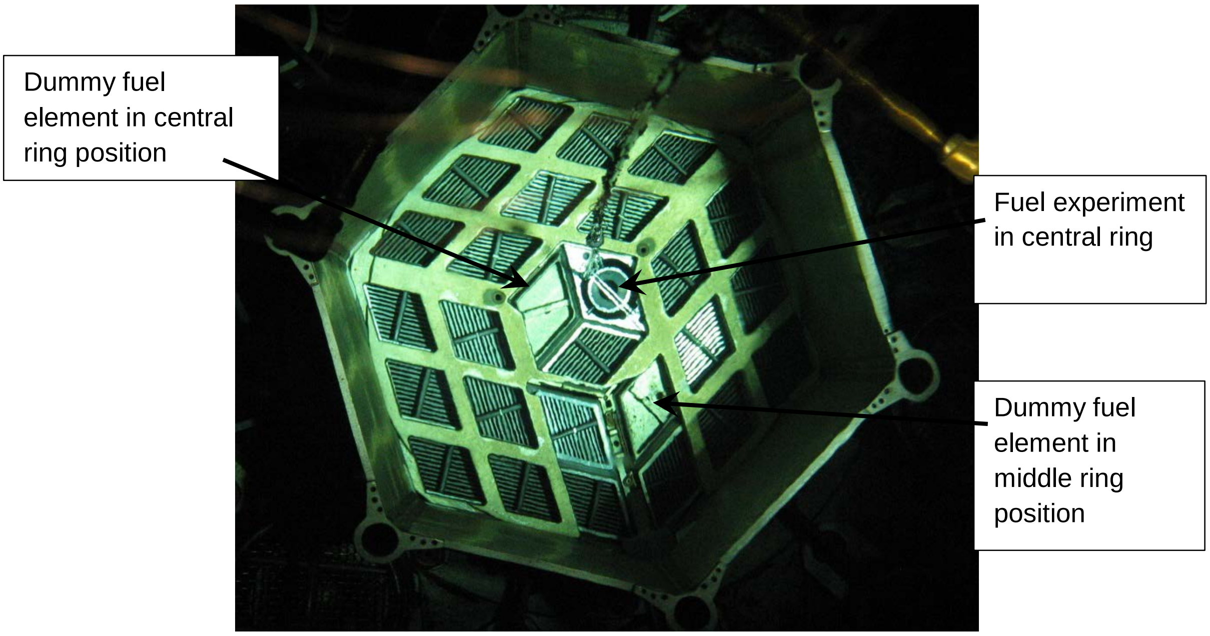

Figure 1. A photograph showing the MITR core with a fuel experiment installed and the other two in-core experimental positions occupied by solid dummy elements as indicated.

2. The in-core water loop

2.1. Facility Description

Circulating water loops at conditions representative of both pressurized water reactors (PWRs) and boiling water reactors (BWRs) have been operated in the MITR for a variety of purposes since the late 1980s. The loop currently installed and in use operates at temperatures up to 300 °C and pressure sufficient to prevent boiling at any point in the loop. It is designed for in-core exposure of material samples and although it is not designed for boiling operation, the chemistry can be controlled to simulate BWR, PWR or other chemistry conditions. In Figure 2, two 3-D renderings show the loop facility as installed in the reactor, with the primary flow loop skid, housing the circulating pump and heaters, positioned near the reactor core tank. The loop is installed through the reactor core tank lid into a dedicated dummy fuel element that positions it in the reactor core. It consists of an outer aluminium containment tube with a gas-filled gap to thermally isolate the high-temperature and pressure titanium autoclave. An external flow loop outside the reactor core tank consisting of a circulating pump and heater provides high temperature coolant which circulates over the samples positioned in core. (See

Section 2.2, below, for a description of one type of sample fixturing and the associated in-core-tank flow path.) Note that, in operation, the primary loop components and tubing that are outside the reactor core tank must be shielded with approximately 10 cm of lead to reduce the fields associated with ^{16}N activity in the circulating coolant. Coolant temperatures and pressures, reactor power and coolant chemistry parameters such as dissolved gas content are monitored and logged.

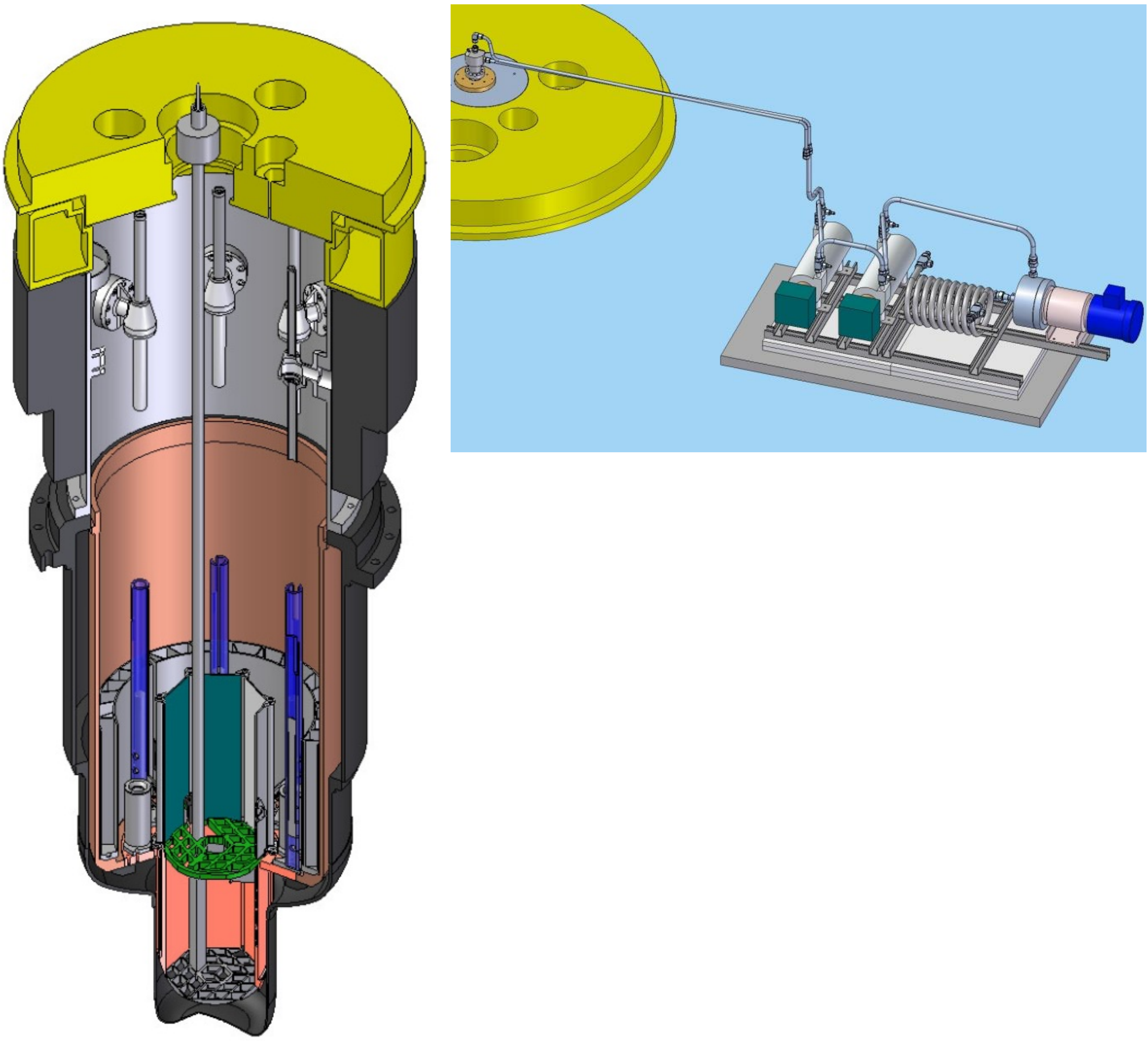

Figure 2. Configuration of the in-core water loop in the reactor. On the left is the in-core containing thimble extending from the reactor top lid to the in-core position. On the right is the main circulating pump, dual heaters and regenerative heat exchanger positioned near the reactor core tank.

A fraction of the circulating coolant is let down from the loop through regenerative and non-regenerative heat exchangers to a clean-up and control system (not shown in the figure). Typically, the main coolant flow rate is about 0.1 kg/s, while the cleanup flow rate is about 0.01 kg/s. Note that, when possible, in-core flow areas are sized to provide prototypical flow rates at the in-core sample surfaces. As noted above, the water loop has generally been operated in either PWR or BWR chemistry mode. In PWR mode, hydrogen cover gas in the coolant storage tank is used to provide approximately 20 cc/kg dissolved hydrogen and lithium hydroxide and boric acid are added at the levels specified for a particular test. In BWR mode, either hydrogen water chemistry (HWC) or normal water chemistry (NWC) conditions can be produced by controlling the hydrogen levels in the coolant storage tank. This is usually accomplished by using a helium/hydrogen mixture as the cover gas and adjusting the cover gas pressure for fine control. Dissolved hydrogen and oxygen are measured on the loop letdown and on the coolant makeup tank. Conductivity is also measured at these points and pH is monitored in PWR mode as an indication of the boron and lithium levels. Batch samples of the loop coolant are taken periodically for gamma spectroscopy and other chemistry measurements. Other chemistries can also be provided as necessary. Past experiments have included zinc injection under PWR conditions and chemical injection for 16N carryover control under BWR conditions. Loop pressure and the letdown flow rate is controlled by a positive displacement charging pump in conjunction with a backpressure regulator. An auxiliary pressurizer is used to avoid boiling in the loop if the main charging flow is lost.

2.2 SiC/SiC Composite Cladding Tests

The major test program in the water loop for the past several years has been investigation of the corrosion behaviour and mechanical property response of SiC/SiC composite tubing under irradiation. This tubing has been proposed as a replacement for Zircaloy™ cladding material for LWR service as part of a fuel system that will tolerate beyond design basis accident conditons. For these tests, the water loop is configured to position sets of samples in the in-core region. Typically, these samples are in the form of either tubes or flat coupons. Figure 3 shows the in-core configuration for a test that is currently in the reactor as part of a Westinghouse-led, DOE-funded program on accident tolerant fuel development. The test is scheduled for 230 full-power reactor days of operation, with an interim sample examination and changeout of some samples at the midway point. SiC/SiC composites have also been proposed for use in BWR channel boxes. Samples from a prototype SiC/SiC channel box were irradiated in the water loop as corrosion coupons and in passively loaded configurations to measure the creep response.

Extensive discussion of the results of these experiments is beyond the scope of this paper, but it should be noted that this work has contributed to the improvement of SiC/SiC composite tubes for cladding applications and the demonstration of viability of the best performing materials for up to 2-3 years of accumulated exposure in PWR conditions. Irradiation has been found to affect corrosion rates, mechanical behaviour, and the survival of a number of proposed bond methods. The BWR channel box material was found to have high corrosion rates under irradiation with dissolved oxygen levels representative of BWR NWC. Further investigation is underway to determine if the material will survive at lower oxygen levels. Coatings to protect the SiC in high oxygen conditions are also being investigated.

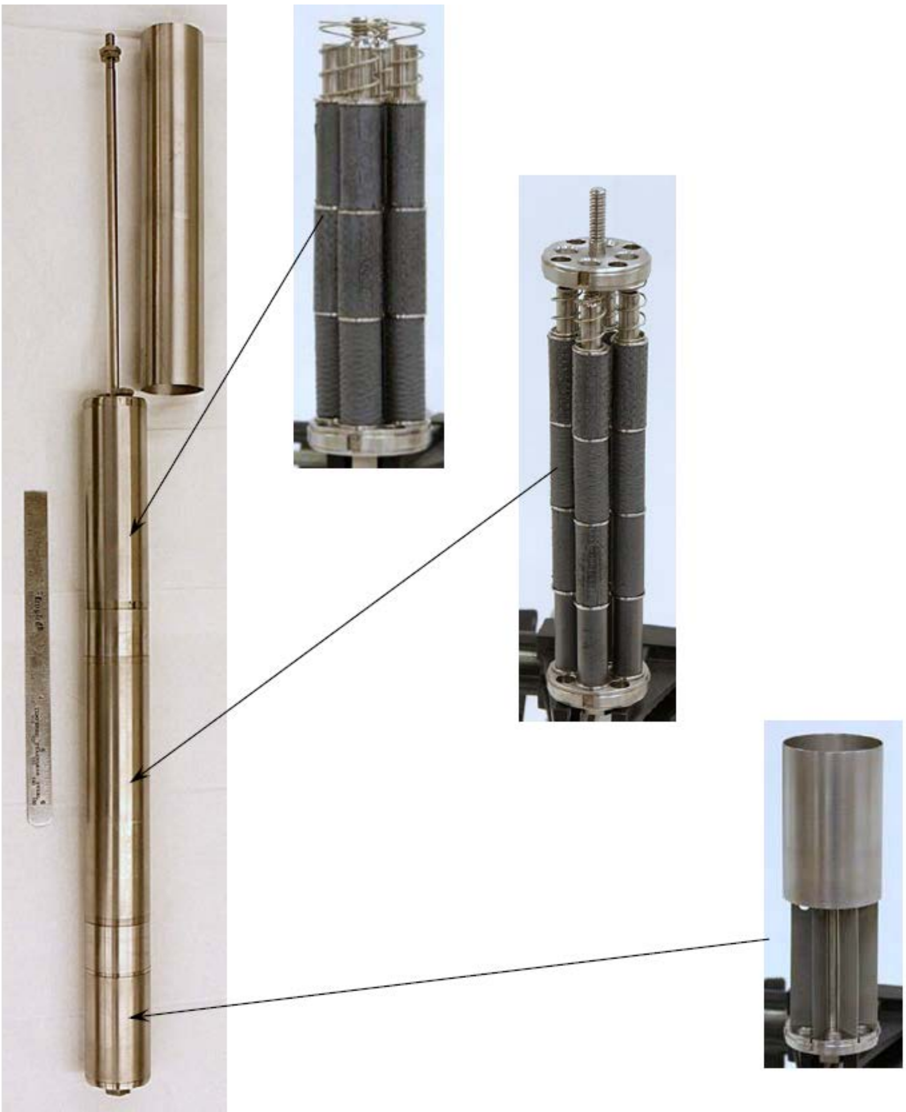

Figure 3. In-core configuration for the Westinghouse Accident Tolerant Fuel irradiation. The sample stack shown on the left is composed of the three sample capsules as indicated.

3. Controlled temperature inert gas irradiation facility

3.1. Facility Description

Like the water loop facility, this facility is installed in core using a dedicated dummy fuel element with a cylindrical opening to position the in-core section. It consists primarily of a titanium tube of approximately 5 cm diameter that extends from the core position to the gas space at the top of the reactor core tank as shown in Figure 4. The tube itself does not penetrate the core tank boundary but gas lines and thermocouples (and other types of instrumentation connections, if necessary) are fed through the upper section of the reactor core tank. The in-core sample assembly (ICSA) tube has an S-bend to prevent radiation streaming from the core. Samples are loaded and unloaded through the flanged upper closure of the ICSA and can be loaded from or unloaded into a shielded transfer cask if necessary. The current ICSA is designed for high temperature operation based on nuclear heating of the sample capsules and additional “susceptor” material (generally titanium) if necessary. By adjusting the capsule mass and the width of the gas gap between the sample capsules and the cooled thimble wall, design temperatures ranging up to the approved maximum temperature of 850 °C can be achieved. By adjusting the proportions of helium and neon in the mixed cover gas, temperature can be controlled over a range of several hundred degrees for typical capsules at full reactor power. Typically, a constant flow rate of helium is used with the neon flow rate varied under feedback control to maintain a desired temperature.

A wide range of sample types and sizes can be accommodated within the constraints imposed by the thermal design and the requirement that sample capsules pass through the S-bend in the ICSA tube to reach the in-core space. A common approach to sample fixturing design is to use reactor grade graphite encapsulated in titanium as a sample holder. By ensuring good thermal contact between the samples and the graphite, uniform temperatures throughout the sample volume can be achieved. Graphite is also a low activation material and it is easily capable of withstanding temperatures up to the maximum approved for the facility. A more detailed description of the capsule design for an experiment currently underway is given in the following section.

3.2 High Temperature Inert Gas Facility Utilization

A variety of experiments have been conducted using the high temperature ICSA facility. Most of these have been instrumented with thermocouples and have used feedback temperature control as described above. After the initial demonstration of the temperature control capability, the first set of sample irradiations were to investigate the irradiation response of MAX phases materials - layered hexagonal carbides and nitrides developed and produced by Barsoum and co-workers at Drexel University. These materials possess attractive combinations of the properties of metals and ceramics and are of interest for high temperature reactor application. Four sample capsules containing three sample types for each of seven different materials were irradiated simultaneously. By adjusting the capsule masses and gas gaps, two of the capsules were operated at approximately 300 °C, while the other two were operated at approximately 700 °C. Two of the capsules were removed and replaced with new capsules containing fresh samples in order to produce an overall test matrix with three different irradiation times up to a maximum of approximately 9 months in-core exposure. Samples from these irradiations were shipped to Savannah River National Laboratory for post irradiation examination (PIE). Two one-month irradiations of fiber-optic sensors manufactured by Luna Innovations, Inc. were also carried out. Each irradiation had a single capsule containing six sensors. The first irradiation operated at 700 °C and the second at 800 °C. The sensors were shipped to Ohio State University for PIE.

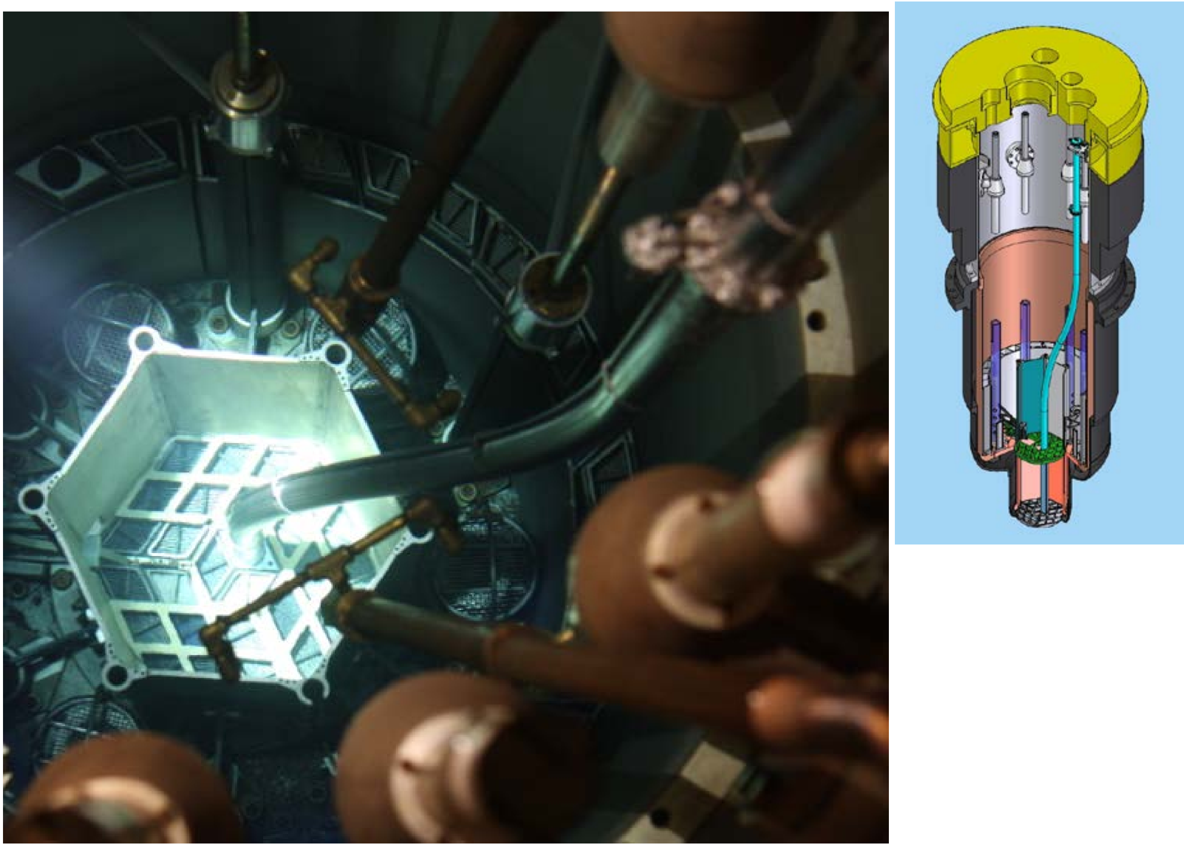

Figure 4. A photograph (left) and 3-D rendering (right) of the In-Core Sample Assembly facility installed in the MITR.

Recently, more highly customized irradiation capsules have been used in the facility. A 1000-hour irradiation of "flibe" (66.7% 7LiF – 33.3% BeF2 salt) at 700 °C was completed last year. This irradiation was carried out as part of a joint program between MIT, University of California, Berkeley and University of Wisconsin, Madison to develop a fluoride-salt-cooled, high temperature reactor (FHR). The sample capsule used is shown in Figure 5. In this case, nickel is used as the capsule material because it is far more resistant to attack by flibe than titanium. Small samples of various materials proposed for use in the FHR were placed in the graphite holder seen in Figure 5 and then each sample cavity was filled with molten flibe. Two of the sample cavities were lined with metal alloys to investigate corrosion effects in the absence of graphite. Following irradiation, the capsule was disassembled in a hot cell within the MITR containment building for transfer to a glove box to melt the flibe (melting point 456 °C) and extract the samples. Segmenting the capsule into three sections reduced the dose rates at the glove box by permitting the handling of only one third of the samples at a time. Another unusual design feature of the flibe irradiation was that the capsule was sealed and provided with gas inlet and outlet tubes to allow a helium sweep of the capsule, independent of the helium/neon gas environment in the ICSA. This feature was implemented to study the behaviour of tritium, produced in the flibe salt primarily by neutron capture in ^{6}Li, despite the

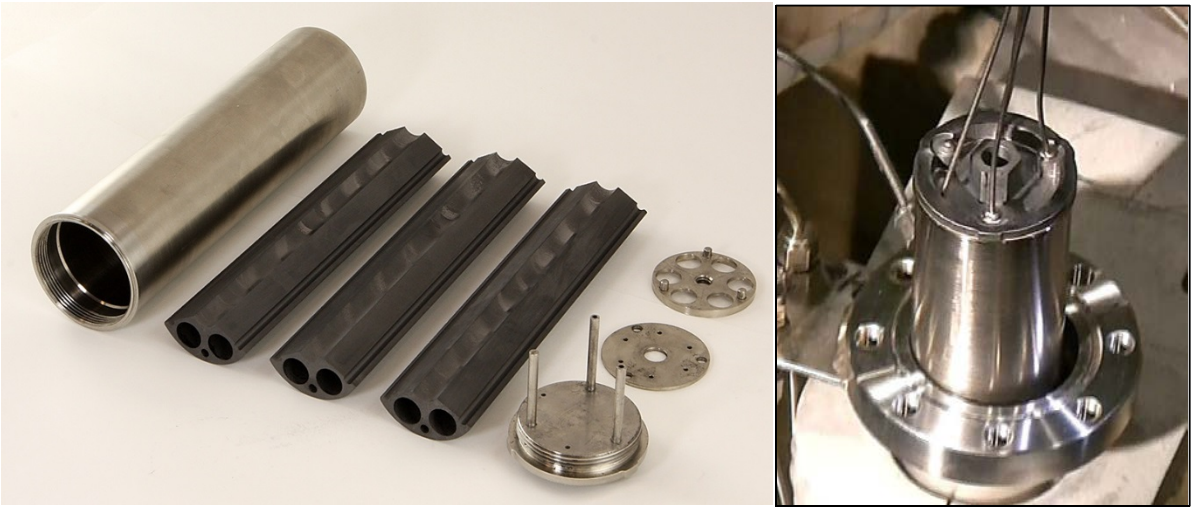

Figure 5. The components of the flibe irradiation capsule (left) and the fully assembled capsule as it was being inserted into the irradiation facility (right). The sample cavities in the graphite segments contained material samples covered with flibe. Two thermocouples and inlet and outlet gas tubes were sealed into the capsule lid and can be seen in the right-hand photograph.

fact that the flibe is enriched to 99.99% ^{7}Li. Tritium production and transport is an important issue for FHR design. The independent capsule gas sweep, together with tritium instrumentation in the off-gas system, permitted separate measurements of the tritium released above the molten salt and the tritium that was transported through the graphite and the nickel capsule wall and released into the main system gas stream. Unfortunately, after about 200 h of operation the gas outlet tube from the capsule clogged, possibly due to the solidification of BeF_2 evolving from the molten flibe, and measurement of the direct release of tritium was no longer possible.

The irradiation program currently underway in the high-temperature ICSA is studying the response of ultrasonic transducers to irradiation at temperatures up to approximately 450 °C. This program is funded by the US DOE under the ATR NSUF and is led by a team from Pennsylvania State University, with contributions from Idaho National Laboratory and other national laboratories supported by the Nuclear Energy Enabling Technologies Advanced Sensors and Instrumentation (NEET ASI) program. Ultrasonic transducers that can operate in reactor environments can be used to make sensors to measure a variety of parameters of interest in irradiation experiments or for monitoring purposes in operating power reactors. The irradiation capsule for the ultrasonic transducer experiment is similar to the “standard” ICSA design with a graphite sample holder encapsulated in a titanium shell. The unusual feature of this design is that there are ten instrument leads coming from the capsule, of which eight are coaxial. These leads allow real-time monitoring of four piezoelectric transducers, two magnetostrictive transducers, two self-powered detectors (one vanadium emitter and one platinum emitter) and two thermocouples. In addition to the instrumented sensors, the capsule contains a set of melt wires for confirmation of the peak temperature, flux wires and a set of “drop-in” samples of the piezoelectric and magnetostrictive materials. These drop-in samples will be used for PIE measurements of the transducer properties to avoid having to disassemble the sensors in the hot cell.

4. Special Purpose In-core Facilities

4.1. General Facility Characteristics

By their nature, these facilities have relatively few general characteristics. However, the facilities designed and built to date fall into two general categories: fuel irradiations and special purpose inert gas irradiations. A fuel irradiation facility is described in some detail in the following section, followed by a brief description of two special purpose inert gas irradiations.

4.2 The Hydride Fuel Irradiation Facility (HYFI)

By definition, research reactors in the US are not permitted to irradiate test fuel in independently cooled loops. Therefore, all fuel irradiations at the MITR are required to reject heat passively to the reactor primary coolant. Additionally, under the terms of a license amendment obtained in 2003 permitting the use of a limited amount of fissile materials (less than 100 gm of U-235 or equivalent) in in-core experimental facilities, all fissile material must be doubly encapsulated for irradiation. These constraints have led to a design approach using an outer encapsulation in contact with the reactor primary coolant at approximately 50 °C, with a heat-transfer gap containing lead-bismuth eutectic (LBE) in contact with the sample fuel cladding. This allows the cladding to operate at prototypical temperatures and fulfils the double-encapsulation requirement.

A fuel irradiation funded by the NSUF and led by researchers from University of California, Berkeley illustrates these principles. The fuel samples and irradiation capsules for the HYFI irradiation are illustrated in Figure 6. Uranium zirconium hydride fuel pellets, cut and ground from a TRIGA fuel compact, were encapsulated in Zircaloy™ cladding with a flanged top closure to accommodate a thermocouple. Each sample to be irradiated was then sealed into a titanium capsule as shown, with LBE filling the gap between the fuel sample and the inner diameter of the capsule. The relatively thick wall of the titanium capsule was chosen to ensure that the LBE was fully melted at the normal operating condition of the capsule. A previous experiment had shown that this is desirable in order to maintain consistent heat transfer through the LBE and capsule and thus maintain stable temperatures. Each capsule had a gas sampling line attached, which also served as a conduit for the central thermocouple and an additional thermocouple on the clad surface (not shown in the illustrations). The capsule cover gas was sampled periodically for fission products in order to determine if a clad failure had occurred. If the rate of fission gas release increased rapidly, indicating possible fuel clad failure, the capsule was removed from the reactor. A stack of three samples capsules (or solid dummy capsules, if necessary) was irradiated in an in-core position in a dummy fuel element with provision for MITR primary coolant flow on the outer surfaces of the sample capsules. The capsules could be independently removed and replaced during reactor outages.

Several fuel samples were irradiated for periods up to several months, with linear heating rates similar to proposed peak rod conditions. Two of the samples were removed from operation when increasing fission gas concentrations in the sampled covered gas indicated failure of the sample cladding. Mechanical problems with the irradiation capsules eventually led to early termination of the experiment. The irradiated fuel samples were recently shipped to PNNL for PIE.

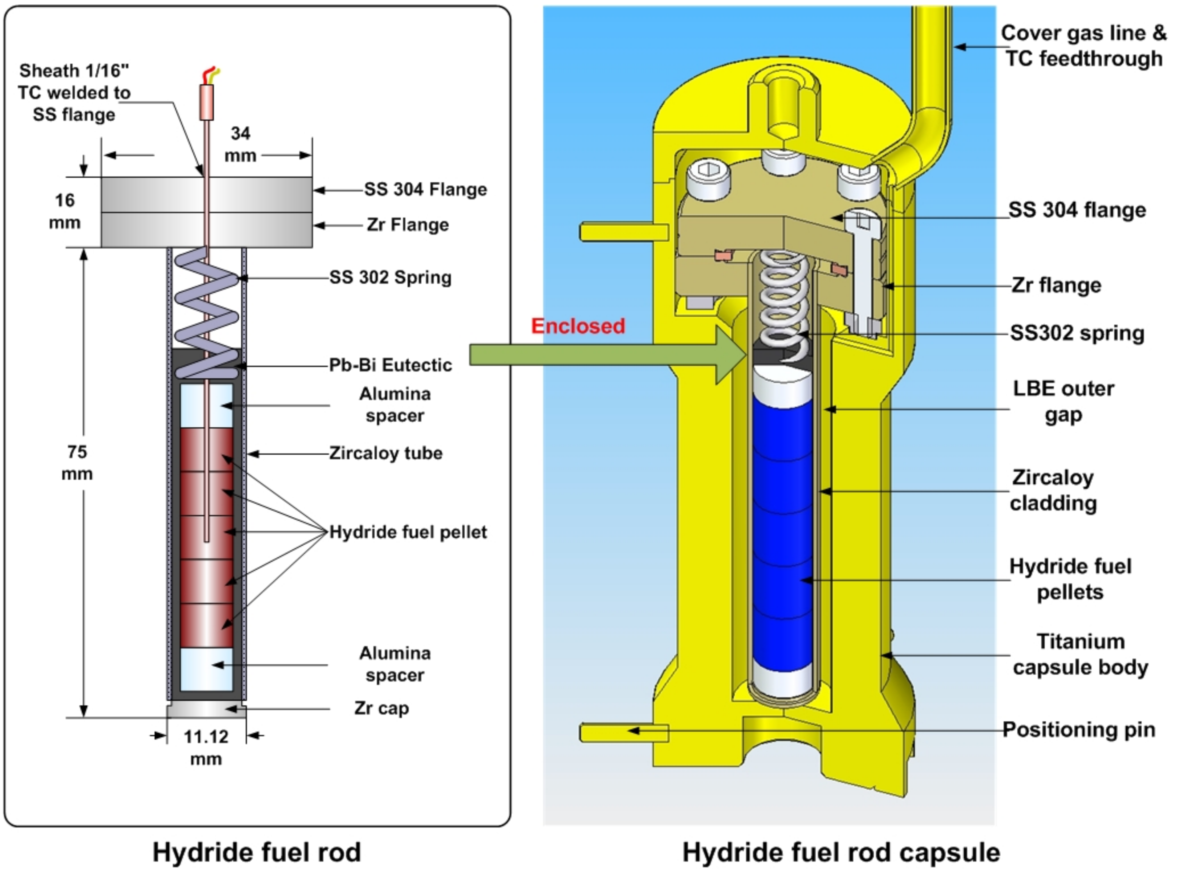

Figure 6. The HYFI irradiation fuel sample (left) and irradiation capsule (right). The fuel samples were instrumented with a central thermocouple and an additional thermocouple on the clad outer surface (not shown).

4.3 Special Purpose Inert Gas Irradiations

Often, the motivation for constructing a special purpose facility for an inert gas irradiation is to avoid the geometric constraints associated with the ICSA facility, in particular, the limitation on the length of capsule that will pass through the S-bend. The first special purpose facility was designed to operate at very high temperatures, up to 1400 °C, on nuclear heating and helium/neon cover gas for temperature control. This required a very high density of nuclear heating material to increase the volumetric heat generation rate, and reflective shields to reduce the radiative cooling losses. The high density was achieved using tungsten sample holders as seen in Figure 7. In order to partially counteract the large negative reactivity of the tungsten holders, the dummy fuel element used for this irradiation was designed to increase the water volume on the surface of the element without increasing the flow channel width. These requirements would have been difficult or impossible to meet using a high temperature ICSA capsule, and the high post-irradiation activity of the tungsten holders would have precluded handling in the ICSA system transfer cask. The very high temperature facility was designed for storage within the reactor core tank to allow for several months of decay between the end of irradiation and transfer of the system out-of-core for disassembly.

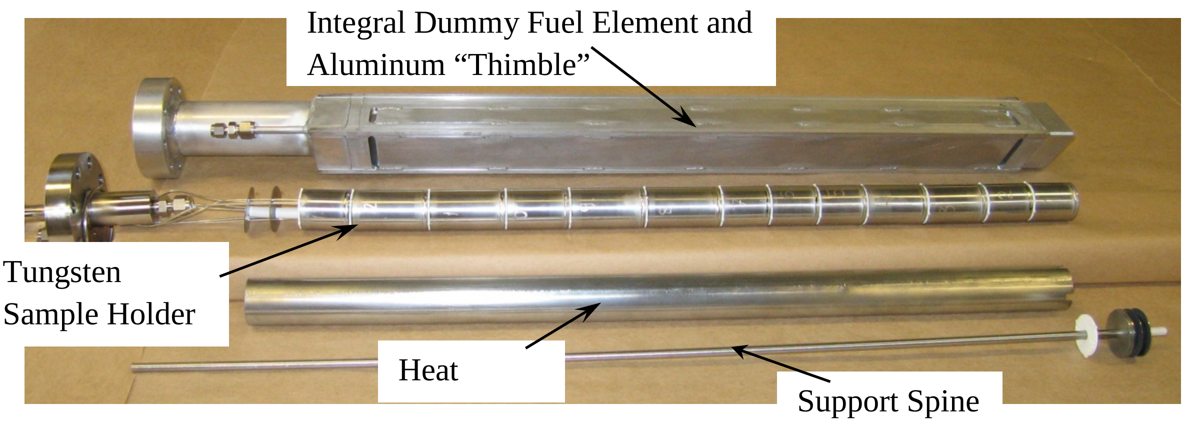

Figure 7. Components of the very high temperature irradiation system. From the bottom of the dummy element to the flange is approximately 75 cm.

The most recently irradiated special purpose inert gas facility was a second test of materials in molten flibe, similar to the ICSA flibe irradiation described above. Again, a major motivating factor for the use of a special purpose facility was to allow for a longer irradiation and larger volume capsule. In this case, the capsule contained two stacked graphite sample holders and accommodated about 300% of the volume of flibe that was irradiated in the high temperature ICSA. A number of other design features, such as a larger gas outlet tube to avoid clogging and a "fast sampling" tube to measure short half-life activities were also implemented.

5. Summary

There is an active program of in-core irradiations at the MITR that makes significant contributions to the study of materials and fuels for current generation and proposed next generation reactors. The neutron flux and spectrum are similar to those of a light water power reactor, and long-term irradiations accumulating significant damage are routinely performed. Test environments available include prototypical PWR and BWR coolant conditions, other water coolant environments, inert gas irradiations up to 850 °C in capsule irradiations and up to 1400 °C in special purpose facilities, and special environments such as molten salt. A wide range of on-line instrumentation is used, including thermocouples, self-powered neutron and gamma detectors, and ultrasonic transducers. Fuel can be tested in small quantities at LWR-relevant power densities and temperatures. Irradiated experiments are transferred to on-site hot-cells for disassembly and PIE. When necessary, radioactive samples can also be shipped off-site for specialized PIE.

6. Acknowledgements

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