Review of Transient Testing of Fast Reactor Fuels in the Transient REActor Test Facility (TREAT)

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Abstract. The restart of the Transient REActor Test (TREAT) facility provides a unique opportunity to engage the fast reactor fuels community to reinitiate in-pile experimental safety studies. Historically, the TREAT facility played a critical role in characterizing the behavior of both metal and oxide fast reactor fuels under off-normal conditions, irradiating hundreds of fuel pins to support fast reactor fuel development programs. The resulting test data has provided validation for a multitude of fuel performance and severe accident analysis computer codes. This paper will provide a review of the historical database of TREAT experiments including experiment design, instrumentation, test objectives, and salient findings. Additionally, the paper will provide an introduction to the current and future experiment plans of the U.S. transient testing program at TREAT.

Key Words: in-pile testing, transient, fuel performance, TREAT.

1. Introduction

The Transient REActor Test (TREAT) test facility located at Idaho National Laboratory (INL) is currently undergoing a resumption of operations by 2018 in preparation for developing and future transient testing experiment programs [1]. Operated by Argonne National Laboratory (ANL) at the time, the facility executed 6604 reactor startups and 2885 transient irradiations while operating from 1958 to 1994, when it was placed in operational standby. TREAT has a rich history of experimentation describing off-normal reactor conditions. The unique flexibility of the facility has allowed for supporting wide variety of experiment goals. Over approximately 35 years of operation, more than 900 experiments were performed targeting a wide range of technical issues. Primary issues addressed by a variety of experimental programs include phenomenology of fuels and materials, transient performance, operational safety, accident consequences, and validation of analytical and computational models. The primary mission of the TREAT facility comprising most of its operation was supporting sodium-cooled fast reactor (SFR) programs as the principal in-pile test facility in the U.S. In this mission, TREAT experiments supported several reactor programs including the Experimental Breeder Reactor-II (EBR-II), Fermi I, the Fast Test Flux Facility (FFTF), Clinch River Breeder Reactor (CRBR), the Southwest Experimental Fast Oxide Reactor (SEFOR), the Integral Fast Reactor (IFR) as well as the Power Burst Facility (PBF) and other water reactor and space power programs.

The focus of this paper is an overview of the historical SFR testing mission of TREAT and an introduction to a renewed program. The organization of the paper is to provide an introduction to transient reactor conditions of interest for in-pile experimental programs followed by summaries of the TREAT facility and primary historical experimental infrastructure. General overviews of the historical experimental campaigns for oxide and metal fuel tests then follow. Finally, an introduction to current and future TREAT experiments supporting SFR.

2. Transient Conditions and Supporting Experimental Approaches

Although other factors may be important to fuel damage, cladding temperature during reactor transients plays a key role in damage assessment as most cladding failure mechanisms are strongly temperature dependent. The cladding temperature is characteristic of the transient event which are classified as Anticipated Operational Occurrences (AOO), Design Basis Accidents (DBA), and Beyond Design Basis Accidents (BDBA). AOO events are expected to occur at least once in the lifetime of a reactor while accidents are not expected to occur but are theoretically possible. Within-design-basis events are of strong interest in the design and development stages through licensing. In SFRs, common DBA events include Transient Overpower (TOP), Loss-of-Coolant Flow (LOF), and Loss-of-Heat-Sink (LOHS) accidents.

A special class of BDBA events include Anticipated Transient Without Scram (ATWS), where automatic scram systems are assumed to fail and only passive reactivity feedback effects. Although the probability of occurrence of BDBA is very low, ATWS events have been of significant interest in fast reactor safety. Inherent safety mechanisms can be used to prevent or mitigate serious potential outcomes. Still, the threat that the consequences of very low-probability BDBA events poses to public health and safety drives this interest. The main concerns being: the SFR core is not in its most reactive configuration, the large fission product and plutonium inventory available, and the large volume of liquid sodium. Generic ATWS events that have been the focus of study are double fault events including the Unprotected Transient Overpower (UTOP), the Unprotected Loss-of-Coolant Flow (ULOF), and the Unprotected Loss-of-Heat-Sink (ULOHS) accidents [2]. In all cases, which cover a wide range of thermal conditions, experimental programs are needed to understand the impact of associated temperature histories on cladding damage accumulation and margins.

Reactor transient testing and much of the historical TREAT database may be divided into two categories distinguished by the point of cladding breach. Operational transient testing focuses on behavior during anticipated transients up to the point of cladding breach. Reactor safety testing studies the cause and nature of cladding rupture and post-failure fuel behavior during accident conditions [3]. Transient testing experiment programs may span into both categories of testing. For operational transient testing used to support FFTF and CRBR safety programs, the general objectives were to: 1) demonstrate core component performance allowed by the plant protective system (PPS), 2) establish the margin between PPS capability and component failure, and 3) confirm validity of design procedures (criteria, methods, etc.) [4][5]. The U.S. Department of Energy (DOE) and Power Reactor and Nuclear Fuel Development Corporation (PNC) (Japan) Operational Reliability Testing (ORT) Program was carried out in the 1980s for oxide fuel largely using the EBR-II and TREAT facilities [3].

Historically, both in-pile and out-of-pile experimental facilities were developed to cover the range of experimental conditions required by the spectra of off-normal/transient events.

- Several out-of-pile test facilities were established to support transient testing. Separate effects facilities included the Fuel Cladding Transient Tester (FCTT) for stress rupture behavior of oxide fuel claddings and the Fuel Behavior Test Apparatus (FBTA) for penetration of cladding due to fuel-cladding chemical interactions in metallic fuels. To understand the synergistic roles of fission-gas pressure and fuel-cladding chemical interactions the Whole-Pin Furnace (WPF) system was developed and installed in the

Alpha-Gamma Hot Cell Facility at ANL [6][7]. The WPF allowed for testing heating rates up to 30 °C/s and peak temperatures of 1100 °C for times of several days. This facility has since been fully decommissioned but similar systems are currently being developed at INL. Many other such out-of-pile facilities including severe accident simulators existed (some are still available) across the U.S.

- Operational transient testing was performed in EBR-II at the former ANL-West site in Idaho (present-day INL Materials Fuels Complex (MFC)) in addition to extensive steady-state irradiation testing [8]. Transient testing programs included the Operational Reliability Transient (ORT) program for operational transient testing of oxide fuel using the metal driver fuel of EBR-II [9] and the Shutdown Heat Removal Test (SHRT) program, which successfully demonstrated the ability of the metal-fueled fast reactor to survive unprotected ULOF and ULOHS without core damage [10]. In the overpower transients, test ramp rates covered ranges of 0.1 to 10% $\Delta P/P_0$ per second up to 100% overpower. Peak temperature achieved in these tests were 890 °C. The upper level of transient rates available in EBR-II overlapped the lower bound of possible ramp rates in TREAT.

- The Sodium Loop Safety Facility (SLSF) in the Engineering Test Reactor (ETR) was designed to address whole core accidents to address issues related to reactor design, licensing, and operation. The ETR provided prototypic fuel conditioning and operation for up to 35 days. The SLSF provided prototypic thermal-hydraulic conditions for up to 37-pin test configurations. A total of seven experiments were performed in SLSF, six of which, had the purpose of whole core accident simulation. The final experiment investigated a local fault event of propagation dynamics accompanying molten fuel release into the coolant [11].

- For intense, shorter duration transients, the TREAT facility was used to perform transients covering a wide range of events from mild transients to hypothetical core-disruptive accidents (BDBA). A description of the TREAT facility and capabilities follows in the next section.

3. Overview of the TREAT Facility

3.1 Reactor Facility

The TREAT reactor is an air-cooled reactor with fuel composed of graphite a dilute dispersion of UO_2. The core is 1.22 m long with 10 cm x 10 cm driver fuel assemblies laid out on a 19x19 configurable array. The test region is typically provided in the center of the core by the removal of one or more of the central fuel assemblies. Direct line of sight view of the center experiment position is provided by four experiment slots around the core. Historically these slots were utilized by a variety of diagnostic devices included high speed video apparatus. Currently two of these slots are occupied a neutron radiography facility and a fast-neutron hodoscope for on-line detection of experiment fuel motion [12].

Several special features make the TREAT a unique, flexible and cost-effective test facility:

- Being an open, air-cooled (not a safety function), heat-capacity machine allows for easy, close-in access to experiments and special diagnostics. These features allowed the majority of experiment designs to be tailored for rapid insertion and removal, which proved to provide rapid-turnaround, high-throughput rate of experiments, and cost effective utilization. Important for future utilization, this design allows for rapid transition between experiments supporting radically different boundary conditions (e.g. water ⟵⟶ sodium).

- A programmable, automatic reactor control system allows for a multitude of transient power prescriptions. Being a heat capacity reactor, the transient is limited by the amount of energy the core is capable of absorbing. The maximum energy available for a transient is 2500 MJ with a peak power of approximately 20 GW. The duration of operation can range from fractions of a second to several tens of minutes. Steady-state operation is limited to 100 kW. The experimental approach utilizes a "time window" simulation to capture issues and behavior of interest for time periods corresponding to the behavior of interest. Thermal pre-conditioning may be accomplished using preheat and/or as part of the initial phase of a power transient. Irradiation preconditioning is performed in steady-state reactor facilities [13]. An additional noteworthy feature of the reactor control system is the capability of allowing triggers from experiment instrumentation to drive reactor operation. This approach was used to initiate power bursts upon experiment coolant void detection or to signal reactor scram at clad failure to discontinue experiment propagation and "freeze" experiment conditions.

- The proximity to support facilities provides well-established capabilities for experiment design; modeling and simulation; assembly/disassembly; instrumentation development assembly, and testing; pre- and post-irradiation characterization; fuel design and fabrication. The INL MFC is located hundreds of meters from TREAT with the Hot Fuel Examination Facility (HFEF), Irradiated Materials Characterization Laboratory (IMCL), and numerous fabrication and other supporting facilities. The Advanced Test Reactor (ATR) steady-state irradiation facility is also located only several kilometers from these facilities.

3.2 Historical Experimental Facilities

As described previously, the flexibility of the TREAT-experiment interface allowed for a multitude of experimental devices to be used to support a wide range of reactor programs. In general, the complexity of experimental devices evolved with the program understanding and needs. Early in experimental programs, testing objectives focused on basic phenomenology in simple capsules which were inexpensive and quick. As program priorities and technologies developed, the need for increasingly prototypic reactor conditions drove experiment design to more complex designs that were informed by the simpler testing results and developed modeling approaches.

Summaries of the experiment devices used to support SFR testing in TREAT are found in [14][15]. Early TREAT experiments were performed on single-pin metallic fuel pins in opaque dry capsules using thermocouples for instrumentation [16] Dry capsules were instrumented to detect time of cladding breach using electrical continuity between cladding and a surrounding sheath [17]. A capsule with transparent window was soon designed and used to capture high-speed video (~4000 frames/s) of cladding failure and fuel dispersal, which was later modified to allow testing of pre-irradiated fuels [18]. The next logical progression of test configuration was a stagnant sodium autoclave that was pre-heated to 260°C prior to transient irradiation. In addition to thermocouples, a pressure transducer was included in the capsule. Also in these tests neutron absorbers were used to shape the axial power profile. In the late 1960s, concerns over sodium coolant pressure pulses resulting from refractory fuel-coolant interactions drove the need to develop a high-pressure sodium autoclave. This device was instrumented with a pressure transducer, thermocouples, and a linear motion detector to measure the expulsion of a piston above the sodium level during high energy transient events. With a system design pressure of 41.4 MPa, the maximum pressure pulse experienced out of 8 tests performed was 20.2 MPa with a calculated nuclear-mechanical energy conversion of approximately 0.2%. This test deposited 2130 J/g-UO_2 while the maximum energy deposited in any test was 3020 J/g-UO_2 [19].

By the mid-1960s, objectives progressed to require more realistic coolant conditions through the foundational design of an integral sodium loop. Though capsule testing remained important throughout the lifetime of the reactor test programs, the integral loop design provided the basis of the majority of the high impact testing that followed over the next two decades. The Mark I loop provided operating temperatures of 400 °C and was designed to allow 7-pin clusters of EBR-II fuel [20]. A total of 10 tests were performed before the program moved onto a higher-pressure rated loop, the Mark-II, enabled by the Annular Linear Induction Pump (ALIP) [21]. The Mark-II loop design represented a mature design but was limited to fuel pin lengths corresponding to EBR-II pins. In the late 1960's through the 1970's, 24 tests were performed in Mark-II loops.

In the early 1970s, accident model development had progressed to the point of requiring integral experiments to understand the combined effects of sodium boiling, fission-gas release, cladding melting and relocation, and fuel motion on full-length pins. A gas-pressure-driven, once-through, two-tank sodium system was designed to accommodate full-length fuel pins and provide full scale hydraulic resistance through all stages of simulated loss of flow. The Mark-II loops did not have these capacities, however, the once-through system only allowed fresh fuel tests where the loops allowed preirradiated and mixed-oxide fuel. The once-through system housed 1 or 7-pin test assemblies and enough sodium to provide up to two minutes of flow in the 7-pin configuration. Instrumentation included thermocouples, pressure transducers, permanent-magnet flowmeters, and voltage taps along the test train to detect sodium boiling [22]. In 1977, the ~tenth and final test performed in this system was designed to provide feedback from the loop inlet flowmeter to the reactor control computer to initiate a scram upon detection of fuel pin failure. This mode of operation would become important for many of the more advanced test programs to follow.

In the early 1980s, full-active-length fast-irradiated fuel became available from FFTF and PFR. The Mark-III loop was created as a "stretched" modification of the Mark-II loop. It was designed to accommodate 7-pin bundles with a version configured for the bottom-plenum design of PFR fuel and a version for the top-plenum designs of the U.S. The Mark-III loop represents the flagship experimental device for performing transient irradiations in TREAT and has been used extensively in the sophisticated experimental programs carried out in the 1980s on both oxide and metallic fuels. The device was designed to provide prototypic sodium flow rates and fuel-bundle pressure drops [23]. Loop instrumentation typically included many thermocouples, multiple pressure transducers and multiple electromagnetic flowmeters. Acoustic sensors for detecting cladding rupture and coolant void sensors were also used in some loop tests.

In the progression of experiment facilities described, fuel motion diagnostics has remained a key feature of nearly all fuel testing. The first years of testing employed a high-speed video system, which was limited to only image surface features. Quickly, experimentation progressed to use liquid sodium and development led to the fast-neutron hodoscope which allowed measurement of fuel motion by detection of collimated fast neutrons and gamma rays emitted from the test fuel specimens in an opaque environment [12]. The fast-neutron hodoscope played a critical role in evaluating fuel motion during transient events since the mid-1960s. The fuel motion occurring prior to fuel failure and post-failure was some of the most valuable data obtained in TREAT tests for evaluating reactivity effects on reactor design safety and providing validation of hypothetical core disruptive accidents [24][25].

Other test devices were deployed and/or designed in great detail for TREAT installation to support other technologies. One noteworthy design was part of the TREAT upgrade project, which would allow prototypic thermal-hydraulic testing of up to 37 pin bundles of irradiated fuel. Examples of these devices include: devices supporting Light Water Reactor (LWR) research: Standard Temperature and Pressure (STP) opaque water capsules, STP high-speed-video water capsule [26], a flowing steam system for the Source Term Experiments Program (STEP) [27], a steam recirculating loop to support LOCA studies in Heavy Water Reactor systems was designed late in the program; a helium loop system, the Gas Reactor In-pile Safety Test (GRIST-2) project was designed to support safety studies for gas-cooled fast reactors [28]; etc.

4. Historical SFR Fuels Experiments

Over the course of more than 30 years, TREAT performed an expansive array of SFR fuel experiments following the design evolutions of both metallic, oxide, and some carbide fuels. In-pile reactor safety testing studies on oxide fuel have been carried out in various test reactors around the world including extensive studies performed in TREAT. For metallic fuel, in-pile severe accident studies have mainly been performed in TREAT. Summaries of the last decade of experiments performed in TREAT, most relevant to modern fuel systems, are found in [15][30]. Along with [15], a description of the evolution of TREAT fuel experiments since its beginnings is found in [31].

5. Brief Introduction to Future Testing in TREAT

With the resumption of operations at the TREAT facility nearing completion, experimental facilities and supporting infrastructure are well under development. In conjunction with hardware and facility renewal, transient testing experiment programs are also being developed to support a variety of reactor technologies. The design strategy for experimental devices is based on the successful package-type configurations of the past. Early emphasis of experimental programs is on capabilities supporting LWR fuel systems under the DOE Accident Tolerant Fuels (ATF) program. For this purpose, a high-pressure capsule stack (x4) is being developed that will allow testing of four isolated 15 cm long rodlets of LWR fuel at Pressurized Water Reactor (PWR) conditions (300 °C and 15.5 MPa). The overall design of the capsule assembly is based on the package-type configuration of most of the experiment hardware of the past. Pre-conceptual designs for longer single capsule systems and flowing water loops are also being pursued with the intention of supporting Reactivity Initiated Accident (RIA) and Loss-of-Coolant Accident (LOCA) studies [32].

Revitalization of the historical sodium testing capabilities of the past has also begun. The focus of these efforts are on: 1) capsules allowing for testing in dry and stagnant sodium environments, 2) a modern version of an already advanced Mark-III sodium loop, the Mark-IV loop, which is expected to leverage the successful legacy of these devices, while incorporating advancements available in modern instrumentation and data collection. Of course, the design could also potentially incorporate features needed to support specific contemporary reactor technologies.

One of the most important distinctions of the future program from historical testing programs is the significant advances in computation that have occurred in the past decades. Of course, an extensive database of SFR safety research exists after more than 60 years of safety research on SFR systems worldwide and that experience will help guide future testing priorities. Even so, continued improvement of SFR safety and performance will rely on the increasingly detailed and accurate predictive capability. As it has in the past, progression towards this goal will require complementary experiment programs (both in-pile and out-of-pile) and model and simulation development.

6. Summary and Conclusions

The TREAT facility has played a crucial role in the development of modern SFR technologies. Three decades of testing in the reactor have served a multitude of built and operated SFRs as well as several advanced design programs. The reactor design provides relatively spectacular access for experimentation and instrumentation to achieve a wide range of experiment objectives. The ability of specific experimental capabilities to adapt to support changing program needs was proven to be nimble and efficient through a variety of test devices and experimental goals.

Current LWR fuels testing programs are in full design of transient experiments on novel fuel designs. Supporting hardware and infrastructure are being put in place to provide the foundation of future testing programs to support a variety of LWR, SFR, and other advanced reactor technologies. The transient testing experiment strategy is following the historical success of package-type loop configurations including revitalization of the Mark-series loop.

The restart of TREAT comes at an opportune time when in-pile facilities and resources have become scarce in much of the world. The facility design provides unique flexibility and capabilities that will allow it to serve a multitude of contemporary and advanced reactor designs and play a critical role in future fuel development and reactor safety programs.

7. References

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