The TREAT Experiment Legacy Supporting LWR Fuel Technology

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ABSTRACT

The Transient Reactor Test (TREAT) facility has served a major role in the development and understanding of nuclear fuel technologies. Recently restarted, the TREAT facility operated for 35 years supporting a wide range of science and technology experimental objectives prior to being placed in standby in 1994. Early in its mission, TREAT performed an expansive array of experiments focused on nuclear driven, high temperature material interactions. Numerous fuel types were tested in a variety coolant conditions. Many experiments were performed on oxide fuels in early TREAT testing in a variety of claddings, water and sodium environments, including high speed video testing. As SPERT/PBF/LOFT testing programs grew, TREAT's dominant mission to support Fast Breeder Reactor program testing also expanded. During the remaining 25 years of TREAT's first life, a variety of additional experiments were planned and executed to support LWR technologies. For LWR fuels, the focus on rapid overpower behaviors (Reactivity Initiated Accident, RIA) shifted to include Loss-of-Coolant-Accident (LOCA) and fission product release experimental studies. This paper presents a review of historical TREAT experiments supporting LWR technologies and summarizes major experimental outcomes. With TREAT now restored to operation, a new generation of fuel safety testing will be introduced that will support current and next generation fuel technologies.

KEYWORDS: *fuel safety research, reactivity initiated accident, loss of coolant accident, transient testing, TREAT*

1. Introduction

Fission-heated fuel safety research experiments have played a fundamental role in the long history of nuclear fuel development, qualification, refinement, and licensing. Several experimental programs have been devised and executed since the 1950's to study the off-normal behavior of nuclear fuels and materials. Starting largely with whole reactor experiments simulating transient behaviors, the experimental philosophy shifted to develop into experimental facilities testing fuel in capsules or loops. The Transient Reactor Test (TREAT) facility represents the first of the latter type of test facilities, where the reactor is designed specifically to perform experiments on fuel experiments placed into the reactor. The unique design of the facility provided very rapid evaluation of high power and energy behaviors of wide-ranging nuclear fuel forms and compositions under various coolant environments, uniquely looking at behaviors ranging up to melting or vaporization of materials.

The reactor operated from 1959 to 1994 when it was place in operational standby. During that time, it addressed wide-ranging research needs, with primary research programs addressing phenomenology of fuels and materials, transient performance, operational safety, accident consequences, and validation of analytical and computational models. During its first decade of operation TREAT provided a strong, ground breaking portfolio of experiments that supported LWR technology. Soon, growing liquid-metal fast reactor (LMFR) programs (i.e. EBR-II, FFTF, CRBR, IFR) programs [1]) combined with new (and in some cases very advanced) LWR capabilities in the Special Power Excursion Reactor Test Capsule Driver Core (SPERT CDC), Power Burst Facility (PBF), and Loss Of Flow Test (LOFT) facility, phased the LWR mission out of the TREAT facility to a large extent. Still, experiments in the 1970's and 1980's leveraged the flexibility of the facility to take advantage of periods of expedited experimental needs and gaps in availability of other facilities to perform significant experimental campaigns, again paving the path towards programs planned in other facilities.

Current nuclear technology has been founded through an extensive experiment history around the world. Many large programs and facilities existed in the U.S. to support testing of nuclear fuels and materials. The need to capture crucial data from these programs has long been desired by the research communities. As part of the effort to restart the TREAT facility, an experimental database was created in an attempt to preserve the historical record from the first 35 years of the facility's life. The database has been made available to select U.S. research organizations with the intent to make more broadly available, though many general release is hampered by the need to review a large number of documents for public release. The database represents an important step to ensure preservation of costly, and in some cases, likely never repeatable experimental data. Current discussion and planning is still ongoing for a more encompassing strategy to specifically capture the significant fuels and materials irradiation testing database from U.S. transient testing facilities at TREAT, SPERT, LOFT, PBF, etc. The TREAT database may provide the base to continue expanding to include these other important experimental programs. Currently limited funding is available to expand the TREAT database. As part of the current, U.S. "TREAT" fuel safety research program, a significant collection of historical experimental documents and data from other Idaho based programs has been generated along with significant knowledge transfer across a generational gap in U.S. research programs. Certainly, significant challenges remain for this information to be more accessible and of value more broadly.

The historical record is crucial for understanding the safety and performance of modern technologies, since it is foundational to our knowledge base. With the development of new nuclear fuels, materials, and reactor designs, the historical record (including less known tests) is also invaluable to formulating ideal strategies licensing that must rely on an experimental data basis. In other words, understanding how we arrived at the current situation for modern fuels with existing fuel safety limits and outstanding issues (e.g. Fuel Fragmentation Relocation Dispersal (FFRD), transient CHF behaviors, etc.) is extremely important to begin studying revolutionary fuel systems (e.g. U_3Si_2 fuel in SiC cladding) that begin from fundamental behavioral investigations.

With the recent restart of the TREAT facility, this paper focuses on the vast capabilities of the TREAT facility by highlighting significant experimental campaigns carried out during the 30+ years of operations in the first life of the TREAT facility. It concludes by highlighting the current activities centered on LWR testing in TREAT (specifically Accident Tolerant Fuels (ATF)) and an overview of the strategy to support broad testing needs of modern R&D for current LWR, advanced fuels, and advanced reactors. The first fuel-bearing experiments in a generation will be begin in the TREAT facility just prior to this conference.

2. Overview of the TREAT Facility

The TREAT facility was engineered specifically as a flexible reactor facility for performing reactor safety experiments involving fuel disruption (up to and including fuel melting and vaporization). The core test region is provided by removal of one or more of the 10-cm-square fuel assemblies. There is convenient top, bottom, and lateral access to this test region. TREAT is an air-cooled transient reactor with a nominal core energy release capability in excess of 2,500 MJ, roughly an order of magnitude greater than that of most other transient reactors. Each TREAT transient can be programmed to be delivered over a preselected time. TREAT can, for example, deliver this energy in a short-duration burst or it can be set up to run at about one megawatt for nearly an hour; also it can be run steady state at 100 kW. The reactor core is 1.2 m high (same as the Advanced Test Reactor also at Idaho National Laboratory). TREAT has a neutron radiography facility for before-and-after radiography without requiring over-the-road shipping to other facilities. In addition, the facility-integral TREAT hodoscope provides an in-situ fuel motion monitoring capability such that fuel movements can be evaluated before any experiment handling is performed after a test. This is a particularly valuable feature for experiments in which test fuel may be disrupted because the sample might be disturbed significantly by handling unless it is "frozen" in-situ (practical with sodium but not with water). Axial distributions of test fuel and fission products can be measured by gamma scanning of the entire length of the experiment package during withdrawal from the reactor.

3. Historical Experiments

As previously highlighted, the major contribution to nuclear energy production R&D from the TREAT facility was found in extensive testing of oxide and metallic fuels for sodium-cooled reactor designs [1]. However several experimental campaigns were carried out that provided foundational experimental conclusions and, arguably, set the direction for experimental facilities and test campaigns even to today. The LWR experiments performed in the TREAT facility demonstrate an important component of an ideal experimental approach to evaluating fuel behavior. The TREAT LWR experiments were largely of the integral-type with the purpose of phenomena and failure mode investigations that led more extensive separate effects testing. In all cases, more elaborate integral scale testing followed in other facilities around the world that provide the core basis for accident simulation validation.

3.1. RIA Experiments

During the first 10 years of TREAT's life, hundreds of experiments were performed to understand nuclear heated interactions of nuclear fuels, materials, and coolants. Primary relevant LWR experiments are reviewed in detail in [2][3]. In addition to the experiments described in those references with greatest relevance to the LWR industry, many experiments were performed in water under RIA power conditions on aluminum-plate type fuels for research applications as well as important experiments supporting the development and qualification of the fuel for the Power Burst Facility (PBF). The PBF experiments utilized advanced instrumentation techniques to measure fuel and cladding elongation, cladding temperature, and coolant pressure and the test strategy focused the behavior of a unique advanced fuel design on cyclic RIA power cycling (up to 100 repetitions on a single rod). These tests included margin to PBF design evaluations, defected cladding tests, water-logged rods over a wide range of target energy depositions. Examples of post-irradiation metallographs taken from the PBF-TREAT experiments are shown in Fig. 1. The PBF reactor was uniquely designed for a specific mission to drive fuel safety experiments. The fuel rods were composed of ternary ceramic fuel pellets (20.6 UO_2, 61.8% ZrO_2, 7.6% CaO), a ceramic thermal insulator between the pellets and cladding, and a stainless steel cladding tube.

In sum, these experiments provided foundational data for the LWR industry. They also were first-of-a-kind in experimental approach to reactor safety research. The CEN experiments, in addition to data from experiments at the Special Power Excursion Reactor Test (SPERT) facility, provided the early basis for regulatory requirements for energy deposition during RIA events in LWRs for fresh and low burnup fuel [4].

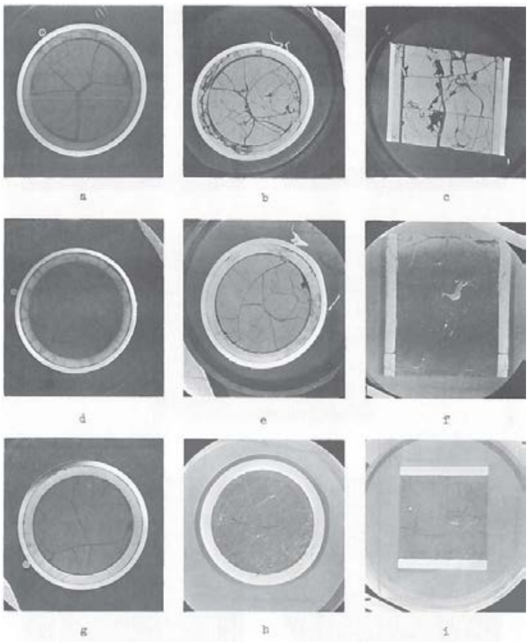

Fig. 1. Images of PBF prototype rods after transient irradiation in the TREAT facility.

3.2. Fission Product Release (FPR) Experiments

In the early 1960’s, a series of four experiments were conducted to study the release of fission products from fuel during a reactor accident in which the fuel is destroyed by fast transient melting. These tests marked some of the foundational approaches to such studies that evolved over the next several decades at facilities around the world. Fig. 2 shows diagrams representing the experiment design. These experiments called the Fission Product Release Experiments focused on (1) a clean reactor core where the only fission products available for release are those formed during the transient and (2) specimens with tracer-level irradiation to build up an inventory of fission products. In both cases the atmosphere of the experiment was an oxidizing mixture of air and steam. The fuel used in these experiments was UO_2 in stainless steel cladding. The specimens were electrically preheated to 800°C and subjected to transient power histories with a ~0.5s duration natural pulse shape to achieve fuel melting. Cladding temperatures were measured by thermocouples. In these experiments, no differences were obseved due to variations in oxidizing atmospheres which was attributed to a large fraction of oxygen in the autoclaves. The transient release of aged fission products from UO_2 was less than the release of fission products formed during the transient because of the tracer level preirradiation effects and higher volatility of recently born fission products. One primary conclusion was that the pattern of release was similar compared to related slow heating experiments but that the fast transient cases resulted in less release. Approximately a third of volatile fission products were released from the fuel but near 1% left the autoclave. These experiments represented some of the first in-pile capsule experiments to explore these phenomena and were carried out complementing out-of-pile and slower in-pile experiments in other reactor facilities. Detailed description of these tests are found in [5].

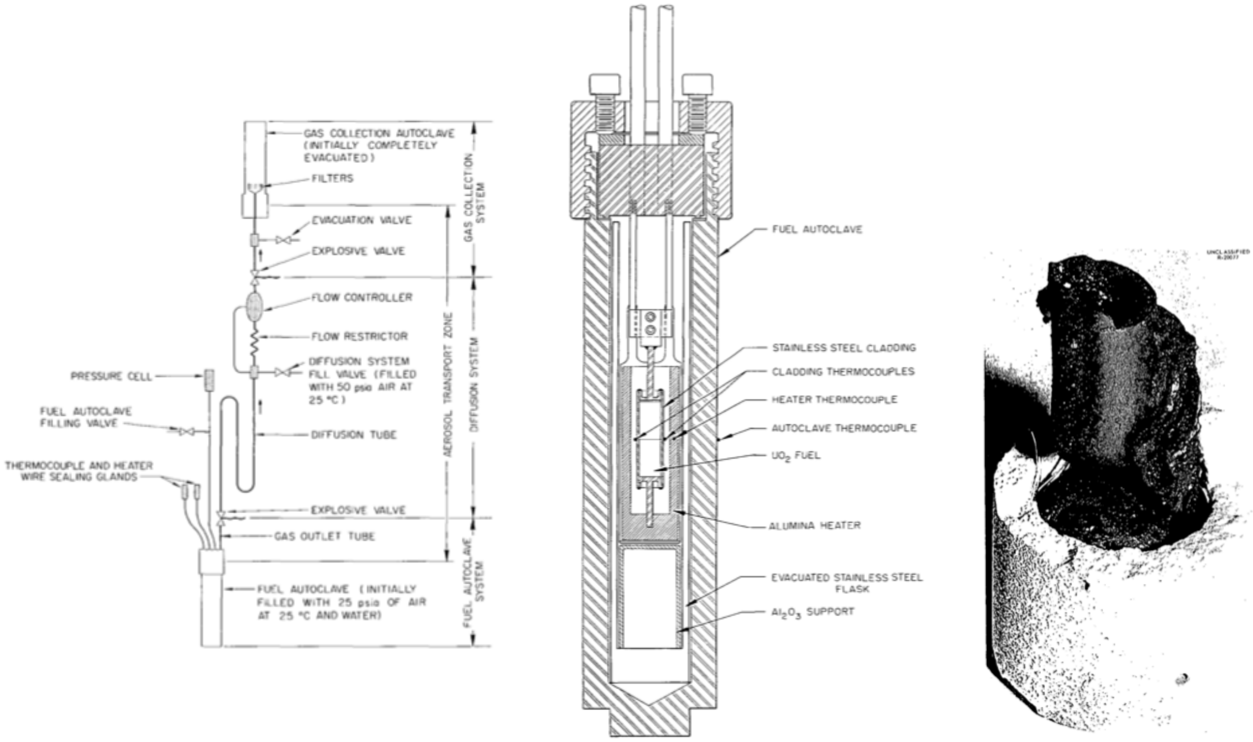

Fig. 2. Test apparatus used for the fission product release experiments, (left) experiment flow diagram, (middle) autoclave diagram, lower fuel region from experiment 3 showing hollowed out melted UO2 [5].

3.3. Fuel Rod Failure Experiments (LOCA)

The Fuel Rod Failure (FRF) experiment series was performed in TREAT in 1970 to study the behavior of Zircaloy-4-clad UO_2 fuel rods during LOCA in LWRs. Two experiments were carried out to investigate three primary objectives: (1) dimensional changes in the fuel rods possibility to impair emergency core coolability, (2) the extent of cladding oxidation and implications on core integrity, (3) amount and chemical form of released fission products. These TREAT experiments simulated the decay heat in post-blowdown events of a LOCA. The experiments used a cluster of seven 27 inch fuel rods. The center rod was preirradiated to produce a fission product inventory and irradiation effects in the cladding and fuel. These experiments represented some of the earliest integral LOCA experiments ever performed, and along with later PHEBUS and NRU tests, the only tests ever done with fuel bundles, which also included preirradiated fuel rods. Detailed description of the test results are found in laboratory reports [6][7] and published in [8].

The tests were carried out using a flowing steam + helium device. The experiment vessel was placed inside the TREAT reactor while the remaining components were placed on the reactor top or nearby. The flowing steam carried fission products from ruptured rods through a filter system to collect aerosols and iodine. Flow rate was controlled to simulate conditions after blowdown in BWR and PWR LOCA events. The fuel rods were pre-pressurized to simulate fission gas pressure in a BWR and placed in 1500 psi steam at 750°F for two days prior to assembly. Experiment instrumentation included pressure transducers connected to two fuel rods and thermocouples spot-welded to the cladding of four rods. The inner preirradiated rod was inserted into the center position inside 6 fresh fuel rods in a hot cell at the present day INL site. All rods were supported from the top to prevent rotation but allow bowing. In FRF-1, the center rod was preirradiated in the Material Test Reactor (MTR) with a peak linear power of 49.9 kW/m to a peak burnup of 650 MWd/MTU. For FRF-2, the center rod was preirradiated in the MTR and Engineering Test Reactor (ETR) with a peak linear power of 45.6 kw/m to a peak burnup of 2800 MWd/MTU.

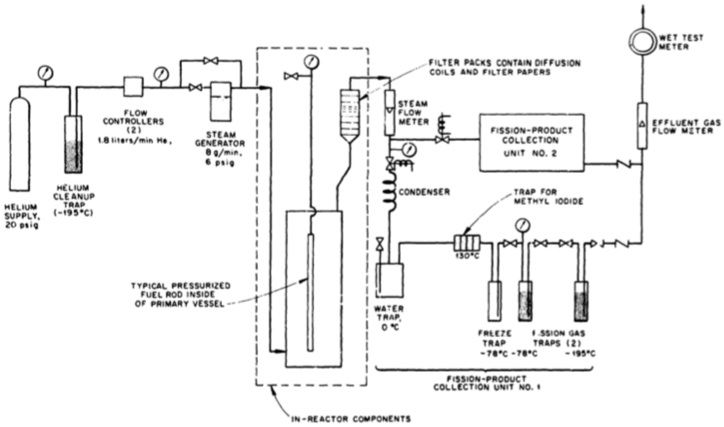

Fig. 3. Schematic overview of the experimental system used for the FRF experiments (from [6][7]).

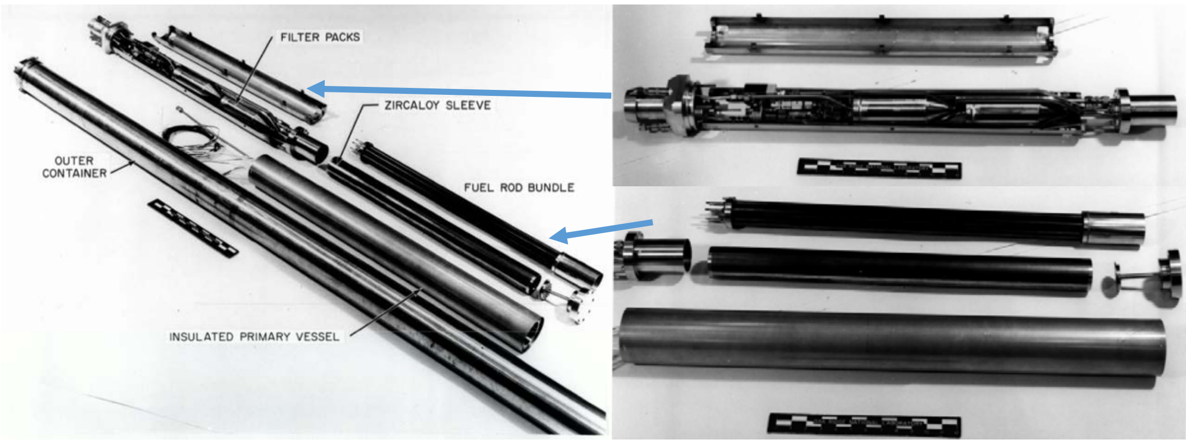

Fig. 4. FRF experiment assembly [6][7]. (Top) TREAT in-core components showing vessel, fuel rod assembly, and filter assembly, (Middle) fuel rod assembly, (Bottom) filter assembly.

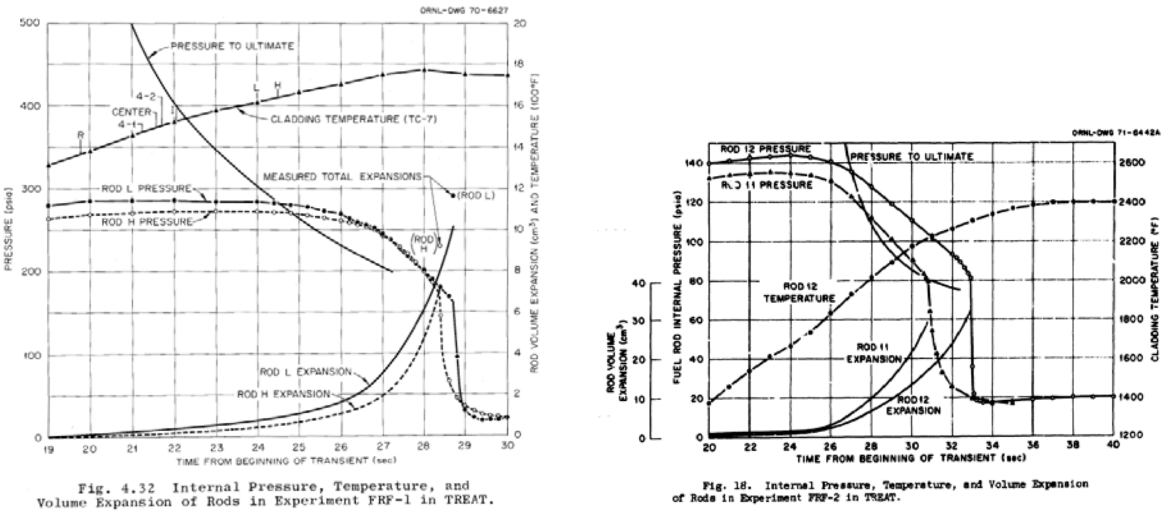

Fig. 5. Experimental pressure, temperature, and rod elongation for the FRF-1 (left) and FRF-2 (right) experiments [6][7].

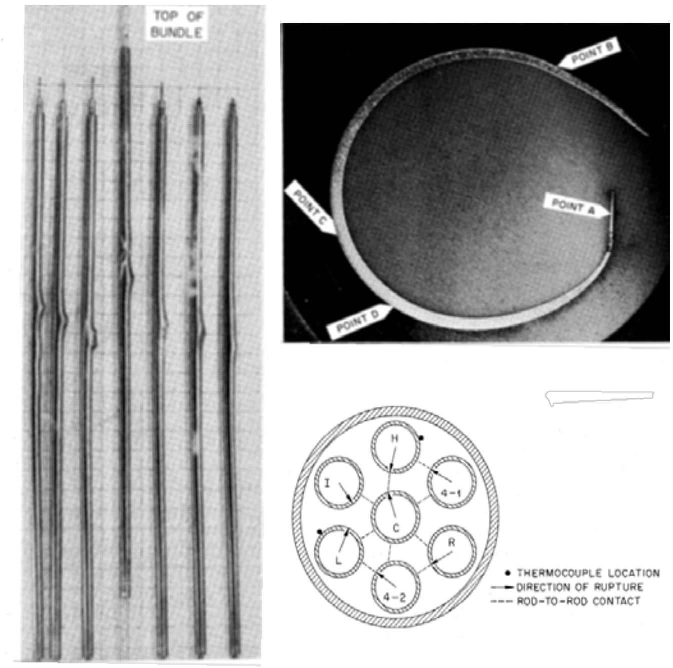

Fig. 6. Results of the FRF experiments [6][7]. (Left) Photographs of the rods showing bowing and ballooning. (Top-Right) Transverse cross section of rod through point of rupture. (Bottom-Right) Schematic top view of bundle showing direction of rupture and rod-to-rod contact [6][7].

In both experiments, the swollen and failed regions of fuel were in similar axial positions, which evidenced the stronger influence of temperature as compared to cladding wall thickness and strength. The tests showed 36% to 60 % average circumferential expansion of the cladding. The mechanical performance of the fuel rods such as maximum expansion and rupture characteristics was in good agreement with out-of-pile tests at that time.

These experiments were successfully carried out as an early leader to following advanced experimental programs that followed looking into LOCA conditions and fission product release. At the time, they were the most realistic simulated LOCA conditions incorporating fission heating in cladding, rod-to-rod (bundle) and preirradiation effects. The FRF experiments were considered proof tests of the data and models derived from other out-of-pile experiments (e.g. tube bursting). The FRF-2 experiment data has been maintained as a validation case for the FRAPTRAN fuel performance code with an input file available in [9]. Follow-on experiments were planned though never executed in the TREAT facility.

3.4. Source Term Evaluation Project (LOCA – Severe Accident Fission Product Release Experiments)

After the accident at Three-Mile Island (TMI), significant interest arose in the nuclear industry for fission-product behavior during severe accidents. In particular, source term studies became a dominant focus for many experimental and modeling R&D projects. These source term studies focused on the type, quantity, and timing of radioactive releases. A large, EPRI-lead international consortium was organized included the U.S. Nuclear Regulatory Commission, the U.S. Department of Energy, Ontario Hydro of Canada, and Belgonucleaire. In particular, data from the TMI event indicated that iodine release was much smaller than predicted by assumptions being used at the time. Therefore, great interest was spurred to answer the question whether LWR plants were being licensed under regulatory assumptions that were too conservative by a large margin. The following review comes primarily from [10][11][12][13][14] and other program documents not currently publically available.

A series of four experiments were conducted at the TREAT facility to part of an EPRI-lead project called the Source Term Experiments Project (STEP). These experiment were to serve a verification role for the many out-of-pile separate effects experiments and modeling efforts of the time to support use in regulatory matters. The TREAT STEP experiments would provide a composite picture of source term behavior by means of an in-pile, integral test of reactor fuel under realistically simulated, accurately controlled, and carefully instrumented conditions. In particular, nuclear heating of the fuel in such a test provides several advantages over alternative means: (1) avoidance of atypical structures and materials; (2) control of the heating process during fuel disruption; and (3) avoidance of atypical hot-spots and electromagnetohydrodynamic forces. These experiments were also viewed as providing expeditious results to guide later, even more elaborate experiments being planned under the program and elsewhere (e.g., the Severe Fuel Damage experiments at the Power Burst Facility). The entire life cycle of the TREAT experiment series was planned and executed aggressively over 2-3 years with the final test occurring in early 1985.

The four test matrix was formed based on two parameters of importance for fission product release, transport, and consequences including time of core uncover and pressure. Four sequences were selected based on predicted PWR (2) and BWR (2) accident predictions to evaluate time and pressure. Ultimately, the high pressure, short time scenario (BWR TW was replaced by a duplicate high pressure, long duration scenario but including the addition of a Ag/In/Cd material to simulate the presence of a control rod in the center of the experiment. Though a range of conditions could have been explored, the preselected time-energy sequence of focus for these experiments was from fuel exposure to steam to fission product release. The experiments were designed to provide data regarding the physicochemical properties, near the point of origin, of the biologically important volatile fission products released early in such accidents.

Each experiment consisted of four pre-irradiated, Zircaloy-4-clad UO_2 fuel pins contained in a pressure vessel connected to a system (with similarity to the FRF tests described above) providing flowing STEAM conditions. The fuel specimens in these tests were selected, in part, based on availability ultimately using specimens irradiated to 30-36 GWd/T average burnup in BR-3 in Belgium with rod average powers of ranging from 219 W/cm to 345 W/cm. The rod out diameter was 9.5 mm with an overall length of 1 m. The STEP tests were instrumented with several temperature, pressure, flow transducers, and two hydrogen detectors based on a hydrogen-permeable, palladium-silver tube. Fission products entrained in the steam flow were collected in three elaborate sampling stations and two aerosol sampling canisters located along the downstream axial length of the test train. Multiple multi-stage filters were located downstream of the canisters in the view of gamma-ray spectrometers to measure accumulated radioactive fission products as a function of time. Filter systems were employed on the main flow outlet stream to capture the bulk of particles exiting the primary vessel.

STEP-1: The first experiment simulated a large-break LOCA with failure of the emergency core cooling system for a PWR with good representation of peak temperatures for 20 min (low pressure with short core uncovering time). Significant cladding oxidation occurred and large amounts of volatile fission products were released from the fuel and measured.

STEP-2: The second experiment simulated loss of flow with failure of high pressure emergency core cooling and long term decay heat removal in a BWR with core uncovery occurring approximately 24 hours later (low pressure with long core uncovering time). Maximum temperatures were below that reached in STEP-1 resulting in lower cladding oxidation and fission product transport from the fuel.

STEP-3: This experiment focused on a station blackout in a PWR with loss of main and auxiliary feedwater systems. The core is exposed after approximately an hour (high pressure with long core uncovering time). The test was ultimately operated at near BWR pressure. Despite similar power and flowrate, enhanced natural convection in this test is assumed to have caused overall fuel temperatures to be lower than predicted and lower cladding oxidization and fission product release resulted compared to the previous experiments.

STEP 4: Ultimately, the final experiment was not selected as a high pressure, short duration scenario. Instead, the experiment was designed to be equivalent to the third experiment with the exception of the addition of a Ag/In/Cd material to simulate a control rod in the center of the experiment. The purpose was to investigate the effect of volatile materials in a control rod on aerosol formation and transport. Similar outcomes to the STEP-3 test resulted likely due to enhanced heat transfer from the fuel.

Peak temperatures in the experiments ranged from 2200 K to 2900 K. Large, favorable pressure differentials across cladding resulted in significant ballooning in the first two tests. High system pressure in the second last tests prevented ballooning. Transient fuel motion occurred during the experiments (primarily from axial thermal expansion) and was measured online using the TREAT hodoscope system in tests 1, 3, and 4. Detailed measurement of Zircaloy-steam reaction are reported and compared to code calculations in the summary reports. Aerosol size distributions and concentrations were measured and relative quantities, morphologies, and chemical characteristics were determined for released fission products including cesium, iodine, tellurium, molybdenum, and rubidium.

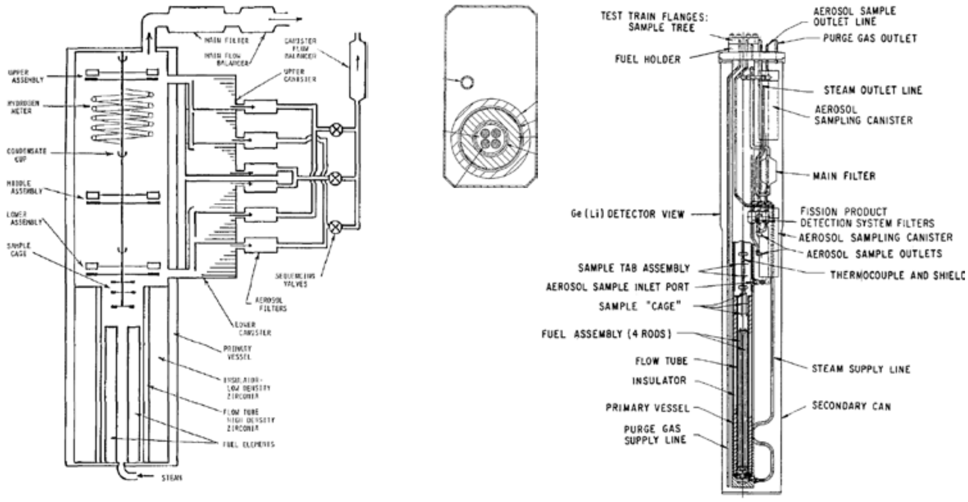

Fig. 7. Overview of the STEP experiments vehicle design [11]. (Left) Schematic of the in-core vehicle plumbing used for fission product detection, (Middle) Transverse cross section of vehicle at reactor center showing specimen layout, (Right) In-core test vehicle.

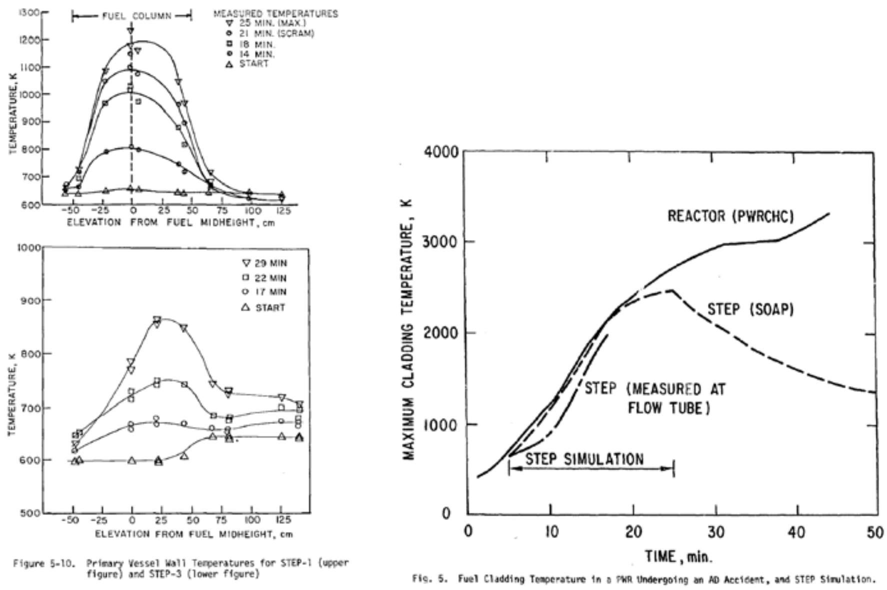

Fig. 8. Experimental temperatures from STEP experiments. (Left-Top&Bottom) Measured flow vessel wall temperatures during STEP-1 (Top) and STEP 4 (Bottom) [11], (Right) Temperatures from STEP-1 compared to simulated large break LOCA with ECC failure [12].

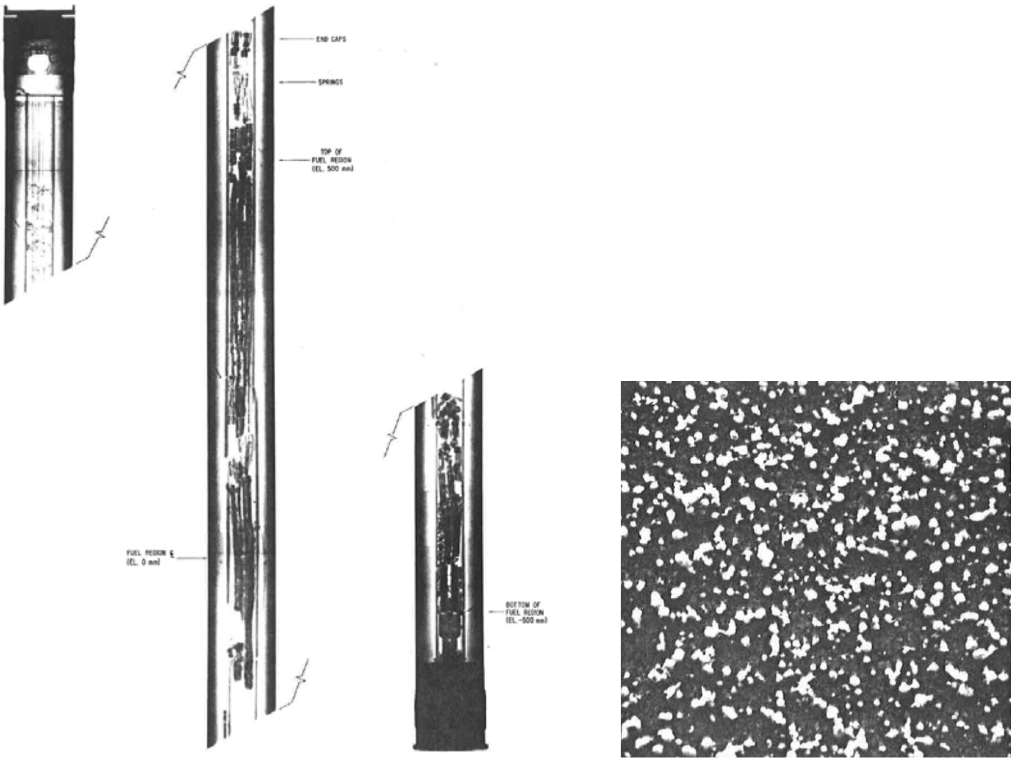

Figure 9. (Left) Post-irradiation neutron radiograph of STEP-1 experiment test train showing disrupted fuel [12], (Right) Example SEM image of particulate used to measure size, quantity, and distribution.

3.5. Related (NPR AN)

During the first lifetime of TREAT experiments, several other experiments were initiated to varying levels of design development. Though terminated early (and not LWR), the New Production Reactor (NPR) program developed a significant test program to support water testing of the NPR for LOCA, RIA and LOF experiments. In the late 1990's, noteworthy proposals were drafted to develop a PWR loop for TREAT and restart the reactor to perform detailed studies on high-burnup LWR fuel [15].

4. Discussion and Future Experiments

Though LWR experiments were not a major component of the last 25 years of TREAT's experimental history, the LWR experiments summarized in this paper were crucial to developing fundamental understanding of LWR fuel behavior under integral-type conditions for RIA, LOCA, and extended LOCA conditions for fission product release behavior. Each of these tests appear to be the first of their kind for many similar testing programs carried out in the years that followed. The integral RIA studies in the TREAT facility were not only the first of a kind, but, along with results from the SPERT program that followed shortly after, first demonstrated the fuel coolability limit still relevant for fresh fuel (UO_2, Zr-based claddings) today. The LOCA tests were the first integral tests to demonstrate bundle effects on fuel swelling under integral conditions and were crucial to validate out-of-pile separate effects testing that formed the initial LOCA safety criteria. In all cases, the experimental approaches used in the tests were quite modern in terms of measurements and overall sophistication. The data is still used for code validation purposes. The experimental designs have been remarkably important to modern experimental designs incorporating water environments and for exploration of future fission product monitoring systems in the facility.

The restart of the TREAT facility has largely been driven to support the needs of the ATF fuels campaign. Thus, significant capability development has focused on needs of ATF fuels, which largely mirror testing needs for LWR fuel technologies, *historical* and modern. The testing needs of revolutionary fuel concepts will require much more focus on historical approaches to identify fuel failure phenomena, development mechanistic failure predictive models, and show compliance with existing fuel safety criteria. A range of capabilities are at various stages of development from dry capsules to heat sink devices that provide well-controlled heat transfer conditions to more prototypic, integral liquid flowing devices [16]. The purpose is to provide a suite of testing platforms to support high throughput, rapid turnaround separate-effects to prototypic integral experiment objectives. Strong emphasis is devoted towards LOCA and RIA simulation but also for examining potential for ramp testing and other power-to-cooling mismatch studies (see paper in this conference on current LOCA developments by KAMERMAN et al.).

First experimental test campaigns have initiated during calendar year 2018. Initial testing will focus on detailed understanding of experiment performance in the TREAT facility including crucial evaluation of reactor-to-fuel-specimen energy coupling. Reactor power simulations of LOCA and RIA transients are currently being performed and evaluated with in-situ flux measurements. The first experiment series bearing fuel specimens are planned to initiate just prior to this conference with Zircaloy-4-clad UO_2 specimens in a dry capsule to provide a baseline of several levels of high temperature fuel behavior to beyond melting conditions. In addition, advanced instrumentation and modeling and simulation tools are being aggressively qualified to support near- and long-term testing objective needs for both current LWR technology and ATF testing needs. These developments will pave the way to first water-based testing beginning next year in static water capsule devices capable of high temperature and pressure environments.

5. Summary and Conclusions

This paper provides a summary of historical experiments performed in the TREAT facility that provided foundational R&D support of LWR technology used by the modern industry. The experimental database spans a wide variety applications with forward-thinking experimental designs and instrumentation strategies. An electronic database has been established for experiments performed at the TREAT facility, which is expected to be publicly available within in the next few years. In addition, the new existence of a fuel safety research program in the U.S. has provided the ongoing consolidation of many historical fuel safety experiment data and documents. The historical LWR experiments performed in the TREAT facility were first-of-their-kind and demonstrated the advantageous flexibility and capability of the facility to accomplish the goals of a modern experimental program. They also show the progression of investigation for modern LWR technology. In some cases preirradiated fuels were tested in these experiments, though typically consisting of quite low burn up specimens, making the tests less relevant to contemporary issues that have strong focus on high burn up fuel behaviors. Still, the TREAT LOCA tests are still used as validation cases for modern fuel performance codes and are still quite unique for having included multiple fuel rods for rod-to-rod ballooning impacts.

The successful experimental designs used historically in the TREAT facility have been invaluable to developing modern capabilities showing great flexibility to perform controlled power experiments in complicated coolant environments. Significant developments are underway in U.S. DOE programs to establish capabilities and test programs in the TREAT facility (and out-of-pile) to support current LWR technologies, advanced LWR fuel concepts including ATF, advanced reactor fuels and designs, and other missions. A key component of the modern fuel safety research program has been and will continue to be leaning on the lessons learned from a rich history of transient testing in Idaho. New transient testing programs will provide continued opportunity to capture the historical database. The first fuel experiments are beginning in the TREAT facility in September 2018; while commissioning of an integral static water capsule will begin at the end of 2019.

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